Table 4-38. Key Aspects and Technical Assumptions in the Human Intrusion Scenario in the TSPA-SR Model

Key Aspect	Associated Technical Assumptions
Infiltration of water down the borehole and into the penetrated waste package	Borehole diameter Flux into borehole based on infiltration, climate, catchment basin focusing Seepage through borehole into penetrated waste package
Mobilization and release of radionuclides from the penetrated waste package	Type of waste package penetrated Potential dissolved and colloidally transported radionuclide inventory Surface area and volume of waste exposed Waste form and cladding degradation Thermal and geochemical conditions in waste package Dissolution rates
Transport of radionuclides down the borehole to the water table	Borehole flow and transport properties including porosity and sorption Borehole length
Transport of radionuclides through the saturated zone	Borehole location Nominal case saturated zone flow and transport properties
Exposure pathways and dose calculation at the receptor location	Nominal case biosphere pathways and properties

For human intrusion in the supplemental and revised supplemental TSPA models, the DOE first determined the earliest time at which the waste packages would degrade sufficiently that a human intrusion could occur without recognition by the driller. Consistent with final NRC licensing-related regulations (10 CFR 63.321 [66 FR 55732]), the DOE estimated that this could not occur until about 30,000 years after closure. The DOE has also analyzed this human intrusion scenario and included the analysis in the final EIS. The results of a human intrusion at 30,000 years are also summarized below in Section 4.4.4.2.2.

4.4.4.2 Results

4.4.4.2.1 TSPA-SR Model Results

The 100-year human intrusion results were calculated probabilistically by the TSPA-SR model. Figure 4-197a shows the mean annual dose rate for 100,000 years resulting from a human intrusion 100 years after repository closure, together with the 95th, 50th (i.e., median), and 5th percentile curves from a set of 300 simulations. The peak mean total expected dose equivalent from human intrusion during the first 10,000 years is approximately 0.008 mrem/yr, and the peak mean total expected dose equivalent for the 100,000 year period of analysis is also approximately 0.008 mrem/yr. TSPA-SR human intrusion scenario performance assessment results are documented in more detail in Section 4.4 of the *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a).

Even though the human intrusion scenario was not evaluated to a million years, the doses would not be expected to be significantly greater than the doses out to 100,000 years because the sorbing, long half-life radionuclides that would start to contribute beyond 100,000 years are not significant contributors to dose. In addition, the million-year human intrusion doses would be less than the nominal case doses.

4.4.4.2.2 Revised Supplemental TSPA-SR Model Results

The first failure of the Alloy 22 waste package material due to general corrosion is projected to occur after approximately 30,000 years. The DOE has, therefore, determined that the earliest time a human intrusion could occur without recognition by a driller (intruder) is at least 30,000 years. This determination was based on analyses presented in Volume 1, Appendix A of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a). Additional information supporting the timing of the human intrusion (at 30,000 years) is presented in *Total System Performance Assessment Sensitivity Analyses for Final Nuclear Regulatory Commission Regulations* (Williams 2001b). The compressive strength and ductility of the metals from which the drip shields and waste packages are fabricated differ significantly from the rock that would surround them. Drillers (intruders) would notice these differences. For example, the drilling assembly would buckle and bend when the bit attempted to penetrate the titanium drip shield and waste package (drill bits that are designed for rock do not easily penetrate metal, particularly titanium). The drillers (intruders) should, therefore, recognize that they have attempted to drill into some material other than rock for at least as long as the drip shield or waste packages are intact. Analyses calculate a 95 percent probability that the waste packages will remain intact for about 30,000 years. Thus, the human intrusion is not expected to occur within the first 10,000 years.

The analysis of the human intrusion scenario at 30,000 years is presented in *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE 2002) and *Total System Performance Assessment—Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain—Input to Final Environmental Impact Statement and Site Suitability Evaluation* (Williams 2001a, Section 6.4). The human intrusion scenario at 30,000 years considers an "intruder" to be someone drilling a land-surface borehole using a drilling apparatus and the common techniques and practices that are currently employed in exploratory drilling for groundwater in the region around Yucca Mountain. In the scenario, the intruder drills directly through a degraded waste package and subsequently into the uppermost aquifer underlying the Yucca Mountain repository. The intrusion then causes the subsequent compromise and release to groundwater of the contaminated waste in the penetrated waste package.

Two human intrusion scenarios were simulated and discussed in *Total System Performance Assessment—Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain—Input to Final Environmental Impact Statement and Site Suitability Evaluation* (Williams 2001a, Section 6.4). One intrusion occurs at 100 years after repository closure, and the other occurs at 30,000 years, the earliest time after disposal that the waste package would degrade sufficiently that a human intrusion could occur without recognition by the drillers. The results of the revised supplemental TSPA model simulations for a human intrusion at 100 years after closure showed a peak mean dose of 0.0048 mrem/yr (at 875 years) over the 10,000-year period. This is similar to the dose projected by the TSPA-SR model (approximately 0.008 mrem/yr) for a 100-year human intrusion (Figure 4-197a). The revised supplemental TSPA model projected a peak mean annual dose over a million-year period of 2.3×10^{-3} mrem/yr for the 30,000-year human intrusion, as shown in Figure 4-197b. The peak mean dose occurs at 108,000 years (Williams 2001a).

The DOE conducted a sensitivity analysis to consider an unlikely event (igneous intrusion) in the evaluation of the human intrusion scenario and documented it in *Total System Performance Assessment Sensitivity Analyses for Final Nuclear Regulatory Commission Regulations* (Williams 2001b, Section 6.2). The report discusses the scenario of a human intrusion preceded by an unlikely igneous intrusion event with a mean annual probability of an igneous intrusion at the location of the repository of 1.6×10^{-8} (CRWMS M&O 2000a, Table 3.10-5). In the analysis, it was determined that the mean annual dose due to a human intrusion following an unlikely igneous intrusion can be approximated by multiplying the conditional dose of a human-intrusion event by the probability of the initiating igneous-intrusion event by the probability of the drillers not detecting the waste package (assumed to be equal to one if the drilling is preceded by an igneous intrusion event).

The conditional human intrusion dose would be a function of when the initiating igneous intrusion occurs. The worst case would be if the intrusion occurs immediately following the loss of institutional controls. This calculation was presented in *Total System Performance Assessment—Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain—Input to Final Environmental Impact Statement and Site Suitability Evaluation* (Williams 2001a) for a human intrusion at 100 years, and the resultant maximum mean dose was 4.8×10^{-3} mrem/yr (Williams

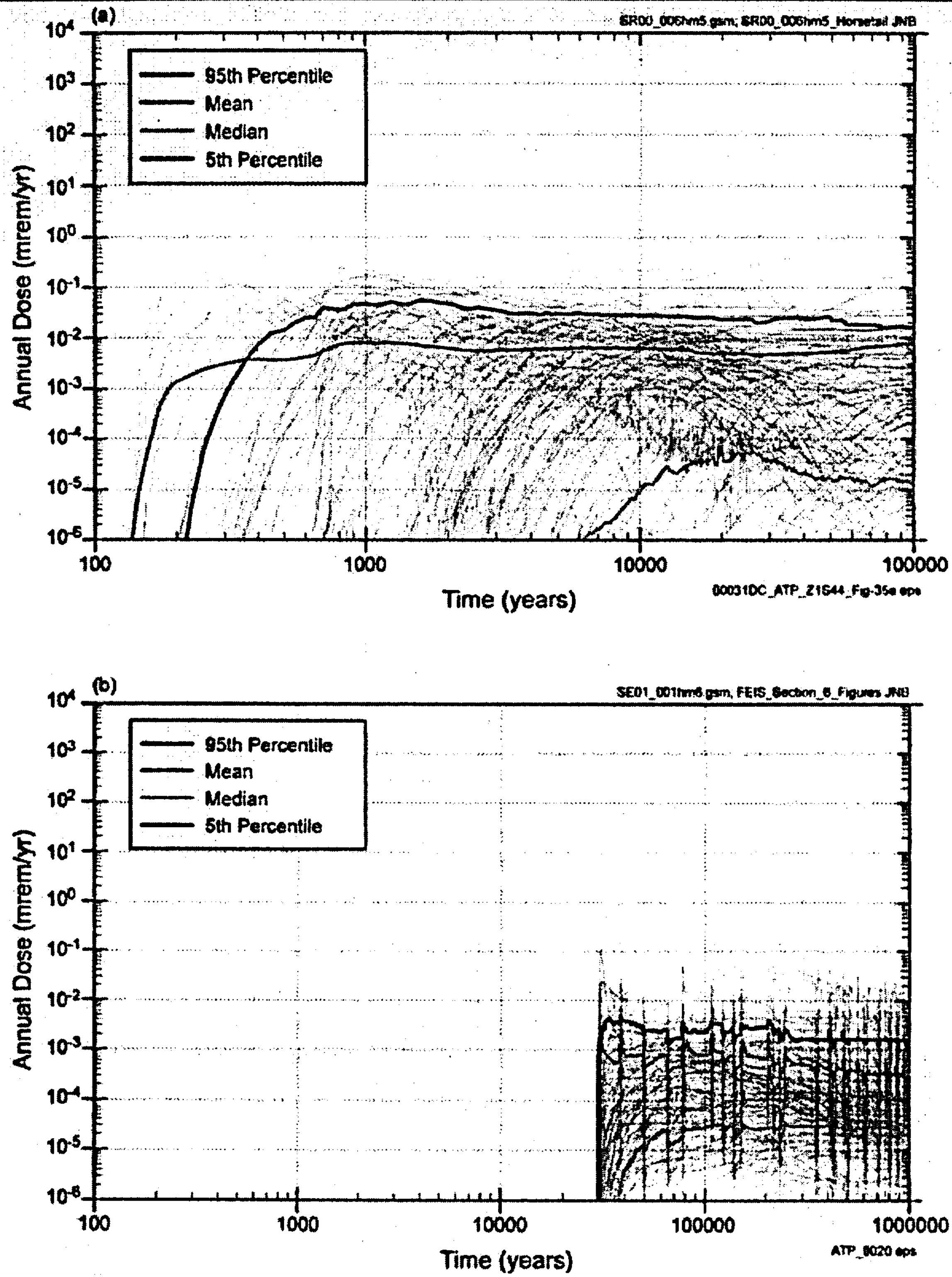


Figure 4-197. TSPA-SR Model and Revised Supplemental TSPA Results of Annual Dose to a Receptor for the Human Intrusion Scenario

(a) Human intrusion at 100 years projected by the TSPA-SR model. (b) Human intrusion at 30,000 years projected by the revised supplemental TSPA model. Source: CRWMS M&O 2000a, Figure 4.4-11; Williams 2001a, Figure 6-12.

2001a, Table 6-1). Considering the probability of the igneous initiating event occurring sometime in 30,000 years (4.8×10^{-4}), the earliest time after disposal that drillers would not recognize they had been penetrated a waste package, it was determined that the approximate maximum mean dose due to a human intrusion following an igneous intrusion would be the probability (4.8×10^{-4}) times the maximum mean dose (4.8×10^{-3} mrem/yr) or 2.3×10^{-6} mrem/yr. The maximum mean dose is much lower than the maximum mean dose due to the igneous intrusion alone of 4.3×10^{-4} mrem/yr (Williams 2001b, Section 6.2).

4.4.5 Sensitivity Analysis and Evaluation of Robustness of Repository Performance

The TSPA-SR model and supplemental TSPA model results for the nominal scenario are presented in Section 4.4.2, those for the disruptive events scenario are presented in Section 4.4.3, and those for the human intrusion scenario are presented in Section 4.4.4. These results illustrate a range of possible performance that is affected by the uncertainty in the individual component models and parameters used to describe the behavior of the system. An important goal of performance assessment is the clarification of the significance of these uncertainties on the overall performance of the potential repository. This section describes the sensitivity analyses conducted on these three scenarios. More details of these results and conclusions are included in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a, Sections 5.1 and 5.2) and Volume 2, Section 3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b).

Two different approaches have been used to evaluate the contribution of the uncertainty of different component models and parameters to the uncertainty in the dose rate projected by the TSPA-SR model. The first method, called either "stochastic sensitivity analysis" or "uncertainty importance analysis," uses the full suite of results available from the Monte Carlo-type analyses presented for each of the three scenarios. In this method, various statistical techniques are used to determine which models and parameters most significantly affect the mean and the variance (or spread) of the dose distribution. Other statistical techniques have been used to examine the models and parameters that most significantly affect the extremes (e.g., the top 10 percent) of the resulting dose distribution. These later analyses are informative because it is frequently the extremes (or tails) of the dose distributions that most affect the mean of the projected dose.

The second method, generally called "one-off sensitivity analysis," examines the effects of each component model or parameter on overall system performance. These analyses are conducted by fixing a particular model or parameter at either its expected value (generally the median or 50^{th} percentile value) or a specified extreme value (e.g., either the 5^{th} or 95^{th} percentile value). In these analyses, all other models and parameters other than the fixed value are still sampled from the uncertainty distribution used in the base case analysis. These sensitivity analyses are used to display the effect of the change on the mean predicted dose (or other measures of the system or subsystem performance) as well as on the variance (or spread) of the predicted dose. By fixing a particular parameter that significantly affects the spread of the overall system performance results, especially if the effect is on the upper 5 to 10 percent of the projected dose consequences, it is possible to directly examine the significance that particular parameter has on the resulting performance of the system.

Both of the above sensitivity analysis methods have been used in evaluating the sensitivity of the projected dose of the Yucca Mountain repository system. Detailed discussion of a wide range of TSPA-SR sensitivity analyses are presented in the *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a, Section 5). All analyses described in Volume 2, Section 3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) were conducted as one-off comparisons in which all models and input parameters are the same as those used in the TSPA-SR model except for the model or parameters being examined. Therefore, differences in performance measures between these results and those of the TSPA-SR base case provide insights into the importance of uncertainty in individual model components. Details may be found in Volume 2, Section 3.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). Sections 4.5.1, 4.5.2, 4.5.3, and 4.5.4 in this document report on TSPA-SR sensitivity analyses. Section 4.5.5 reports on supplemental sensitivity analyses reported in Volume 2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b).

4.4.5.1 TSPA-SR Model Nominal Scenario Sensitivity Analysis

The nominal performance scenario includes the models and parameters (and their corresponding uncertainty) for all relevant FEPs as they are expected to evolve over time for the Yucca Mountain repository system. The nominal performance results indicate a broad uncertainty in the predicted dose for several tens of thousands of years. As time proceeds, there is a broad range of possible dose rates. The causes for the broad range in projected dose rates are described in this section, and additional discussion of intermediate results is presented in the *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a, Sections 4.1, 5.1, 5.2, and 5.3).

Stochastic sensitivity analyses of the nominal performance scenario indicate that the dominant factors in the uncertainty of the projected dose rate (i.e., the variance of the predicted dose rates) vary with time from 40,000 to 100,000 years. Prior to about 40,000 years, there is an insufficient number of realizations with nonzero doses to perform representative stochastic sensitivity analyses. The temporal evolution of the importance is illustrated in Figure 4-198. The uncertainty importance factor is a measure of the importance of the uncertainty in the parameter as this uncertainty affects the uncertainty in the predicted dose; the higher the uncertainty importance factor, the more that parameter affects the distribution or spread in the projected doses.

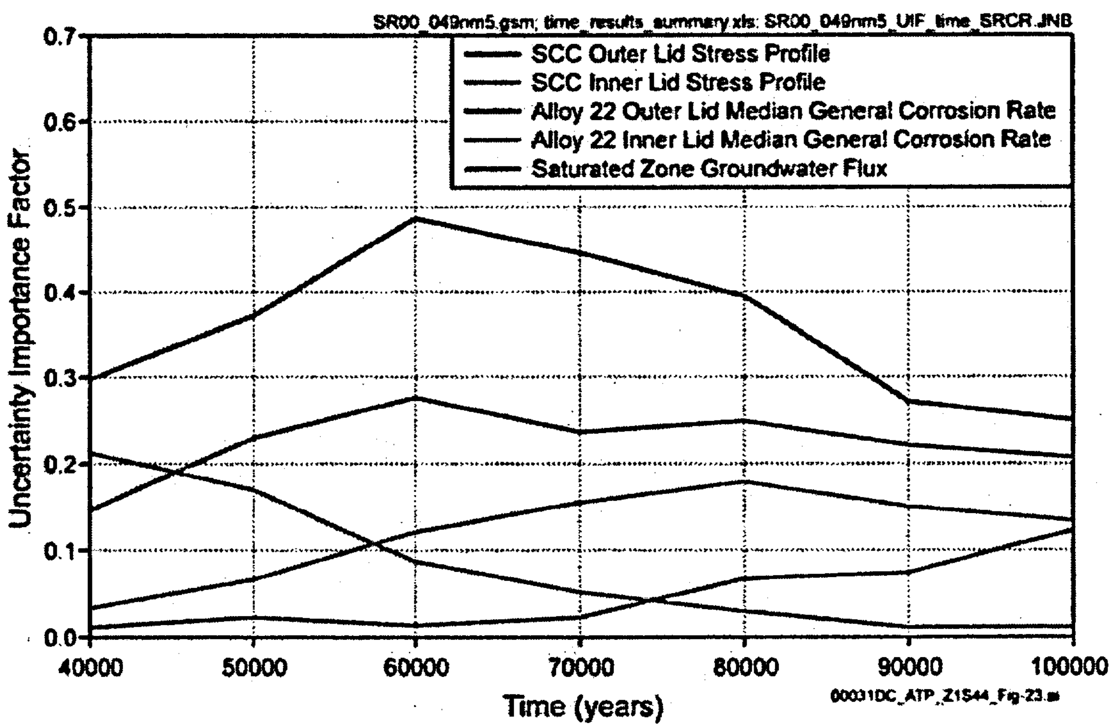


Figure 4-198. Summary of TSPA-SR Model Stochastic Sensitivity Analyses for Nominal Scenario—Parameters Affecting Dose Rate Uncertainty at Various Times
SCC = stress corrosion cracking. Source: CRWMS M&O 2000a, Figure 5.1-4.

The principal uncertainty importance factors determined from the regression analysis (ranked in their order of significance at 40,000 years) are:

- Uncertainty in the stress profile at the welds of the stress-mitigated outer Alloy 22 closure lids of the waste package that are subject to stress corrosion cracking
- Uncertainty in the stress profile at the welds of the stress-relieved inner Alloy 22 closure lids of the waste package that are subject to stress corrosion cracking
- Uncertainty in the median value of the general corrosion rate of Alloy 22 in the region of the outer Alloy 22 closure lid
- Uncertainty in the median value of the general corrosion rate of Alloy 22 in the region of the inner Alloy 22 closure lid
- Uncertainty in the groundwater flux in the saturated zone.

These five factors explain about two-thirds of the total variance of the dose results at 40,000 years.

In addition to examining the uncertainty importance factors as a function of the time at which a particular dose is projected to be received by the individual receptor (as is done in Figure 4-198), additional stochastic sensitivity analyses have been performed at four discrete dose levels (10, 1, 0.1, and 0.01 mrem/yr). The results of these uncertainty importance analyses are illustrated in Figure 4-199. These results confirm the results illustrated in Figure 4-198 in that it is the stress profile and corrosion rate of Alloy 22 that most significantly affects the degradation of the engineered barriers and determines the timing and magnitude of the distribution of doses projected to be received by the receptor.

The previous list illustrates the significant effect that uncertainty in the waste package corrosion rates and stress states at the closure welds could have on the time it takes for waste packages to breach and on the total amount of degradation of

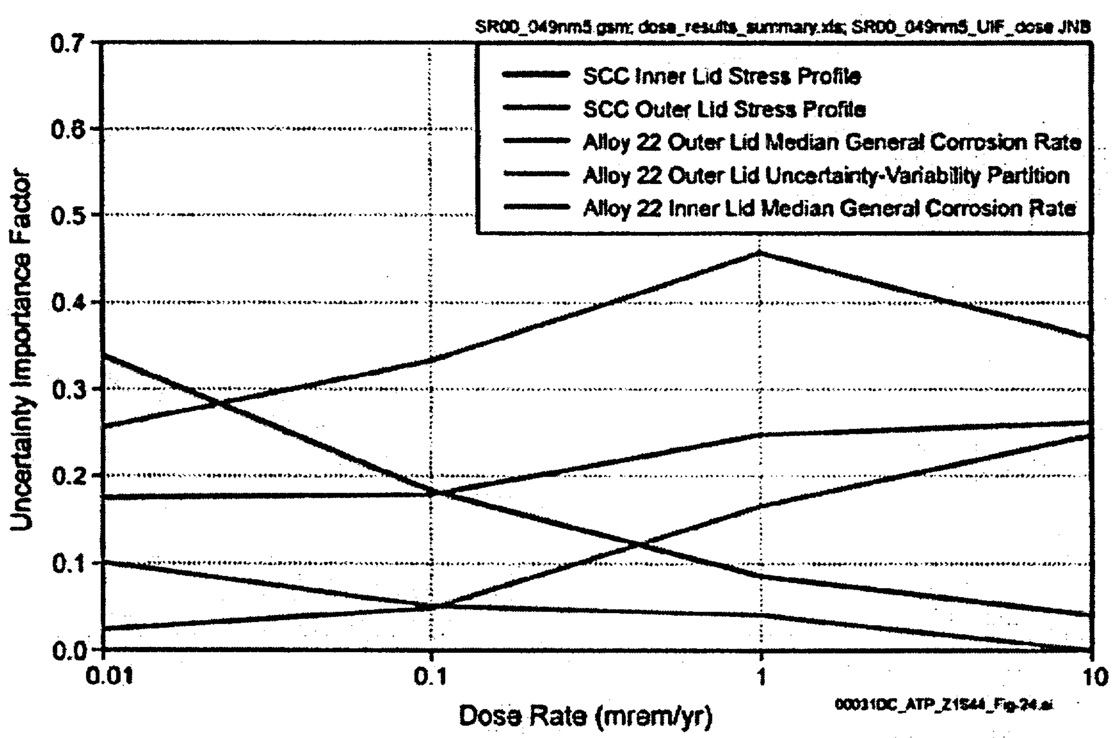


Figure 4-199. Summary of Stochastic TSPA-SR Model Sensitivity Analyses for Nominal Scenario—Parameters Affecting Uncertainty in Time of Dose Rate for Various Dose Rates
SCC = stress corrosion cracking. Source: CRWMS M&O 2000a, Figure 5.1-10.

the waste packages. Until the principal containment barrier of the waste packages is breached, there can be no release from the waste packages and therefore no doses. Therefore, the key parameters that affect the waste package degradation are also the key parameters that affect the total system dose.

The two natural system parameters determined to be most significant are the infiltration rate and the saturated zone advective flux (the former appears more prominent in the 1-million-year analyses described in Section 5.1 of the *Total System Performance Assessment for the Site Recommendation* [CRWMS M&O 2000a]). The infiltration scenario affects the advective transport time through the unsaturated zone (due to the effect on the percolation flux and the height of the water table) and the fraction of waste packages likely to encounter seeping conditions in the repository. The saturated zone groundwater advective flux affects the transport time of key radionuclides through the saturated zone (in particular, moderately sorbing radionuclides like neptunium-237) and the fraction of the total travel path that is in the alluvial aquifers.

In addition to the regression analysis that determines the most significant parameters affecting the variance of the projected dose rate versus time, classification and regression tree analyses have been performed to identify the key variables controlling the extreme realizations (e.g., the top 10 percent). These analyses confirm that the causes of early waste package failures and, therefore, the causes of doses are determined by a few waste package-specific parameters that affect the degradation rates (due both to corrosion and stress corrosion cracking) at the middle lid and outer lid closure welds (CRWMS M&O 2000a, Section 5.1).

4.4.5.1.1 TSPA-SR Model Sensitivity Studies for the Higher-Temperature Operating Mode

Using the factors identified as the most significant contributors to performance based on the stochastic sensitivity analyses described previously, one-off sensitivity analyses have been performed to illustrate the significance of these factors on the mean dose rate. These analyses are performed by fixing one or more models or parameters at their extreme values (5th and 95th percentiles) and then rerunning the calculations with all other models and parameters sampled from their "base case" distributions. In the following figures, comparisons of the one-off sensitivity analyses are made using the mean of the overall performance distribution. Additional analyses described in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a, Section 5.2) evaluate the variance reduction of the projected dose response from these one-off sensitivity analyses. As in the previous sections, these analyses have been conducted out to 100,000 years to gain insights into the system behavior.

Figure 4-200 illustrates the mean in the predicted dose rate when the stress state at both the inner and outer Alloy 22 closure welds is fixed at the 95th and 5th percentile values from the total uncertainty distribution described in Section 4.2.4. The results are compared to the base case results presented in Section 4.4.2. As expected, the timing of a particular mean dose is significantly affected by changes in the stress state. When the stress profile is fixed at the 95th percentile of the considered distribution, the fraction of the lid thickness that must be corroded before stress corrosion processes are initiated is significantly reduced, which significantly reduces the time required for a breach of the waste package at the lid closure welds. Conversely, when the stress profile is fixed at the 5th percentile, the fraction of the lid thickness that must be corroded before stress corrosion processes are initiated is increased significantly, which significantly delays the breach time of the waste package at the lid closure weld.

Figure 4-201 illustrates the mean in the predicted dose rate when the median value of the general corrosion rate for Alloy 22 for the region of both the outer and middle closure lids is fixed at the 95th and 5th percentile values. Again, as the corrosion rate is fixed at a high (e.g., the 95th percentile) value within the range of possible corrosion rates, the time for the mean waste package breach is reduced. Conversely, when the corrosion rate is fixed at a low (e.g., the 5th percentile) value within

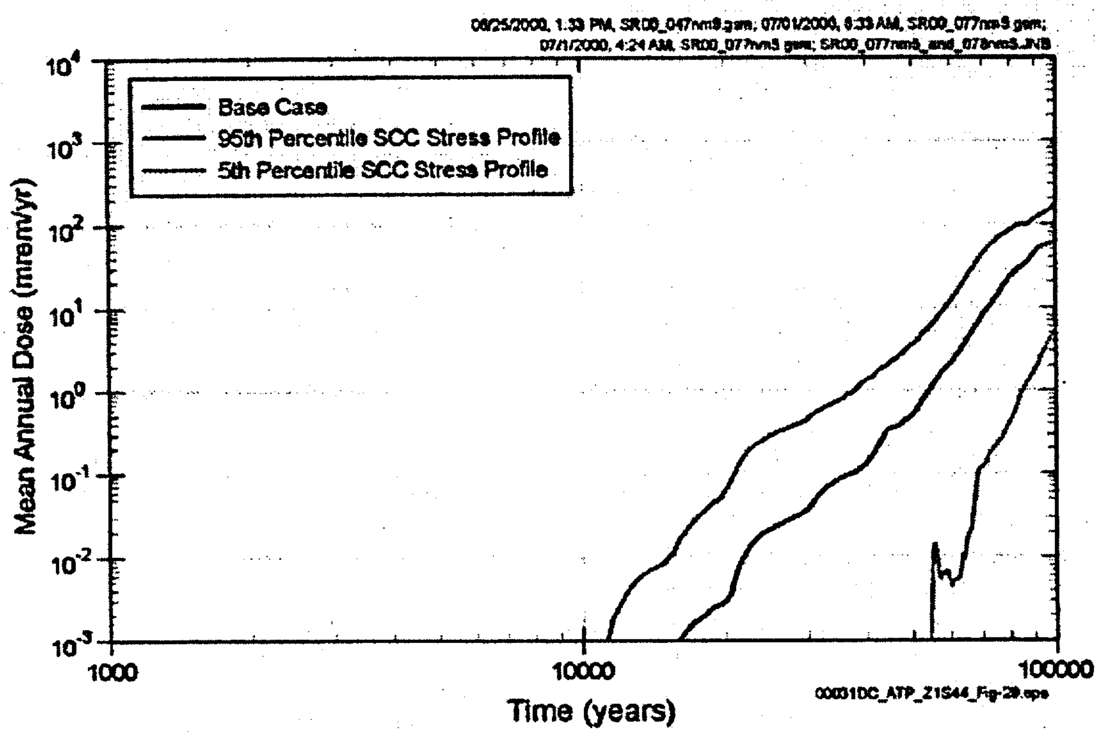


Figure 4-200. Sensitivity of the Mean Annual Dose Calculated by the TSPA-SR Model to Uncertainty in the Stress State at Closure Welds

SCC = stress corrosion cracking. Source: CRWMS M&O 2000a, Figure 5.2-3.

the range of possible corrosion rates, the time for the mean waste package breach is increased. This parameter significantly affects the rate of degradation of the waste packages and the variability in the waste package failures.

In separate barrier importance analyses discussed in Section 4.5.4, all of the important waste package degradation parameters described previously were fixed at their 5th and 95th percentiles. The barrier importance analysis results presented in Section 4.5.4 confirm the individual sensitivity analysis presented above.

Figure 4-202 illustrates the significance of fixing the infiltration rate at the low or high values of the distribution presented in Section 4.2.1. Again, the results are compared to the "base case" results described in Section 4.4.2. The significance of the parameter illustrated in this figure is less than that observed in the waste package degradation sensitivity analysis. The slight difference between the mean dose when the infiltration is fixed at its maximum value and the mean dose when the infiltration is sampled illustrate that the mean dose response of the base case is already significantly affected by the maximum values of the infiltration rates. However, the 5th percentile infiltration rate has a significant effect on the predicted dose response. This result is primarily due to the effect of reduced seepage flux as the infiltration rate is reduced (CRWMS M&O 2000a, Section 5.2).

Figure 4-203 illustrates the significance of seepage to system performance. In this particular sensitivity analysis, the seepage flow focusing factor is fixed at the 95th or 5th percentile. At high ends of the flow focusing factor, more water is allowed to focus on the drifts and may potentially seep if the capillarity of the fractures is insufficient to keep the water in the rock. As expected, seepage has a minimal significance until such times that the dose is dominated by solubility-limited releases (i.e., greater than about 50,000 years) because the more soluble radionuclides, such as technetium-99, can more readily diffuse through the engineered

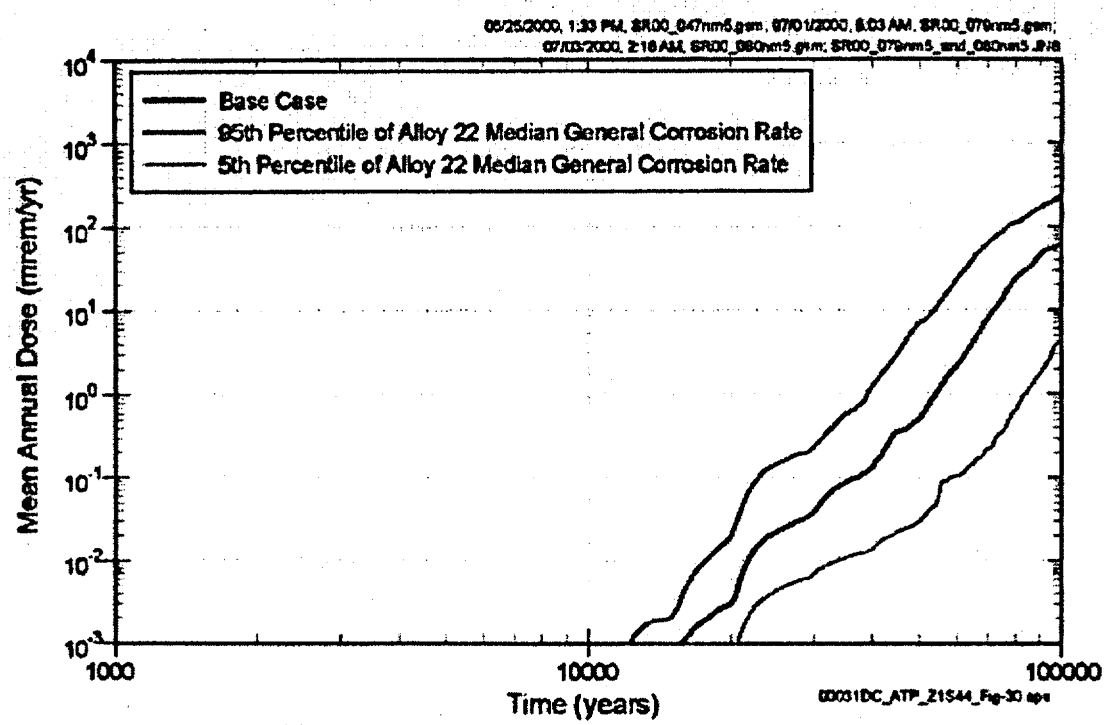


Figure 4-201. Sensitivity of the Mean Annual Dose Calculated by the TSPA-SR Model to Uncertainty in the Median General Corrosion Rate of Alloy 22
Source: CRWMS M&O 2000a, Figure 5.2-9.

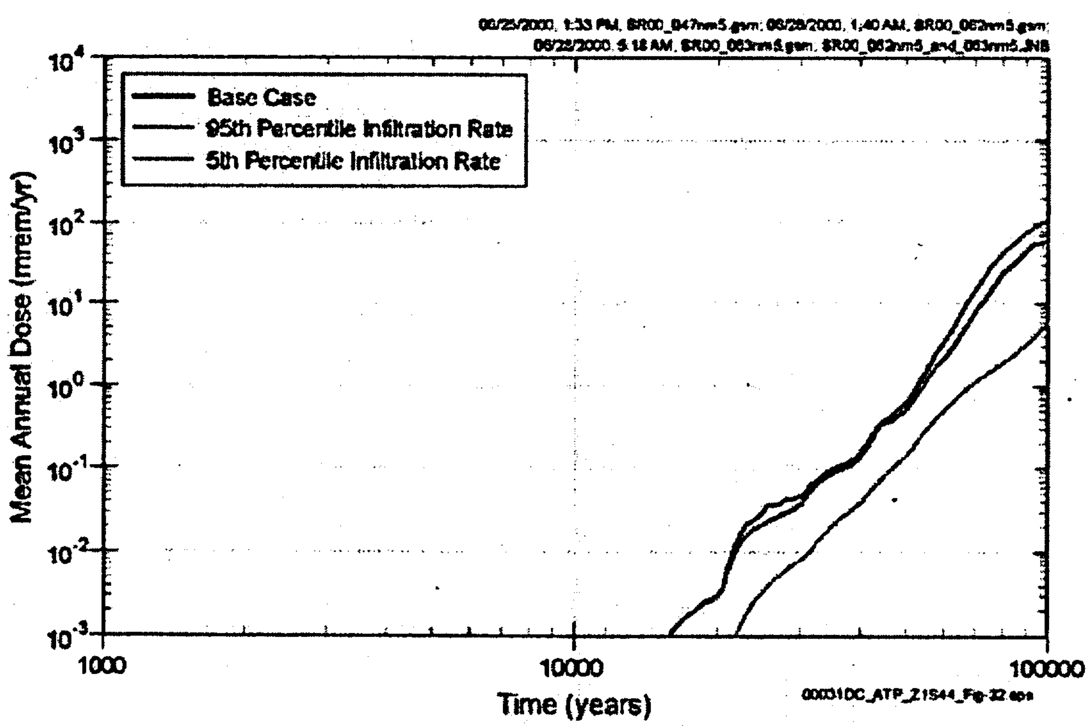


Figure 4-202. Sensitivity of the Mean Annual Dose Calculated by the TSPA-SR Model to Uncertainty in Infiltration Rate
Source: Modified from CRWMS M&O 2000a, Figure 5.2-1.

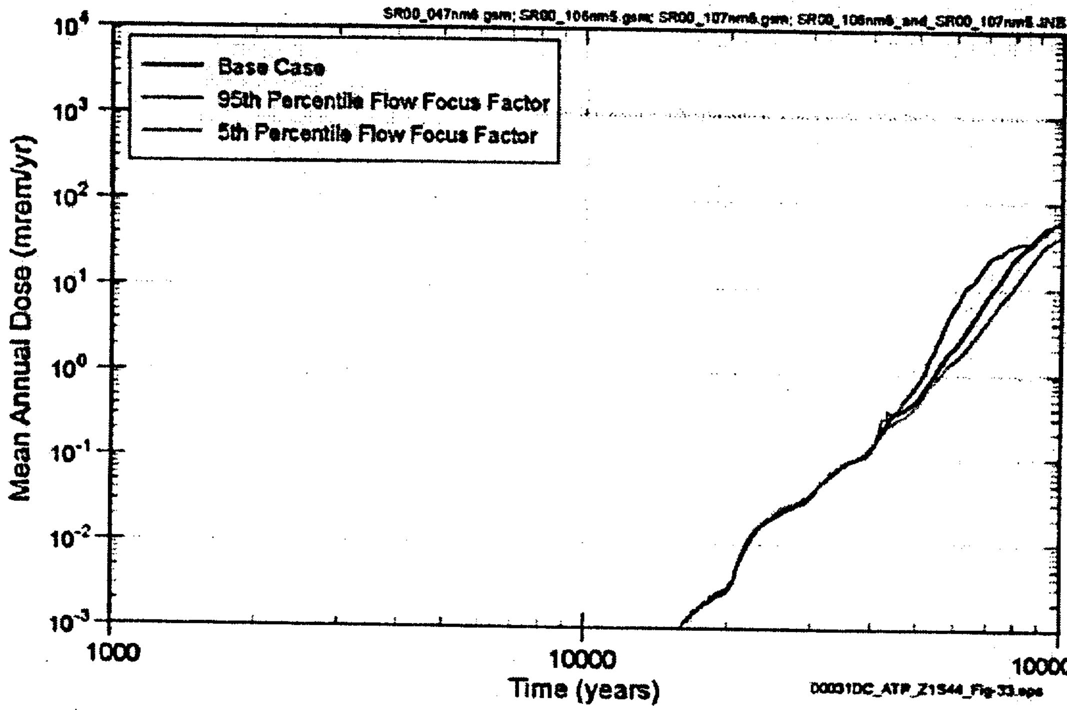


Figure 4-203. Sensitivity of the Mean Annual Dose Calculated by the TSPA-SR Model to Uncertainty in the Seepage Rate
Source: CRWMS M&O 2000a, Figure 5.2-2a.

barriers once they have been breached. In addition, seepage will only be significant when an advective pathway through the drip shields and waste packages is created, which requires a significant fraction of both the drip shields and waste packages to have been degraded.

The above discussion illustrates the significance of some of the most important parameters affecting the projected dose rate for the nominal performance scenario. Other parameters and models that are uncertain and for which sensitivity analyses have been performed are presented in Section 5.2 of the *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a). The barrier importance analyses presented in Sections 4.5.3 and 4.5.4, in which the variables affecting several process model factors are varied simultaneously, identify additional parameters of potential significance to overall system performance.

4.4.5.1.2 TSPA-SR Model Sensitivity Studies for Alternative Design Features and Lower-Temperature Operating Modes

This section describes analyses that have been performed to address alternative operating modes that could result in lower temperatures. Specifically, these modes would not allow temperatures above the boiling point of water to occur in the host rock.

The sensitivity studies for the lower-temperature operating mode described in this section were undertaken using the process and TSPA models described in this section. Section 2.1 contains descriptions of other lower-temperature operating modes. Design studies were used to develop an understanding of the environmental conditions associated with these operating modes. Enhancements were made to the performance assessment models to conduct the performance assessment and sensitivity evaluations of the additional lower-temperature operating modes. These enhancements incorporate the results of efforts to quantify uncertainties and extend the applicable range of the process models (BSC 2001a; BSC 2001b). Of particular interest are enhancements that address the performance-related responses of the design and operating mode, considering temperature-sensitive parameters and coupled thermal-mechanical-chemical-hydrologic processes. This approach is intended to ensure that the performance evaluations appropriately consider the potentially detrimental and potentially beneficial aspects of the repository's performance over a range of operating modes encompassing above- and below-boiling conditions.

Sensitivity analyses were also performed to assess the performance-related impacts of the potential addition of backfill. In theory, backfill could have several desirable attributes, such as limiting the potential effect of rockfall and providing a well-controlled thermal-hydrologic-chemical environment in which the rest of the engineered barriers reside. Backfill also has a potential benefit of reducing the humidity on the waste package surface for several thousand years after closure, delaying the onset of the aqueous corrosion processes that can take place in humid environments. However, backfill could also have negative effects, such as increasing the cladding peak temperature and accelerating the amount and rate of cladding degradation. The effect of adding backfill to the repository design was evaluated, and the results are shown in Figure 4-204. This analysis indicates little net effect of the potential positive and negative performance aspects of backfill.

For the higher-temperature operating mode, the design described in Section 2 includes a 50-year ventilated preclosure period, a linear thermal loading of 1.45 kW/m, and a constant drift to drift spacing of 81 m (266 ft). For the higher-tempera-

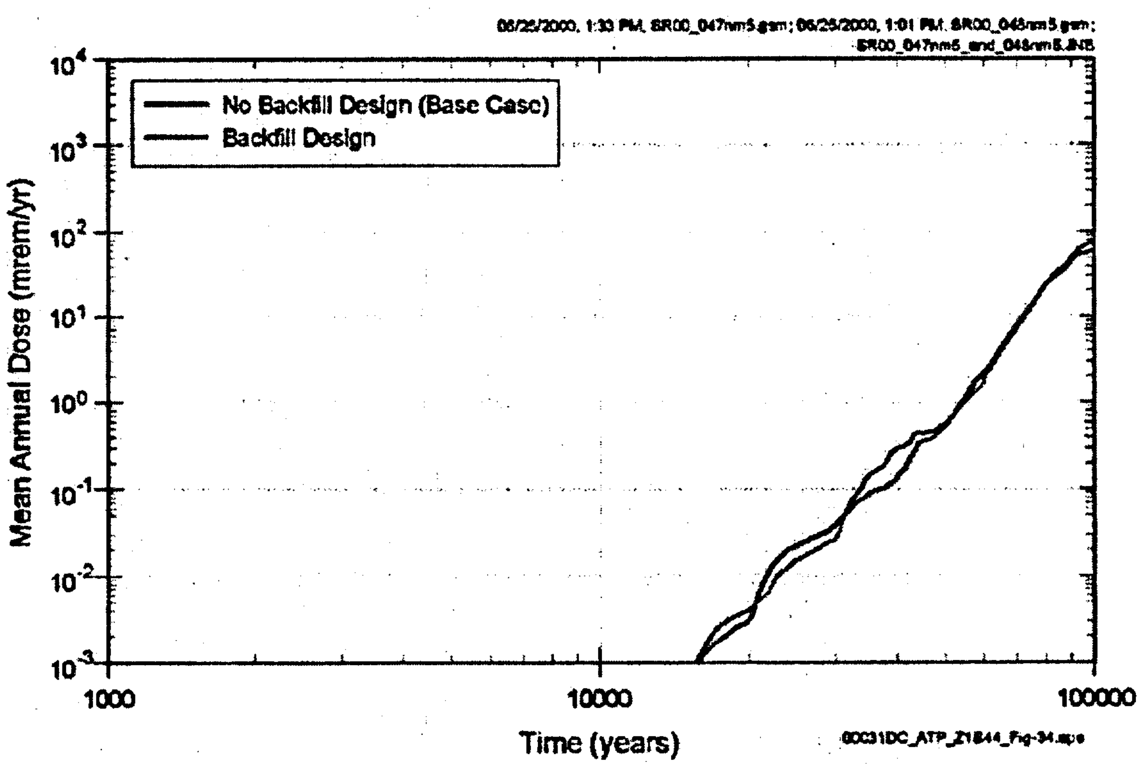


Figure 4-204. Sensitivity of the Mean Annual Dose Calculated by the TSPA-SR Model to Adding Backfill to the Repository Design
Source: CRWMS M&O 2000a, Figure 4.6-5.

ture operating mode, this design would allow boiling to occur several meters into the host rock surrounding the emplacement drifts for tens to hundreds of years after closure, depending on the relative location of the drift (i.e., edge versus center of the potential repository). In this section, the performance of a lower thermal load operating mode based on a reduced thermal loading of 0.90 kW/m and a 100-year ventilated preclosure period is compared to the higher-temperature operating mode. This alternative operating mode would not produce boiling in the host rock. The analysis is summarized in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a, Section 4.6.2).

To facilitate the analysis, the following key assumptions were made:

1. It is assumed that a specific line-loaded, two-dimensional, drift-scale, thermal-hydrologic submodel is representative of an average location in the potential repository footprint. This assumption is based on the selected submodel's physical location relative to the geometric center of the potential repository footprint.

2. Spatial variability in potential repository thermal-hydrologic variables does not have a significant effect on engineered barrier system performance. Therefore, thermal-hydrologic output variables from the line-loaded thermal-hydrologic submodel can be applied throughout the potential repository.

3. The reduction of power output from 1.45 kW/m to 0.90 kW/m is accomplished by increasing the waste-package-to-waste-package spacing. This increase in waste package spacing is accounted for in the two-dimensional model by applying a scaling factor to the original design model's thermal power curve.

4. The reduction in linear power output is accomplished by increasing waste-package-to-waste-package spacing in the emplacement drifts. As a result, the potential repository footprint should also increase accordingly. In the analyses presented here, the effects of an increased footprint on potential repository performance are neglected.

5. The effects and uncertainties associated with coupled thermal-hydrologic-chemical-mechanical processes, such as dissolution and precipitation of minerals and thermally induced fracturing in the host rock, may decrease in magnitude as the potential repository thermal loading decreases. These potential decreases in effects and uncertainties are neglected in the present comparison.

In the lower-temperature operating mode, waste package surface temperatures reach a much lower peak temperature, as expected. In the higher-temperature operating mode, elevated waste package surface temperatures are accompanied by a corresponding decrease in relative humidity around the waste package. This decrease in relative humidity does not occur in the low thermal load case since surface temperatures do not rise significantly.

Curves showing the rate of initial waste package failure (CRWMS M&O 2000a, Section 4.6.2) indicate that there is not a significant difference in the two cases as they are currently modeled. This result illustrates the insensitivity of the waste package corrosion and degradation model to thermal-hydrologic conditions around the waste package. As shown in Figure 4-205, the performance results for the higher-temperature operating modes compared to the lower-temperature modes in both TSPA-SR analyses and supplemental TSPA analyses (CRWMS M&O 2000a; BSC 2001a; BSC 2001b) are similar because neither would expose the engineered barriers (the drip shield and the waste packages) to temperatures or geochemical conditions that would be expected to significantly increase general corrosion or stress corrosion cracking rates, or waste form dissolution rates. The thermal design goals established for the higher-temperature operating mode (see Section 2.3.4.3) were meant to ensure that temperatures or geochemical environments that would promote

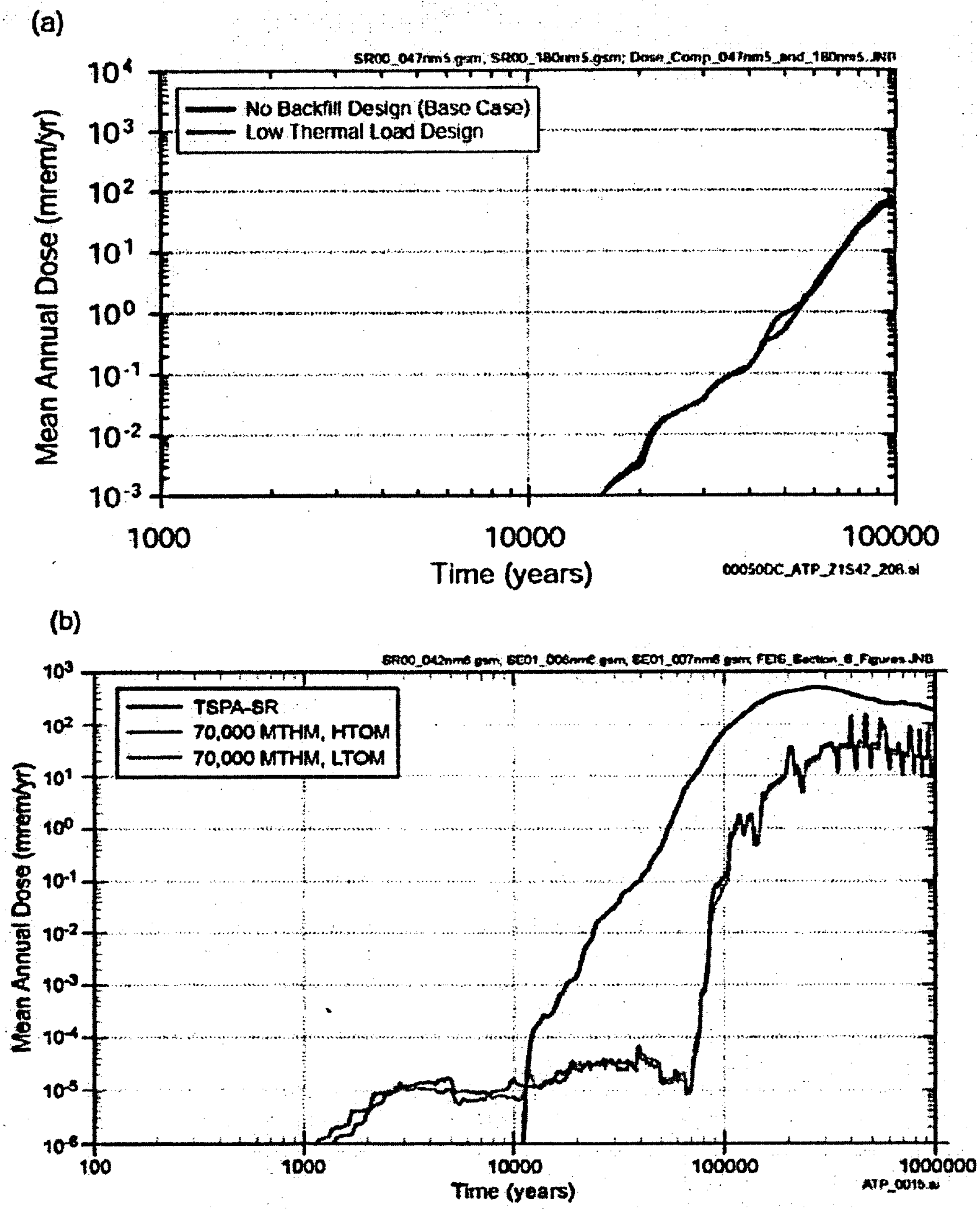


Figure 4-205. Comparison of Doses Projected by the TSPA-SR Model and Revised Supplemental TSPA Model for the Higher- and Lower-Temperature Operating Modes for the Nominal Scenario

(a) Comparison plot of the mean annual dose versus time for the TSPA-SR model for the higher-temperature operating mode and the lower-temperature operating mode for the nominal scenario. (b) Comparison plot of the mean annual dose versus time for the revised supplemental TSPA model for the higher-temperature operating mode (HTOM) and the lower-temperature operating mode (LTOM) for the nominal scenario. Source: CRWMS M&O 2000a, Figure 4.6-10; Williams 2001a, Figure 6-3.

corrosion and stress cracking would not be encountered in the repository. The processes and conditions expected in the natural environment and in the repository emplacement drifts are described in Sections 4.2.2, 4.2.3, and 4.2.5.

In conclusion, although the low thermal load case reduces waste package surface temperatures and increases the relative humidity around waste packages, these effects do not significantly impact waste package performance. In addition, since waste package failure does not occur until after 10,000 years, the thermal-hydrologic conditions for both cases are similar during the period when radionuclides are mobilized. As a result, doses for both cases show very little difference.

4.4.5.1.3 Sensitivity Analyses for Other Lower-Temperature Operating Modes

The effect of heat on the performance of the repository and the associated uncertainties are discussed in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b). These studies considered ranges of drift wall temperatures as high as 200°C (392°F) to below the boiling point of water (96°C [205°F]) at the elevation of the emplacement horizon). An objective of the higher-temperature operating mode is to maintain temperatures in a portion of the rock between the emplacement drifts below the boiling point of water. Supplemental TSPA studies include sensitivity analyses that evaluated repository performance limiting all drift wall temperatures below the boiling point of water. Lower-temperature operating modes to reduce uncertainty about corrosion rates associated with waste package performance have also been evaluated in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b). These evaluations considered temperatures as low as 85°C (185°F). The result of these studies are reported in Volume 1 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a). A summary of these studies are found in Section 4.4.5.5 of this report. In general, projected doses for both the higher- and lower-temperature operating modes are similar. This is true for both TSPA-SR model and revised supplemental TSPA model projections (see Figure 4-205).

The performance assessment aspects of the flexible design and ranges of operating modes for a potential repository at Yucca Mountain are related to understanding the impact that a design component or operational performance objective could have on the performance of the site across a range of environmental conditions. The use of TSPA as a tool in the evaluation of the performance characteristics of the proposed repository design over a range of operating conditions is an important step in the evaluation of the design. The evolution of the design and operating mode information is a process that includes (1) refining specific design requirements and performance goals to recognize performance related benefits that could be realized through design and (2) enhancing components of the design to best achieve the performance-related benefits.

Analyses of a range of lower-temperature operating modes was used to support development of further understanding of potential performance benefits that could be realized through specific repository operating modes (BSC 2001a; BSC 2001b). If a design attribute was shown to have a significant impact on the performance of the repository, then the attribute underwent further evaluation to define the positive contribution or limit the negative contribution of this attribute in a manner that could enhance the performance of the repository. If the performance evaluations indicate benefits to be gained by refinement of the basic design or operating mode concept, the evolution of the design will take advantage of those insights.

4.4.5.2 TSPA-SR Model Sensitivity Analyses for Disruptive Scenarios

As Section 4.3.2 describes, igneous disruption is the only disruptive scenario that has been identified as requiring explicit analysis in the TSPA. Section 4.4.3 describes the TSPA results for the igneous disruption scenario. This section presents the results of two sensitivity analyses examining alternatives to the modeling assumptions used in the TSPA-SR model (CRWMS M&O 2000a, Section 5.2.9). These sensitivity analyses have been performed using 1,000 realizations and a 20,000-year period of simulation. These analyses are presented here to provide insight into the robustness of the TSPA-SR model results for the disruptive scenario performance analyses. The alternative modeling assumptions represented by these analyses are not considered to be realistic, and the mean probability-weighted 50,000-year dose rate described in Section 4.4.3 should be interpreted as the best estimate of future performance for the igneous disruption scenario class.

Figure 4-206 shows a comparison of the probability-weighted 20,000-year mean annual igneous dose rate, as described in Section 4.4.3, with the same dose rate calculated using a fixed annual probability of both eruption and igneous intrusion equal to 10^{-7}, rather than a value for igneous intrusion sampled from a distribution with a mean of 1.6×10^{-8}. For additional conservatism, the conditional probability that an eruptive conduit intersects waste if intrusion occurs is set to 1 in this analysis. This higher probability is the value used by the NRC in analyses reported in their igneous activity Issue Resolution Status Report (Reamer 1999, p. 11). Because the event probability is used directly in the weighting of probabilistic doses, changes in event probability should result in a linear scaling of the mean annual dose. Figure 4-206 confirms this observation. The mean dose calculated using the fixed higher probability (shown in red) is about 17 times higher during the first 2,000 years than the mean dose calculated using the full distribution of probabilities. At later times, the scaling between the curves varies slightly with time, reflecting both the sampling of the time of intrusion and the influence of individual realizations with varying probabilities on the location of the mean at different times.

Figure 4-207 shows the second of the two one-off sensitivity analyses reported here, which is a comparison of the probability-weighted mean annual dose rate, as described in Section 4.4.3, with the same dose rate calculated using the 95th or 5th percentile values for the number of waste pack-

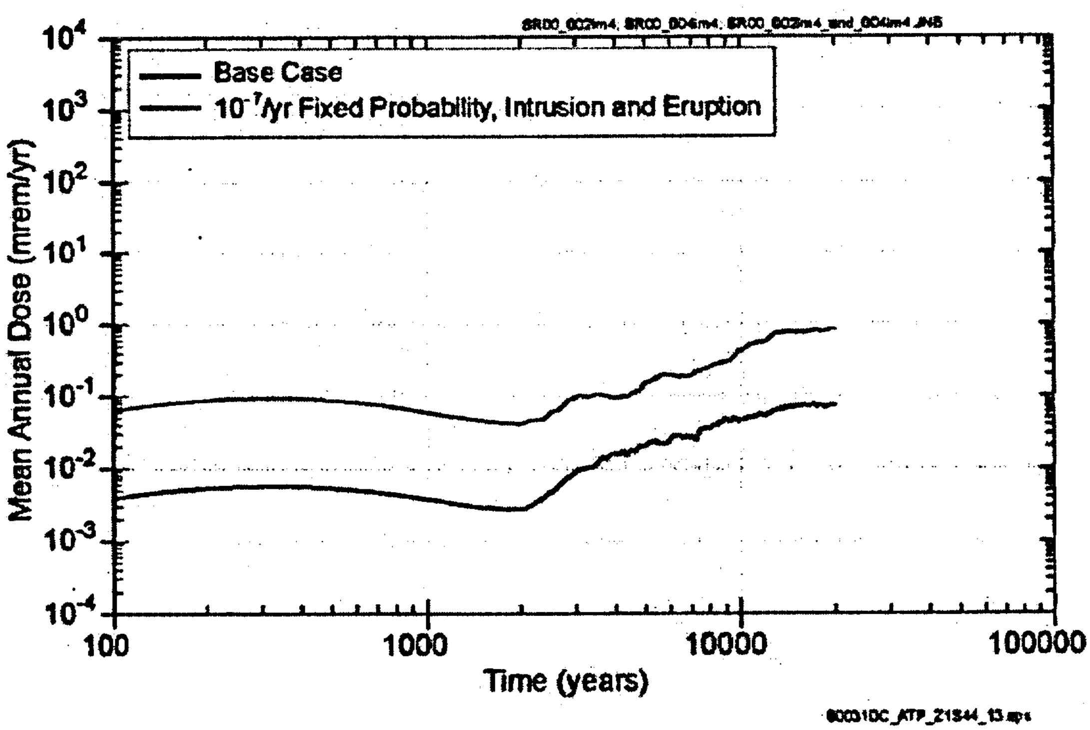


Figure 4-206. Sensitivity of the Mean Annual Dose Calculated by the TSPA-SR Model for the Volcanic Scenario to Uncertainty in Probability of Volcanic Intrusion and Eruption
Source: CRWMS M&O 2000a, Figure 5.2-17.

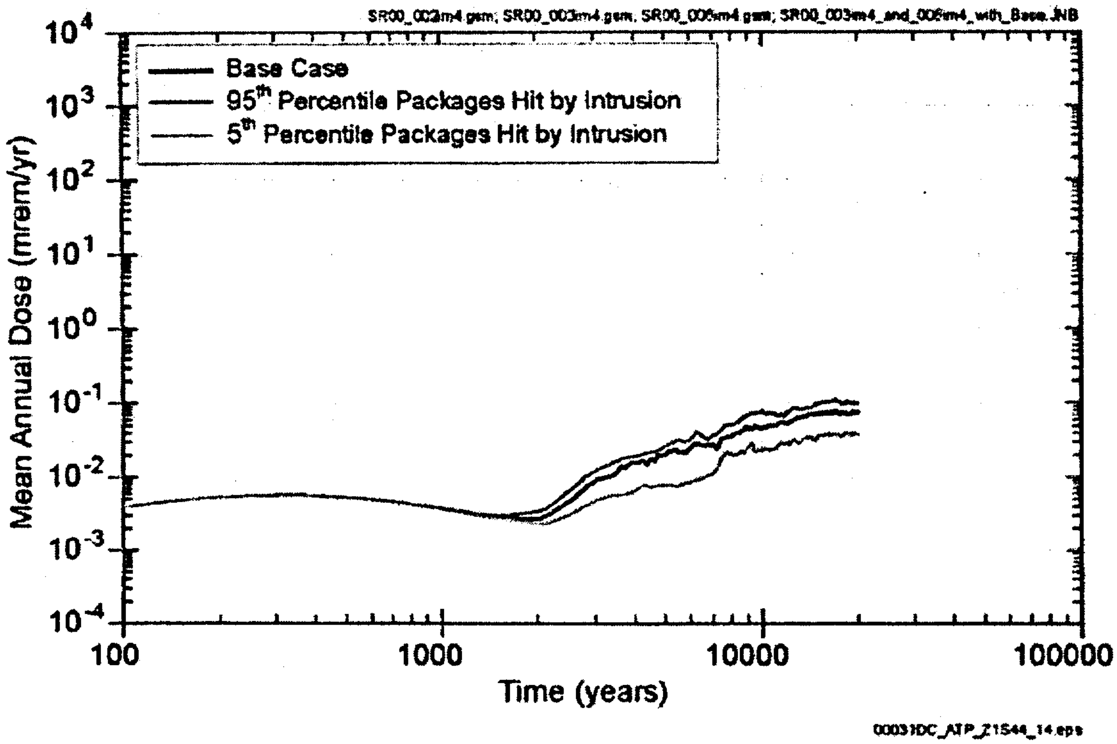


Figure 4-207. Sensitivity of the Mean Annual Dose for the Volcanic Scenario Calculated by the TSPA-SR Model to Uncertainty in the Number and Extent of Waste Packages Damaged by the Volcanic Intrusion

Source: CRWMS M&O 2000a, Figure 5.2-23.

ages damaged by igneous intrusion. This comparison provides insight into the sensitivity of overall performance to uncertainty about the repository's response to igneous intrusion. As Section 4.4.3 describes, packages may be sufficiently damaged by their close proximity to the igneous intrusion that they provide no further protection, or they may be partially damaged due to elevated temperature and pressure in the emplacement drift. Both types of damage are treated as uncertain parameters in the TSPA-SR model. In this sensitivity analysis of 20,000-year performance, these sampled values were replaced with fixed numbers corresponding to the 95th percentiles of the TSPA-SR distributions for the parameters. Specifically, these 95th percentile values are that 219 waste packages are fully damaged by igneous intrusion and that an additional 6,297 packages (more than half those in the repository) are partially damaged. All other parameters have the same values, sampled or fixed, that were used in the TSPA-SR model. Results of this comparison show that performance is only moderately sensitive to the total number of packages that are damaged by intrusion, with peak dose increasing by less than a factor of 2.

Additional sensitivity analyses were performed by the supplemental TSPA model. These results are presented in Volume 2, Section 3.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) and Section 4.4.5.3 of this report.

4.4.5.3 TSPA-SR Model Sensitivity Analyses of the Human Intrusion Scenario

Because the 100-year human intrusion dose is largely determined by the stylized nature of the analysis, results are insensitive to uncertainty regarding physical properties and processes related to the intrusion event.

Stochastic sensitivity analyses, as described in Sections 4.4.5 and 4.5, indicate that the mean annual dose rate following a 100-year human intrusion is sensitive to uncertainty in the parameters that affect transport in the saturated zone. Specifically, the parameters controlling advective flux in the saturated zone and the solubility of neptunium-237 show the greatest statistical importance. A one-off sensitivity analysis showed that degradation of the saturated zone flow and transport processes in the TSPA-SR model could result in a moderate increase in the 100-year human intrusion dose rate (CRWMS M&O 2000a, Section 5.3.7). However, the peak mean annual dose rate for the 95th percentile infiltration rate did not exceed 0.02 mrem/yr over the entire 100,000 years and was about 0.01 mrem/yr at 10,000 years after repository closure for the 100-year human intrusion event.

Sensitivity of the human intrusion scenario at 30,000 years to an unlikely igneous event is reported in *Total System Performance Assessment Sensitivity Analyses for Final Nuclear Regulatory Commission Regulations* (Williams 2001b). These results are discussed in Section 4.4.4.2.2 of this report.

4.4.5.4 Summary of TSPA-SR Sensitivity Analyses

The goal of the TSPA-SR sensitivity analyses, and the other analyses documented in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a, Section 5), is to provide additional insights into the significant contributors to overall system performance. These significant contributors are defined with respect to how much they modify the mean of the predicted dose response as well as their contribution to the overall uncertainty in the predicted dose response. The focus has been on the significance with respect to the individual dose performance (which includes the nominal performance scenario, the disruptive events performance scenario, and the stylized human intrusion scenario). These sensitivity analyses have been conducted out to 100,000 years to gain insights into the repository system behavior. Additional analyses are presented in Section 4.5 where several factors have been varied simultaneously to evaluate the robustness of the repository performance.

4.4.5.5 Supplemental TSPA Sensitivity Analyses For Nominal Performance

This section describes results presented in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b) of one-off sensitivity analyses conducted by modifying the models and input parameters used in the TSPA-SR model. Except for the model or parameter being examined, the one-off sensitivity analyses were conducted using the same models and input parameters as those used in the TSPA-SR model base case, and therefore differences in performance measures between these results and those of the TSPA-SR model provide insights into the importance of uncertainty in individual model components. Analyses are presented for each of the major modeling subsystems, and the results are displayed as system-level annual dose histories for nominal performance and as intermediate performance measures, where appropriate. All analyses described in this section use 100 realizations of the TSPA base-case model (as modified for the one-off sensitivity analyses), and the results are compared to those of the TSPA-SR model base case.

Extended Climate Model—A supplemental sensitivity analysis was conducted in which the extended climate model (CRWMS M&O 2000a, Section 3.2.5) was used, but the rest of the model was the same as the TSPA-SR model base case. A comparison of the calculated mean annual dose to the receptor for this analysis with the TSPA-SR model base case is discussed in Volume 2, Section 3.2.1 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). Each TSPA simulation is a combination of the low-, medium-, and high-infiltration cases (BSC 2001b, Section 3.2.1.1). The probabilities for the infiltration cases are the same for both simulations. The calculated dose peaks during the glacial climates because of the increased seepage during those periods. Relatively little change in dose is projected during the first glacial climate (at 38,000 years) because the drip shields and waste packages will still be largely intact at that time (CRWMS M&O 2000a, Section 4.1), and they divert most of the seepage water around the waste packages during that period. The seepage flow rate increases during the glacial climates and decreases during the interglacial climates, as compared to the TSPA-SR base-case model. The interglacial periods occur at 65,000 years, 137,000 years, etc. (CRWMS M&O 2000a, Table 3.2-4). Despite the large increase in infiltration during the glacial periods, the number of waste packages that are subjected to seepage is less than 20 percent higher than the TSPA-SR base-case model.

Updated Seepage Model and Abstraction—Releases from the engineered barrier system are computed for 30 environmental groups that are based on infiltration, waste type, and seepage condition. Each environmental group is associated with one of five infiltration categories. The range for the categories are 0 to 3 mm/yr, 3 to 10 mm/yr, 10 to 20 mm/yr, 20 to 60 mm/yr, and greater than 60 mm/yr. A supplemental sensitivity analysis was conducted in which the revised seepage abstraction was used, but the rest of the model was the same as the TSPA-SR model base case. A comparison between the mean annual dose for the case with the revised seepage abstraction and the TSPA-SR base case is discussed in Volume 2, Section 3.2.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The results are essentially the same except near 60,000 years, where the sensitivity case is higher than the TSPA-SR model base case. That difference was traced to one realization in which the seepage flow rate was nearly ten times higher than in the TSPA-SR model base case for some environmental groups, causing the pulse of advective releases from the initial cladding failures in the commercial spent nuclear fuel to occur earlier than in the TSPA-SR model base case (BSC 2001b, Section 3.2.2).

The distributions of seepage flow rate are different in the updated seepage abstraction compared to those in the TSPA-SR model base case, but not greatly different. For example, Volume 2, Section 3.2.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) discusses a comparison of the mean seepage flow rate for two of the environmental groups. The increase in seepage after 2,000 years is caused by the change from monsoon climate to glacial-transition climate.

The seepage fractions in the updated seepage abstraction are lower because of the inclusion of the lower-lithophysal seepage data; on average, less than half as many waste packages are exposed to seepage in the updated seepage case.

Effects of Flow Focusing on Seepage—The flow focusing factor implied by the new modeling can be bounded by an exponential distribution, with a minimum focusing factor of 1 and a mean focusing factor of 2. This distribution was substituted for the TSPA-SR base-case distribution for a TSPA sensitivity analysis. The comparison of the computed mean annual dose for the supplemental TSPA analysis with the TSPA-SR model base case is discussed in Volume 2, Section 3.2.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b).

The dose comparison shows little difference between the two TSPA simulations even though there is a significant difference in the amount of seepage. Because of the lesser amount of flow focusing in the sensitivity case, the mean seepage flow rate is lower in that case by nearly a factor of 10 (BSC 2001b, Section 3.2.2). By definition, the mean seepage flow rate is the seepage flow rate averaged only over locations that have seepage. With flow focusing, the percolation flux is higher in the locations that have percolation, which then produces higher seep rates in those locations. At the same time, less flow focusing makes the seepage fractions higher in the sensitivity case; approximately 50 percent more waste packages are exposed to seepage in that case than in the TSPA-SR model base case (BSC 2001b, Section 3.2.2.3). The number of waste packages that always receive seepage actually declines, but the number of waste packages that receive seepage only some of the time increases. A high flow-focusing factor increases the chances of seeping all the time, because the focusing enhancement can increase the local percolation flux above the seepage threshold flux even during the dry, present-day climate.

Comparisons of advective releases from the engineered barrier system for technetium-99 and neptunium-237 are discussed in Volume 2, Section 3.2.2 of *FY01 Supplemental Science and Perfor-*

mance Analyses (BSC 2001b). The advective releases are not as different as might be expected from the difference in seepage between the supplemental TSPA model sensitivity case and the TSPA-SR base case. The simulated releases largely are limited by the rate at which the waste inventory is exposed (e.g., by waste-package and cladding failure) and available for transport. In addition, much of the radionuclide release is diffusive rather than advective, especially for technetium-99 (CRWMS M&O 2000a, Section 4.1.2), and diffusive releases are not affected by seepage in the TSPA model. Together, the large amount of diffusive release and the relatively small change in the advective release as the seepage changes lead to the small change in doses (BSC 2001b, Section 3.2.2).

Effects of Episodic Flow on Seepage—A supplemental sensitivity analysis was conducted in which the episodicity distribution was included in the seepage abstraction, but all other parts of the model were the same as the TSPA-SR model base case. A comparison was made between the mean annual dose for that case and the base case in the supplemental TSPA model (BSC 2001b, Section 3.2.2), and the mean annual dose for the sensitivity case is higher than the TSPA-SR model base case after about 40,000 years. The first general-corrosion penetrations of the waste packages occur at about 40,000 years. Before that, no seepage water enters the waste packages in the TSPA-SR model (CRWMS M&O 2000a, Section 4.1.2).

Effects of Drift Degradation and Rock Bolts on Seepage—Recent results show that there is probably no significant increase of seepage because of drift degradation or the presence of rock bolts, but it was also noted that TSPA results are not expected to be sensitive to a change of only 50 percent in the seepage flow rates (BSC 2001a, Section 4.3.4.6). Because there is uncertainty about the effects of drift degradation on seepage, with significant increases in seepage possible in some locations, the 50-percent seepage enhancement is retained for the analyses in this report.

Thermal Effects on Seepage—Several recent supplemental analyses have been conducted to estimate the amount of seepage reduction during the period when there is a vaporization barrier around the drifts (BSC 2001a, Section 4.3.5). Those analyses found that little, if any, liquid flow reaches the drifts when there is an above-boiling zone around them.

An alternative model is evaluated in which seepage is reduced to zero when the drift wall is above boiling. This change has no effect on doses in the nominal scenario because all waste packages and drip shields remain intact until well past the boiling period in the TSPA-SR model base case (e.g., CRWMS M&O 2000a, Section 4.1). Thus, to determine if there is some effect, this supplemental sensitivity analysis was performed using a case in which there are no drip shields and there is a patch failure in each waste package at 100 years after closure.

Volume 2, Section 3.2.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) reports on a comparison of the mean annual dose for this supplemental sensitivity case with the case that has neutralized waste packages and drip shields and the TSPA-SR model base-case seepage model. The calculated doses are reduced for approximately the first 500 years because of the reduction in advective releases caused by eliminating seepage while the drifts are above boiling. Diffusive releases also are relatively low during this period (in both cases) because heat from the waste packages reduces the moisture content, and thus the diffusion coefficient, in the invert. Volume 2, Section 3.2.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) discusses time-histories of drift-wall temperature that were used for the five infiltration bins. Information about the infiltration bins and the thermal hydrologic model can be found in Sections 3.3.2 and 3.3.3 of *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a). The drift walls are only above boiling temperature (at the repository elevation, the boiling temperature is approximately 96°C [205°F]) for about 300 years or less, depending on the infiltration bin; thus the supplemental seepage model is only changed from the base-case seepage model for about 300 years. The change in seepage for commercial spent nuclear fuel with 20 to 60 mm/yr (0.8 to 2.4 in./yr) infiltration and seepage some of the time is discussed in Volume 2, Section 3.2.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). As expected, the mean seepage flow rate is only different from the base case for a little over 300 years.

Combined Effects of Seepage—In this section, results are presented for a supplemental sensitivity analysis in which the changes of the preceding sections are combined. This alternative seepage model includes:

- The updated seepage abstraction (BSC 2001b, Section 3.2.2.2)
- The updated distribution for the flow-focusing factor (BSC 2001b, Section 3.2.2.3)
- The distribution of the episodicity factor (BSC 2001b, Section 3.2.2.4)
- The reduction in seepage by a sampled factor between 0 and 0.2 when the drift wall is above boiling (BSC 2001b, Section 3.2.2.6).

This supplemental sensitivity analysis was performed to see the effect of changes in seepage during the boiling period using the case with neutralized waste packages and drip shields (CRWMS M&O 2001a, Volume 2, Section 3.4.2). The mean annual dose for this supplemental sensitivity case was compared with the case that has neutralized waste packages and drip shields and the TSPA-SR base-case seepage model in Volume 2, Section 3.2.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). For the modified seepage model, the mean annual dose is about a factor of two higher than the TSPA-SR model at all times. Because the thermal seepage-reduction factor is sampled between 0 and 0.2, there is only one tenth as much seep flow, on average, as in the TSPA-SR model when the drift wall is above boiling. However, the seepage reduction apparently causes no reduction in dose during the boiling period similar to the reduction when seepage was eliminated (BSC 2001b, Section 3.2.2.6). The results show that even a small amount of seepage is enough to provide for release of the highly soluble species (in particular, technetium-99 and iodine-129). The neutralization of waste packages and drip shields makes this supplemental case more advection-dominated than was the TSPA-SR model, especially at early times (BSC 2001b, Section 3.2.2; CRWMS M&O 2000a, Section 4.1).

Releases from the engineered barrier system are computed for 30 environmental groups that are based on infiltration, waste type, and seepage condition. Each environmental group is associated with one of five infiltration categories. The range for the categories are 0 to 3 mm/yr, 3 to 10 mm/yr, 10 to 20 mm/yr, 20 to 60 mm/yr, and greater than 60 mm/yr.

The change in seepage for commercial spent nuclear fuel with 20 to 60 mm/yr (0.8 to 2.4 in./yr) infiltration and seepage some of the time is discussed in Volume 2, Section 3.2.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The mean supplemental seepage flow rate for seepage of 20 to 60 mm/yr for the combined seepage modifications is lower than in the TSPA-SR base-case seepage model by a factor of a little over two at late times and by a factor of ten or more during the boiling period. In these analyses, the mean annual dose increases slightly even though the mean seepage flow rate decreases because the number of waste packages exposed to seepage triples. Over one-third of the waste packages are in locations with seepage in the modified seepage model. The increase in seepage fraction occurs because two changes, lower flow-focusing factors and inclusion of episodicity, tend to increase the seepage fraction (BSC 2001b, Sections 3.2.2.3 and 3.2.2.4), while only one change, inclusion of data from the lower-lithophysal unit in the seepage abstraction, tends to reduce the seepage fraction (BSC 2001b, Section 3.2.2.2).

Analyses of In-Drift Thermal-Hydrologic Conditions—Sensitivity of calculated performance to in-drift temperatures is shown in the comparison of the results for the designs with and without backfill. The results for these two cases (CRWMS M&O 2000a, Section 4.6) suggest that there is little difference between the TSPA-SR model and supplemental TSPA model estimates of system performance (BSC 2001b, Section 3.2.3). However, the use of bounding approximations for assessing thermal effects limits their usefulness in sensitivity studies. The sensitivity to in-drift thermal-hydrologic conditions is inferred, to some extent, in studies (CRWMS M&O 2000a, Sections 5.2.1 and 5.3.1) that examined the range of flow conditions in the unsaturated zone. Variations in unsaturated zone percolation fluxes due to thermal influences result in variations in drift moisture fluxes (BSC 2001b, Section 3.2.3).

Drift Degradation—Supplemental sensitivity studies of drift degradation are reported in Volume 1, Section 6.3.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a). These studies examine uncertainties in the rock fall model multiplier and in the correction for sub-horizontal fractures, consider a wide range in the size of the fractures to determine the effects on rock fall, and are conducted using a large number of realizations in the Monte Carlo simulations. The results do not show significant increases in the estimate of the size or density of rock fall over that obtained previously. Consequently, TSPA calculations of their effects on the estimate of annual dose have not been conducted.

In-Drift Chemistry—Supplemental work regarding the chemistry of incoming seepage and the evolution of chemistry within the emplacement drift has been evaluated at the subsystem level (BSC 2001b, Section 3.2.4). Results of subsystem sensitivity studies of the thermal-hydrologic model and the precipitates and salts model are reported in Volume 1, Sections 6.3.1, 6.3.3, and 6.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a). These analyses have been incorporated in supplemental evaluations of performance.

Results using the supplemental TSPA model show no significant effect on the estimate of mean annual dose (BSC 2001a, Section 3.2.4). Although the solubilities of radionuclides are affected by these changes and the attendant effects on the in-drift chemistry, residence time in the invert is a sufficiently small fraction of the transport time that this effect apparently is not significant. This result is consistent with previous sensitivity studies (CRWMS M&O 2000a, Section 5.2).

Aging of Alloy 22—To evaluate the potential importance of aging due to phase instability of Alloy 22, an analysis was conducted using a supplemental TSPA model for these effects. In this supplemental TSPA model, the probability of aging enhancement to the corrosion rate was assumed to be 0.0001, and the effect of aging was assumed to enhance the general corrosion rate by a factor of 1,000. These values were used because they are considered conservative.

Supplemental TSPA model results using this analysis are discussed in Volume 2, Section 3.2.5 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). In addition to the change in the representation of the aging enhancement, this supplemental TSPA model reflects updates to the TSPA-SR base-case waste package and drip shield model discussed previously.

This supplemental analysis shows that the TSPA-SR model is conservative after 20,000 years. Before this time, the annual dose in the supplemental TSPA model is greater than that in the TSPA-SR base-case analysis; however, the estimated mean annual dose is small (i.e., a peak mean dose of 0.08 mrem/yr over a 10,000-year period), even using the extreme parameters of this model.

Stress Corrosion Cracking—Additional considerations of the sources of uncertainty in the stress corrosion cracking model (BSC 2001a, Section 7.3.3) result in supplements to the stress corrosion cracking model. The effects of these refinements on the estimate of mean annual dose are evaluated in three supplemental calculations that address the expanded range of considerations for the residual stress profile of the closure weld regions, the threshold stress for stress corrosion cracking crack initiation, and the orientation of the weld flaws (BSC 2001b, Section 3.2.5). The models used for these analyses are discussed in Volume 1, Section 7.3.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a), and updates to the TSPA-SR base-case degradation model are discussed in Volume 2, Section 3.2.5 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The effects of waste package degradation on all of these changes are summarized in

Volume 1, Section 7.4.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a).

General Corrosion—The comparison of the TSPA-SR model and supplemental TSPA model results using a probability distribution for general corrosion that reflects uncertainty shows a difference in the estimate of mean annual dose due to different parameter distributions and different conceptual models used. In the TSPA-SR model, conservative assumptions lead to earlier failure by corrosion and shorter waste package lifetimes. The neglect of uncertainty due to variability effectively narrows the temporal failure distributions for the drip shield and waste package associated with general corrosion. This narrowing leads to later initial failures of the drip shields and waste packages. The earliest release is delayed in the supplemental TSPA model compared to the TSPA-SR model because of this later initial failure of these components (BSC 2001b, Section 3.2.5).

A second source of uncertainty in the TSPA-SR model is the temperature dependence of the general corrosion rate. Supplemental analyses of the temperature dependence of the general corrosion rate for Alloy 22 have been conducted (BSC 2001a, Section 7.3.5). The net effect of the temperature dependence is that, while waste package temperatures are greater than 60°C (140°F) (for approximately the first 10,000 years), Alloy 22 general corrosion rates are higher than in the TSPA-SR model. After 10,000 years after closure, the general corrosion rate is lower and the net effect is that, on average, the rate of waste package failure is much lower in the supplemental TSPA model than in the TSPA-SR base-case model.

The results of the estimates of the annual dose using the updated supplemental waste package degradation rate are compared with the results using the TSPA-SR base-case (temperature independent) general corrosion model in Volume 2, Section 3.2.5 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). Volume 2, Section 3.2.5.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) compares the TSPA-SR base-case result to the results in which the temperature dependence of the general corrosion rate is developed from the potentiostatic polarization test results. Because the rate of waste package failures is much lower in the first 100,000 years in the supplemental TSPA model, the mean annual dose is significantly lower than the TSPA-SR base-case result. The peak mean annual dose during the first 100,000 years is approximately 0.1 mrem/yr. The variance in the estimate of annual dose using the supplemental temperature-dependent model is discussed in Volume 2, Section 3.2.5.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). Because of the larger variance in the degradation rate due to the treatment of uncertainty, variance in the supplemental estimate of annual dose using the temperature-dependent model is somewhat larger than that for the TSPA-SR base-case general corrosion model.

Early Failure of the Waste Package—In the TSPA-SR model, no releases occur before 10,000 years. In reevaluating the potential of early failure mechanisms and their potential consequences, a more conservative approach resulted in the inclusion of improper heat treatment leading to the possible subsequent failure of a few waste packages in the TSPA supplemental analysis. The nonmechanistic early waste package failure assumes failure of both the inner and outer Alloy 22 lids and the stainless steel inner lid. To ensure that the potential consequence of early waste package failures is treated conservatively, it is included in the nominal scenario, not as a sensitivity analysis. Volume 1, Section 7.3.6 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a) describes the model developed for the potential consequences of improper heat treatment of a waste package.

Evaporative Reduction of Seepage—Supplemental analyses (BSC 2001a, Section 8.3.1) indicate that, because waste packages remain intact during the period when evaporative reduction of the flow might be important, little effect on radionuclide release is expected. Comparison of supplemental analyses including this effect with the results using the TSPA-SR base-case model confirm this expectation: there is no discernible difference between the results with or without the evaporative reduction (BSC 2001a, Figure 3.2.6.1-1).

Condensation Under the Drip Shield—Supplemental sensitivity analyses using the TSPA-SR base-case model for invert and drip shield temperature indicate condensation under the drip shield throughout the potential repository, even in areas in which no seepage occurs. In this event, there would be advective flow throughout the repository during the early thermal period in spite of the drip shield.

Because the effects of condensation are not significant while the waste packages remain intact, the sensitivity analysis modeled the potential effect of condensation by assuming an early failure of a waste package (BSC 2001b, Section 3.2.6.2). For the supplemental sensitivity analyses, the fraction of water reaching the waste package is sampled from a uniform probability distribution that ranges between zero and one.

In the assumed scenario, the peak mean annual dose rate is increased in the first 10,000 years from about 10^{-2} mrem/yr to slightly less than 10^{-1} mrem/yr. However, the effect would be negligible in nominal scenarios in which waste packages do not experience the early failures. This effect remains screened from TSPA analyses for several reasons (BSC 2001b, Section 1.3).

- Thermal conditions required for water to condense under the drip shield are short-lived, limited only to the first few thousand years after emplacement when the waste packages remain largely unaffected by degradation (BSC 2001b, Section 3.2.6.2; see also Section 4.2.5 of this report).
- The model does not account for bias in the thermal model, which overestimates the potential for condensation to occur (BSC 2001b, Section 3.2.6.2).
- Condensate will tend to flow down the underside of the drip shield rather than falling directly onto the waste package (BSC 2001b, Section 3.2.6.2).

Geometrical Constraint on Flow through the Waste Package—The fraction of water flowing through the waste package may be different from the assumption used in the TSPA-SR model base case to define the fraction. There are geometrical constraints on this flow if the waste package breaches are not directly under the drip shield breaches. The supplemental analyses in Volume 1, Section 8.3.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a) develop a more realistic representation for this fraction. The difference in the resulting estimate of annual dose is discussed in Volume 2, Section 3.2.6.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b).

Patch breaches do not develop on the drip shield until well after a 10,000-year period. Consequently, the supplemental TSPA model shows no change in the estimate of mean annual dose from that calculated with the TSPA-SR base-case model. After 10,000 years, the geometrical constraint reduces the flux through the waste package, resulting in a reduction in the release of neptunium-237 and other solubility-limited radionuclides. The effect is a modest reduction in the mean annual dose due to this change in the source term (BSC 2001b, Section 3.2.6.3).

Bathtub Effect—Before development of any breach in the bottom of the waste package, water entering a breach in the top of the waste package would accumulate as in a bathtub. After a breach is formed in the bottom, the water could drain. The impact of this accumulation and subsequent drainage is considered in Volume 1, Section 8.3.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a).

The effects on estimate of mean annual dose rate projected by the supplemental TSPA model are analyzed in Volume 2, Section 3.2.6.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The effect on the estimate of annual dose is small. First, only a fraction of the waste packages in the potential repository are contacted by water, and only a fraction of these have a sufficient delay of breaching in the bottom (after breaching in the top) to provide a significant bathtub effect. In addition, although accumulations in this small fraction of waste packages can be substantial (on the order of 1 m^3 [35 ft^3] of water for an average waste package in an area with seepage), the accumulated water drains rapidly and has a short-term effect. Diffusive releases from the engineered barrier system depend only on the concentration of radionuclides in the water and not the additional mass and therefore are not affected. Advective releases from the engineered barrier system are affected but only over a short period, and the net effect averaged over an entire time step of several hundred years is small.

In-Package Chemistry—The TSPA-SR base-case model for in-package chemistry is a simple mixing cell (CRWMS M&O 2000a, Section 3.5.2.1). As indicated in Volume 1, Section 9.3.1 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a), the TSPA-SR model uses conservatively high dissolution rates. The supplemental TSPA analyses consider a model in which these dissolution rates are represented more realistically. Calculations of total system performance using this supplement in-package chemistry range are compared with the results using the TSPA-SR model in Volume 2, Section 3.2.7 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). Section 3.2.7 (BSC 2001b) also discusses the comparison of the mean annual dose curves calculated with the two models (TSPA-SR model and supplemental TSPA model) and the range of realizations calculated using the supplemental TSPA model.

The results show significant effects after about 40,000 years when the contributions from neptunium-237 and other actinides begin to dominate the estimate of mean annual dose (BSC 2001b, Section 3.1.1). The change in chemistry does not have a significant effect on the waste form degradation rate that controls the concentrations of technetium-99 or iodine-129, radionuclides that dominate the annual dose estimate in the first 40,000 years (BSC 2001b, Section 3.1.1). The modified water chemistry, however, affects the solubility limits of neptunium, plutonium, and other actinides; therefore, it affects the annual dose from the associated isotopes with concentrations that are determined by these solubility limits.

Commercial Spent Nuclear Fuel Cladding Degradation—The supplemental TSPA analyses (BSC 2001a, Section 9.3.3) consider information regarding the range of uncertainties in models for cladding degradation processes in addition to that considered in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a). In particular, the probability distribution for creep and stress corrosion cracking during dry storage is modified to reflect a more realistic representation of creep failure. The result is a reduction in the estimate of early cladding failures: the mean fraction of early cladding failures is reduced from about 8 percent to about 1 percent. The supplemental analyses (BSC 2001a, Section 9.3.3) also consider the probability distributions for localized corrosion and cladding unzipping. The ranges for these distributions are expanded to take into account additional uncertainty considerations. The effect in this case is greater variance in the distribution of cladding failures. The supplemental analyses also consider perforations due to static loading by fallen rock. These considerations lead to increased degradation of the cladding after sufficient degradation of the waste package and drip shield occurs to permit the rocks to rest directly on the waste form.

The effects of the additional cladding degradation uncertainty on the annual dose estimate are evaluated in Volume 2, Section 3.2.7.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The conceptual model for the cladding degradation is summarized in Volume 1, Section 9.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a), and the implementation for these analyses is described in Appendix A. The small reduction in the mean annual dose in the period projected by the supplemental TSPA model arises largely from the reduction in early cladding failure. The other changes to the supplemental cladding degradation model do not result in significant changes to the estimate of mean annual dose in the first 100,000 years (BSC 2001a, Section 3.2.7.2).

In-Package Radionuclide Solubility Limits—Supplemental analyses reported in Volume 1, Section 9.3.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a) consider a wider range for the uncertainty in the effect of the controlling phases for plutonium, neptunium, thorium, and technetium. The effects of the extended range of uncertainty in these concentration limits, along with the effects on in-package chemistry (e.g., pH), are assessed in Volume 2, Section 3.2.7.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). Volume 2, Section 3.2.7.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) compares the mean annual dose taking these supplemental effects into account with the results using the TSPA-SR base-case model. Volume 2, Section 3.2.7.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) analyzes the contributions of the various radionuclides to the total mean annual dose estimate when these effects are taken into account.

The results of the supplemental analysis (BSC 2001a, Section 3.2.7.3) show significant changes after the waste packages are breached. As indicated previously, neptunium-237 dominates the annual dose estimate in the nominal scenario (BSC 2001b, Section 3.1.1); therefore, changes to the solubility limit of this radionuclide have a large effect on the estimate of total mean annual dose. The estimates for other radionuclides also are reduced by the changes in solubility limits. In particular, the mean annual dose from the plutonium isotopes is reduced by about a factor of three. The overall effect of these supplemental changes is to reduce the estimate of mean annual dose by a factor of more than five.

In-Package Colloid Associated Radionuclide Concentrations—The supplemental TSPA analyses (BSC 2001a, Section 9.3.4) extend the range of uncertainties in these colloid-assisted concentrations and examined sensitivity over this range. The effect of using the colloid concentration and sorption model, considered in Volume 1 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a), is discussed in Volume 2, Section 3.2.7.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b), which compares the mean annual dose curves for the TSPA-SR base case and the new colloid models and presents the range of the results for the new colloid model. There is little difference between the mean annual doses calculated using these models. In part, the small difference reflects the fact that the mean annual dose is dominated by dissolved radionuclides. However, even the comparison of the results for the colloids alone would show little change in the mean annual dose because the mean values of the probability distribution in the two models are virtually the same.

Analyses of Radionuclide Transport in the Engineered Barrier System—Supplemental analyses of radionuclide transport in the engineered barrier system are described in Volume 1, Section 10.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a). These analyses include the treatment of diffusion and sorption in the engineered barrier system. The TSPA-SR base-case model does not account for diffusive transport processes within the waste package. Including in-package diffusive transport in the TSPA-SR model results in a modest reduction in dose calculations (BSC 2001b), Section 3.2.8, Figure 3.2.8-1)

One possible reason for the small difference in these two estimates is the diffusion coefficient used for transport within the waste package. Diffusive resistance to radionuclide transport has been shown to be sensitive to the conservative diffusion coefficient used in TSPA calculations, whether applied in the invert or internal to the waste package (BSC 2001b, Section 3.2.8; CRWMS M&O 2000a, Section 5.3.5).

The second consideration is the effect of sorption on radionuclide transport in the engineered barrier system, in particular, the effect of iron corrosion products on reversible and irreversible sorption in the invert and within the waste package. The TSPA-SR base-case model assumes no sorption of dissolved species within the engineered barrier system. However, Volume 1, Section 10.3.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a) considers the conservative nature of this assumption and develops a model for the sorption of radionuclides in the engineered barrier system.

The reduction in mean annual dose after 10,000 years projected by the supplemental TSPA model as compared to the TSPA-SR model derives from two principal effects. The first is the reduction in concentrations in the liquid phase due to the partitioning of the concentrations between the liquid phase and the solid phases not considered in the

TSPA-SR base-case model, the result of which reduces the source term of many of the contributing radionuclides. The second is the effect of sorption on the diffusive transport, which effectively reduces the diffusion coefficient and increases the diffusion resistance of in-package transport. These two effects combine to decrease the source term and the resulting mean annual dose estimate (BSC 2001b, Section 3.2.8, Figure 3.2.8-2).

Effects of the Drift Shadow Zone—In the TSPA-SR base-case model, radionuclide releases from the engineered barrier system are released into the fracture continuum of the unsaturated zone transport model (CRWMS M&O 2000a, Section 3.7.2). This choice is conservative in that fracture transport is faster than matrix transport.

As an initial estimate of the effect of the presence of the drift on unsaturated zone transport, a sensitivity analysis was performed in which advective releases from the potential repository were released into the fracture continuum of the unsaturated zone transport model, as in the TSPA-SR base case, but diffusive releases were released into the matrix continuum of the model instead (BSC 2001a, Section 11.3.1.6.1).

The sensitivity analysis uses the TSPA-SR base-case assumption of a zero-concentration boundary condition at the drift wall for calculating diffusive releases from the engineered barrier system (CRWMS M&O 2000a, Section 3.6.2.2).

The results of this sensitivity analysis are shown in a comparison of the mean annual dose for this case with that of the TSPA-SR model (BSC 2001b, Section 3.2.9.2). The results show a delay of approximately 10,000 years in the mean annual dose for the drift-shadow case as compared to the TSPA-SR model. The effect is as large as it is because a large portion of the radionuclide releases are diffusive, especially for technetium-99 (CRWMS M&O 2000a, Section 4.1) and because transport through the matrix is slower than transport through the fractures.

Unquantified Uncertainties in the Saturated Zone—The saturated zone flow-and-transport model was evaluated with respect to unquantified uncertainty to determine the parts of the model to change to provide a better representation of the saturated zone. A detailed discussion of the changes made to the saturated zone modeling is presented in Volume 1, Section 12.5.1 and Table 12.5.1-1 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a).

The impact of the changes made to the saturated zone flow and transport model by the evaluation of unquantified uncertainty is shown in Volume 2, Section 3.2.10 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b), which compares the mean annual dose for the TSPA-SR base-case model with the mean annual dose calculated using the supplemental saturated zone model. Volume 2, Section 3.2.10 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) discusses the results for the multiple realizations of the supplemental saturated zone model. There is virtually no difference between the mean annual doses calculated using the TSPA-SR saturated zone model and the supplemental saturated zone model (BSC 2001b, Section 3.2.10). This uncertainty evaluation provides confidence that the saturated zone modeling used in the TSPA-SR model is a relatively robust description of uncertainty in the saturated zone system and that the uncertainty included in the modeling was adequate for the TSPA-SR nominal case.

No Matrix Diffusion in the Saturated Zone—The mean annual dose calculated in the TSPA-SR model (CRWMS M&O 2000a, Section 4.1) is compared with the mean annual dose calculated without the effects of matrix diffusion in the saturated zone in Volume 2, Section 3.2.10 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). This section also discusses the results of the multiple realizations of the TSPA for the saturated transport model without matrix diffusion.The differences between the models with and without matrix diffusion are slight (BSC 2001b, Section 3.2.10.2.2).

Increased Matrix Diffusion in the Saturated Zone—Volume 2, Section 3.2.10 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) compares the mean annual dose calculated in the TSPA-SR model with the mean annual dose calculated with enhanced matrix diffusion in the saturated zone transport model, as described in Volume 1, Section 12.5.2.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a). Volume 2, Section 3.2.10 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) presents the results of the multiple realizations of the supplemental TSPA model for the saturated zone transport model with enhanced matrix diffusion. The differences in expected annual dose between the model with matrix diffusion and the model with enhanced matrix diffusion are generally less than 20 percent, and the simulated doses are somewhat lower for the model with enhanced matrix diffusion, as expected (BSC 2001b, Section 3.2.10).

Minimum Flow-Path Length in the Alluvium—The mean annual dose calculated for the TSPA-SR model base case is compared with the mean annual dose calculated with minimal alluvium in the groundwater transport path in Volume 2, Section 3.2.10 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). There is only a slight difference (generally less than 10 percent) in results, with the minimal-alluvium case showing slightly higher simulated dose, as expected. The results of the sensitivity analysis examining the minimum flow path length in the alluvium are approximately consistent with the expected behavior of the system, when the impacts of glacial climatic conditions and higher specific discharge are considered.

Increased Uncertainty in the Colloid-Facilitated Transport Models—The results of the colloid sensitivity study for the saturated zone are discussed in Volume 2, Section 3.2.10 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The simulated dose for changes only in the saturated zone flow-and-transport model are presented in Volume 2, Section 3.2.10 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). Doses for changes in all relevant components of the supplemental TSPA model (with regard to colloid-facilitated transport) are discussed in Volume 2, Section 3.2.10 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). There is virtually no difference in the mean annual dose using the supplemental representation of colloid-facilitated transport with increased uncertainty compared with the base-case colloid model used in the TSPA-SR model. The two radionuclides that comprise greater than approximately 70 percent of the annual dose in the TSPA-SR model (CRWMS M&O 2000a, Section 4.1) are technetium-99 at earlier times and neptunium-237 at later times. Both of these radionuclides are transported as solute and thus are unaffected by the new colloid model. Also, in the new colloid model, the means of the distributions (for colloid concentrations, sorption coefficients for radionuclides onto colloids, and sorption coefficients for radionuclides onto the rock matrix and alluvium) are similar to values used in the TSPA-SR model, and thus the mean behavior is not expected to be significantly different.

Updated Saturated Zone Flow and Transport Model for Supplemental TSPA Model Analyses—Simulated mean annual doses for the TSPA-SR base-case model (CRWMS M&O 2000a) are compared to the results of the supplemental TSPA model using the updated saturated zone flow and transport model in Volume 2, Section 3.2.10 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). These results indicate that the changes to the updated saturated zone flow and transport model have little overall impact on the simulated mean annual dose in the supplemental TSPA analyses. The influences of higher values of bulk density in the alluvium and lower sorption coefficients for iodine-129 and technetium-99 on the simulated annual dose tend to counteract one another. In addition, the mean annual dose is influenced by a few of the highest-dose realizations at any particular time in a TSPA simulation. If these highest-dose realizations are the ones in which the importance of the retardation of key radionuclides is diminished (e.g., by low sorption coefficients or short path length in the alluvium), then the impact of the updated values of alluvial bulk density on the mean annual dose would be minimal.

Analyses of the Biosphere—Since the biosphere modeling for the TSPA-SR model, the supporting documentation for the biosphere has been revised, including updates of some parameter values.

The relatively small effect of changes to the biosphere model on the expected annual dose for the nominal scenario is reflected in the TSPA results shown in Volume 2, Section 3.2.11 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The mean annual dose using the supplemental uncertainty distributions for the biosphere dose conversion factors (BSC 2001a, Section 13.4.) is compared to the results calculated in the TSPA-SR model (CRWMS M&O 2000a, Section 4.1). The mean annual dose is slightly lower for the new nominal-scenario biosphere dose conversion factors for all times shown in the plot. The reduction in dose from the previously calculated annual dose (TSPA-SR base-case results) to the new result is less than 10 percent in simulated annual dose for most times and does not constitute a large change.

The results of the supplemental TSPA model using the new volcanic eruption biosphere dose conversion factors are presented in Volume 2, Section 3.2.11 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The mean annual dose using the new uncertainty distributions for the volcanic eruption biosphere dose conversion factors in the supplemental TSPA model is compared to the results calculated in the TSPA-SR model in this figure. These results indicate that, at all times, the expected mean annual dose is approximately 2.5 times greater for the new volcanic biosphere dose conversion factors relative to the previous TSPA-SR igneous model. This represents an increase in the mean dose rate relative to previous results and is primarily due to the increase in the respirable fraction of particulate concentration in air within the model (BSC 2001a, Section 13.3.6.2). However, the higher expected annual dose from direct exposure to contaminated volcanic ash using the new volcanic eruption biosphere dose conversion factors is still lower than the expected annual dose from the igneous groundwater pathway at later times (see *Total System Performance Assessment for the Site Recommendation* [CRWMS M&O 2000a, Section 4.2]).

4.4.5.6 Evaluation of Disruptive Events

In this section, analyses conducted to examine the sensitivity of performance to new information related to disruptive events developed since publication of *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a) are described. Two potentially disruptive events are addressed: volcanism (i.e., igneous activity) and seismic events.

An uncertainty importance analysis was carried out for the TSPA-SR results (CRWMS M&O 2000a, Section 5.1) using various statistical methods to identify the most important contributors to the spread in the overall model results and to identify contributors to the extreme, or outlier, outcomes in the model results. The analysis showed that the most important parameters affecting the spread in model results are annual frequency of igneous intrusion and wind speed (BSC 2001b, Section 3.3.1).

4.4.5.6.1 Supplemental TSPA Model Igneous Disruptive Results

Igneous Event Wind-Speed Sensitivity—Volume 2, Section 3.3.1.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) discusses mean probability-weighted annual doses for the supplemental model volcanic eruption case, comparing results from the TSPA-SR model with the mean of a set of 300 realizations that are identical in all regards to the TSPA-SR model except that the alternative distribution for wind speed has been used. Using the Desert Rock Airstrip data as described in Volume 1, Section 14.3.3.5 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a) increases the probability-weighted annual doses by a factor of approximately 2 from TSPA-SR model.

Igneous Event Waste Particle Size—Volume 2, Section 3.3.1.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) discusses probability-weighted mean annual doses from the supplemental model eruptive case only, calculated for the seven waste-particle size distributions in Volume 1 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a, Table

14.3.3.4-1) and the TSPA-SR base-case distribution (CRWMS M&O 2000a, Section 3.10.2.2.2). All other input parameters in each case are identical to those used in the TSPA-SR model (CRWMS M&O 2000a, Sections 3.10.2 through 3.10.4). Calculated annual doses only vary by a factor of approximately 1.3 or less, and performance is relatively insensitive to uncertainty in waste particle diameter within the range considered in these analyses. Consistent with this observation, the distribution used in the supplemental TSPA analyses (BSC 2001b, Section 4) is unchanged from that used in TSPA-SR analyses (CRWMS M&O 2000a, Section 3.10.2.2.2).

Igneous Event Zone 1 and Zone 2 Sensitivity—Volume 2, Section 3.3.1.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) presents a comparison of the 20,000-year probability-weighted mean annual doses calculated for Zone 1 for the TSPA-SR model distribution and for the supplemental distribution. All other models and input parameters used in these cases are the same as those used in the TSPA-SR model (CRWMS M&O 2000a, Sections 3.10.2.2 through 3.10.2.4). The revised distribution results in an increase in the calculated annual dose at all times, with a maximum change of a factor of approximately 2. Volume 2, Section 3.3.1.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) presents the set of realizations calculated for Zone 1 with the revised distribution, with the 95th and 50th (median) curves shown with the mean. Note that the 5th percentile curve plots below the lowest value shown on the y-axis. Volume 2, Section 3.3.1.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) present the same comparison for Zone 2. There is little or no change in the mean annual dose from Zone 2.

Conditional Igneous Events—Conditional mean annual dose histories were calculated for eruptive events at 100, 500, 1,000, and 5,000 years (BSC 2001b, Section 3.3.1.2.4). The mean annual dose history for an event at 100 years is repeated from Volume 2, Section 3.3.1.2.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b), and the mean annual dose histories for events at later times are each derived from 300 realizations analogous to those shown for the 100-year event (BSC 2001b, Section 3.3.1.2.4). The similarity of the curves is consistent with the use of the same sampling of input parameters, and the differences in the initial annual dose at different times is due entirely to radioactive decay. The conditional mean dose in the first year for an eruptive event at 100 years is approximately 13 rem/yr (1.3×10^4 mrem/yr). The first-year conditional dose decreases to approximately one half this level by 500 years after closure, and is approximately 10 percent of this value after 5,000 years.

The conditional eruptive annual doses described in Volume 2, Section 3.3.1.2.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) do not include any dose that might be incurred by direct inhalation of the ash cloud during the eruptive event.

Volume 2, Section 3.3.1.2.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) discusses four annual dose histories discussed previously, with the addition of a conditional mean annual dose history calculated for an eruption occurring at 100 years and with the soil removal rate set to zero. This additional case was calculated to provide graphical confirmation of the relative roles of soil removal and radioactive decay, and it is not intended to represent a realistic estimate of annual doses following a conditional eruption. Soil removal due to agricultural processes is included as part of the set of realistic and reasonable models used in *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a) and Volume 2, Section 3.3.1.2.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). This final calculation confirms that the decrease in annual dose in the years following an eruption primarily is due to soil removal. The gradual decrease in annual dose after year 100 in this calculation is due entirely to radioactive decay, and the curve therefore intersects the first-year annual doses calculated for events at later times.

Volume 2, Section 3.3.1.2.4, Figure 3.3.1.2.4-4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b) shows 500 out of the 5,000 realizations of 50,000-year igneous intrusion annual dose histories calculated for TSPA-SR model (CRWMS M&O 2000a, Section 4.2).

Results shown in this plot are identical to Figure 3.3.3-1 in Volume 2, Section 3.3.1 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b), except that the eruptive releases have been removed. Groundwater releases for each realization are shown weighted by the probability of the intrusive event occurring. In Volume 2, Section 3.3.1.2.4 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b), the same set of realizations are shown without probability-weighting. Peak mean annual dose from the igneous intrusion pathway increases from approximately 0.1 mrem/year in the probability-weighted case to approximately 500 mrem/year, consistent with the overall mean probability of an intrusive igneous event during the 50,000-year simulation of 8×10^{-4}.

Peak conditional annual doses associated with volcanic eruption are significantly higher than those associated with igneous intrusion (BSC 2001b, Section 3.3.1.2.4).

4.4.5.6.2 Supplemental TSPA Model Seismic Activity Analyses

Volume 1, Section 9.3.3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a) considers a broad range of events and accounts for additional uncertainty in the magnitude of damage during the vibratory ground motion event. Volume 1 analyses consider the probability distribution shown in Volume 2, Table 3.3.2-1 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b). The supplemental results using this distribution are compared with the TSPA-SR base-case model results shown in Volume 2, Section 3.2.7.2 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b, Figure 3.3.2.1-1). The differences between these estimates of mean annual dose are small, and the uncertainty and extended range for the probability distribution for damage due to seismic activity have a negligible effect on the estimate of mean annual dose.

4.4.5.6.3 Supplemental TSPA Model Sensitivity Analyses for the Human Intrusion Scenario

No additional sensitivity analyses were performed on the human intrusion scenario by the supplemental TSPA model because the TSPA-SR model results were considered robust. The possible impact of unlikely igneous activity on human intrusion is analyzed and discussed in *Total System Performance Assessment Sensitivity Analyses for Final Nuclear Regulatory Commission Regulations* (Williams 2001b) and is summarized in Section 4.4.4 of this report.

4.5 MULTIPLE BARRIER ANALYSES

Evaluating the significance of the various natural and engineered barriers requires analyzing their relative importance in containing and isolating radioactive waste from the environment. The objective of such analyses is to evaluate the effectiveness and diversity of the barriers to determine the overall resiliency of the repository system to extreme conditions that are unlikely but within the range of those believed physically possible. These analyses address the nominal performance of the system (i.e., without the occurrence of disruptive events, such as igneous activity) and consist of the following elements:

- Identifying the natural features of the geologic setting and the design features of the engineered barrier system that are considered important to waste isolation
- Describing the capability of each of these barriers and their component features to isolate waste
- Identifying those aspects of each feature that provide assurance of acceptable overall performance of the system even if components of the barrier are degraded, deteriorated, or altered.

The following sections summarize the analyses that describe and evaluate the significance of the multiple barriers in the design of the potential repository at Yucca Mountain. Detailed discussions and analyses are presented in the *Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations* (CRWMS M&O 2001a, Volume 2) and *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a, Section 5.3).

4.5.1 Identification and Description of Barriers

The barriers important to waste isolation at Yucca Mountain may be broadly categorized as natural barriers (associated with the geologic and hydrologic setting) and engineered barriers. The engineered barriers complement the natural barriers by prolonging the containment of radionuclides within the repository and limiting their eventual release. The natural barriers consist of the following:

- Surficial soils and topography, which limit water infiltration
- Unsaturated rock layers above the repository horizon, which limit water flux into repository drifts
- Unsaturated rock layers below the repository horizon, which limit radionuclide transport
- Volcanic tuffs and alluvial deposits below the water table, which limit radionuclide transport in the saturated zone.

The engineered barriers consist of the following:

- A drip shield, which limits the water contacting the waste package and the water available for advective transport through the waste package and invert
- A waste package, which limits the water contacting the waste form
- Cladding, which limits the water contacting the commercial spent nuclear fuel portion of the waste
- A waste form that limits the rate of release of radionuclides to the water that contacts the waste
- A drift invert, which limits the rate of release of radionuclides to the natural barriers.

Directly or indirectly, the identified barriers affect several of the key attributes that influence the ability of the natural and engineered systems to contain and isolate wastes from the biosphere. The key attributes, as detailed in Section 4.2, include: (1) limited water entering emplacement drifts; (2) long-lived waste packages and drip shields; (3) limited release of radionuclides from the engineered barrier system; (4) delay and dilution of radionuclide concentration by the natural barriers; and (5) low mean annual dose considering potentially disruptive events. The potential effects associated with disruptive processes and events are addressed in Section 4.3.2. Section 4.2 also presents the technical basis and uncertainty associated with the projected performance of each barrier and describes the physical and chemical processes that operate on or within each barrier.

Table 4-39 illustrates the correlation between the attributes, barriers, and the processes as described in Section 4.2. The process models indicated in Table 4-39 form the individual component models of the TSPA-SR model. As indicated in Table 4-39, multiple processes must be analyzed to evaluate each barrier's ability to contribute to repository performance. Additional discussion of the model components of the TSPA-SR model are found in Section 4.2 of this report and in Section 3 of the *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a).

The supplemental and revised supplemental TSPA models have nearly identical component models. Detailed information on the supplemental TSPA model may be found in *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b), and information on the revised supplemental TSPA model may be found in *Total System Performance Assessment—Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain—Input to Final*

Table 4-39. Correlation of Barrier and Barrier Functions to Key Attributes of Yucca Mountain Repository System and Process Models

Key Attributes of System	Process Model	Barrier	Barrier Function
Limited Water Entering Emplacement Drifts	Climate	Surficial soils and topography	Reduce the amount of water entering the unsaturated zone by surficial processes (e.g., precipitation lost to runoff, evaporation, and plant uptake)
	Net Infiltration		
	Unsaturated Zone Flow	Unsaturated rock layers overlying the repository and host unit	Reduce the amount of water reaching emplacement drifts by subsurface processes (e.g., lateral diversion and flow around emplacement drifts)
	Coupled Effects on Unsaturated Zone Flow		
	Seepage into Emplacement Drifts		
	Coupled Effects on Seepage		
Long-Lived Waste Package and Drip Shield	In-Drift Physical and Chemical Environments	Drip shield around the waste packages	Prevent water contacting the waste package and waste form by diverting water flow around the waste package; therefore limiting advective transport through the invert
	In-Drift Moisture Distribution		
	Drip Shield Degradation and Performance		
	Waste Package Degradation and Performance	Waste package	Prevent water from contacting the waste form
Limited Release of Radionuclides from the Engineered Barriers System	Cladding Degradation and Performance	Spent fuel cladding	Delay and/or limit liquid water contacting spent nuclear fuel after waste packages have degraded
	Radionuclide Inventory	Waste form	Potentially prevent liquid water contacting waste during the period when temperatures in the commercial spent nuclear fuel waste packages are elevated and limit radionuclide release rates as a result low solubilities
	In-Package Environments		
	Commercial Spent Nuclear Fuel Degradation and Performance		
	DOE Spent Nuclear Fuel Degradation and Performance		
	DOE High-Level Radioactive Waste Degradation and Performance		
	Dissolved Radionuclide Concentrations		
	Colloid-Associated Radionuclide Concentrations		
	In-Package Radionuclide Transport		
	Engineered Barrier System (Invert) Degradation and Performance	Drift invert	Limit radionuclide transport out of engineered barrier system
Delay and Dilution of Radionuclide Concentration by the Natural Barriers	Unsaturated Zone Radionuclide Transport (Advective Pathways; Retardation; Dispersion; Dilution)	Unsaturated rock layers below the repository	Delay radionuclide movement to the groundwater aquifer because of water residence time, matrix diffusion, and/or sorption
	Saturated Zone Radionuclide Transport	Volcanic tuff and alluvial deposits below the water table (flow path extending from below the repository to point of compliance)	Delay radionuclide movement to the receptor location by water residence time, matrix diffusion, and/or sorption
	Dilution		
Low Mean Annual Dose Considering Potentially Disruptive Events	Probability of Volcanic Eruption	N/A	N/A: These process and events may affect performance but are not barriers
	Characteristics of Volcanic Eruption		
	Effects of Volcanic Eruption		
	Atmospheric Transport of Volcanic Eruption		
	Biosphere Dose Conversion Factors		
	Probability of Igneous Intrusion		
	Characteristics of Igneous Intrusion		
	Effects of Igneous Intrusion		

NOTE: N/A = not applicable.

Environmental Impact Statement and Site Suitability Evaluation (Williams 2001a).

The following paragraphs summarize the role that the barriers and processes play in the attributes of the Yucca Mountain site.

Barriers and Processes that Contribute to Limit Water Entering Emplacement Drifts—Several barriers combine to limit the amount and distribution of water that may enter emplacement drifts and come into contact with the waste packages. The natural barriers work together to limit water infiltration and seepage flux. Both the amount and distribution of water in the unsaturated rocks at Yucca Mountain are currently controlled by the arid climate regime, which limits annual precipitation in the area, and by the surficial bedrock and soils, which encourage significant loss of water through runoff and evapotranspiration. These processes combine to limit the annual amount of water that infiltrates though the surficial soils. The technical basis and uncertainty associated with the understanding of these processes as they affect performance are described in Section 4.2.1.

The flow characteristics of the welded tuff units at the potential repository horizon limit the amount and distribution of water that can seep into the repository. Much of the limited flux that reaches the repository horizon will be diverted around the drifts by the capillary suction within the fractured rock mass.

Barriers and Processes that Contribute to the Long-Lived Waste Package and Drip Shield—Any water that does seep into the repository drifts will be diverted away from the waste package by the titanium drip shield for as long as the drip shield remains intact. The purpose of this engineered barrier is to protect the waste package from direct contact with water that seeps into the drifts. Another important function of the drip shield is to limit the advective flux of water that can transport waste through the invert materials if waste packages breach earlier than expected during the lifetime of the drip shield. Drip shields also provide a mechanical barrier, protecting waste packages from potential damage from rockfall.

The most important barrier to radionuclide migration is the waste package itself. The characteristics and rates of waste package degradation depend on the environment. Those aspects of the natural environment that affect the performance of the waste package as a barrier consist of the thermal-hydrologic, thermal-chemical, and thermal-mechanical environments.

The waste package lifetime is affected by the degradation modes and rates of the corrosion-resistant Alloy 22 over the surface area of the waste package and at the closure welds. These degradation modes include general and localized corrosion, as well as stress corrosion cracking. The technical basis and uncertainty associated with these degradation modes and their corresponding rates are presented in Section 4.2.4.

In reevaluating the potential of early failure mechanisms and their potential consequences, a more conservative approach resulted in the inclusion of improper heat treatment and subsequent possible failure of a few waste packages in the TSPA supplemental analysis. The nonmechanistic early waste package failure assumes failure of both the inner and outer Alloy 22 lids and the stainless steel inner lid. To ensure that the potential consequence of early waste package failures is treated conservatively, it is included in the nominal scenario, not as a sensitivity analysis. Volume 1, Section 7.3.6 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001a) describes the model developed for the potential consequences of improper heat treatment of a waste package.

Barriers and Processes that Contribute to Limited Release of Radionuclides from the Engineered Barriers—Based on the analyses presented in Section 4.4.2, most waste packages are expected to remain intact for much more than 10,000 years. However, eventually the waste packages will degrade. Once the waste packages have been breached, the possibility exists for water to contact the waste and mobilize radionuclides. Even then, the waste forms (spent fuel or borosilicate glass) will limit the rate of radionuclide release because they are relatively stable solids that will degrade slowly. The thermal-hydrologic-chemical environment inside the waste package affects the rate and amount of degradation of the waste forms. In addition, in the case of most commercial spent nuclear fuel, a significant mechanism that delays the degradation rate of the waste is Zircaloy cladding.

Once radionuclides are released from the waste package, they must be transported through the invert materials before being released to the natural barriers. During the time when the drip shield is intact, or in instances of insignificant seepage into the drifts, the release of radionuclides through the invert is limited to diffusive processes. Depending on the diffusive characteristics of the invert materials, which depend on the in-drift thermal-hydrologic environment, diffusive processes may be a significant barrier to radionuclide migration.

Barriers and Processes That Contribute to Delay and Dilution of Radionuclide Concentration by the Natural Barriers—Once radionuclides have been released from the waste packages and transported through the engineered invert materials, they must then migrate through the unsaturated rock between the repository and the water table, and then through approximately 18 km (11 mi) of saturated volcanic rocks and alluvial deposits between the repository and the point at which the receptor uses the water. Both the unsaturated and saturated rock units provide natural barriers to the migration of these radionuclides. Their combined effect is to limit the mass breakthrough and, therefore, the concentration of radionuclides in the accessible environment.

Processes That Affect Whether Disruptive Events Will Compromise Repository Performance—Because an igneous intrusion or volcanic eruption could degrade the performance of both the natural and engineered barriers of the site, it is important to evaluate the extent to which these processes would affect performance. Evaluations of volcanic and igneous intrusion scenarios consider both the likelihood and consequences of each type of event, including analyses of potential exposure pathways.

The following section details the approaches taken in evaluating the contribution of the various barriers and the significance of the uncertainty associated with each of the natural and engineered barriers on overall system performance.

4.5.2 Approaches to Evaluation of Multiple Barriers

Three separate approaches were used to evaluate the contribution that different barriers provide to the overall performance of the repository system in the absence of disruptive events. The first approach, documented in Section 4.4.2, details the intermediate performance of the total system as the system evolves over time. This approach describes the results of the system's temporal and spatial evolution and the uncertainty in this evolution. For example, the rate of degradation and the mass release rate for several key radionuclides, as well as the uncertainty associated with this evolution, were identified and analyzed for several discrete spatial locations (the waste package, the edge of the invert, the base of the unsaturated zone, and at the accessible environment approximately 20 km [12 mi] from the repository footprint). Examining the intermediate results both in space and time, as detailed in Section 4.4.2, provides one level of indication of an individual barrier's contribution to the overall performance of the system.

The second approach for evaluating an individual barrier's contribution to the system performance has been termed "barrier-importance analysis." In this approach, the key parameters affecting the performance of each barrier are fixed at the extreme of their uncertainty distribution (either the 5^{th} or 95^{th} percentile, whichever leads to maximizing the dose rate over the time period of interest), and all other parameters and models are sampled stochastically as done in the base case. This partial degradation of the function of the barrier leads to more conservative dose projections. By comparing the TSPA-SR model nominal performance base-case results with the degraded performance results, one can examine the relative contribution of each of the barriers. The results of this approach are presented in Section 5.3 of *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a).

The third approach for evaluating a barrier's contribution to the system performance has been termed the "barrier-neutralization analysis." In this approach, the function a particular barrier(s) has in either reducing water contacting waste or reducing the transport of radionuclides is completely removed from the analysis (i.e., the capacity of the barrier(s) to limit the movement of water or radionuclides is set to zero). Neutralization analyses are the extreme of the partially degraded barrier analyses discussed above. They provide insights into the overall system behavior and the importance of the barrier(s) to defense in depth. Further discussion of barrier neutralization analyses are presented in *Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations* (CRWMS M&O 2001a, Volume 2).

Although the parameter values used in the partially degraded barrier analyses are within the range of values considered reasonably possible, the results should not be interpreted as representing the expected behavior of the system. The expected behavior of the system in the absence of igneous disruption is represented by the mean nominal performance result, and the degraded barrier analyses are presented only to provide insight into the resilience of the repository system to extreme conditions. To ensure a balanced interpretation of the degraded barrier analysis, results are shown paired with comparable analyses using the 5th or 95th percentile values (as appropriate) of the same parameters that result in improved performance. The mean result from the full nominal analysis should be interpreted as the best estimate of future performance, and both the degraded and improved performance analyses should be interpreted as being equally likely (or unlikely) to occur.

The following two sections detail the individual barrier analyses conducted to examine the robustness of the repository system's performance. Section 4.5.3 discusses the effects of the natural barriers on overall performance, and Section 4.5.4 covers the engineered barriers' effects on performance. Table 4-40 lists the figures presented in these two sections. In addition to the partially degraded barrier importance analyses, a representative barrier neutralization analysis is presented in Section 4.5.4. Additional details on these analyses and additional analyses are presented in the *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a, Section 5.3) and *Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations* (CRWMS M&O 2001a, Volume 2), as well as *FY01 Supplemental Science and Performance Analyses* (BSC 2001a; BSC 2001b).

4.5.3 Evaluation of Natural Barrier Components

There are two natural barriers above the repository that contribute to the long-term performance of the repository system: (1) the surficial soils and topography, which limit the amount of water that infiltrates into the rock; and (2) the unsaturated rocks, which limit the amount of water that seeps into the repository. Barrier-importance analyses have been conducted for each of these barriers.

The net surficial infiltration flux into Yucca Mountain is affected by uncertainty about future climate states and surficial processes. As explained in Section 4.2, the TSPA-SR analyses considered a range of infiltration rates. In partially degrading the surficial barrier, the assumption applied was that the maximum possible infiltration rate corresponding to the maximum anticipated climate state (the glacial-transition climate) prevailed throughout the duration of the 100,000-year simulated time period. To evaluate the consequences of a more positively performing surficial barrier, a separate scenario was analyzed, extrapolating the present-day climate's minimum infiltration rate throughout the entire duration of the simulated time period. The effects of these partial degradations of the surficial barrier are illustrated in Figure 4-208, which shows that the surficial infiltration rate over the range of likely conditions does not significantly affect the long-term performance of the repository system.

The net seepage into the repository drifts is affected by the percolation flux through the host rock and the rock properties around the emplacement drifts. To investigate the significance of seepage to system performance, a scenario was constructed that fixed (1) the infiltration rate at the maximum value of the three potential distributions,

Table 4-40. Partially Degraded Barrier Importance Analyses Figures

Key Attributes of System	Process Model Factor	Barrier	Figure
Limited Water Entering Emplacement Drifts	Climate	Surficial soils and topography	4-208
	Net Infiltration		
	Unsaturated Zone Flow	Unsaturated rock layers overlying the repository and host unit	4-209 and 4-211
	Coupled Effects on Unsaturated Zone Flow		
	Seepage into Emplacement Drifts		
	Coupled Effects on Seepage		
Long-Lived Waste Package and Drip Shield	In-Drift Physical and Chemical Environments	Drip shield above the waste packages	4-213
	In-Drift Moisture Distribution		
	Drip Shield Degradation and Performance		
	Waste Package Degradation and Performance	Waste package	4-214 and 4-218
Limited Release of Radionuclides from the Engineered Barriers	Cladding Degradation and Performance	Spent fuel cladding	4-215 and 4-218
	Radionuclide Inventory	N/A	
	In-Package Environments		
	Commercial Spent Nuclear Fuel Degradation and Performance	Waste form	4-216
	DOE Spent Nuclear Fuel Degradation and Performance		
	DOE High-Level Radioactive Waste Degradation and Performance		
	Dissolved Radionuclide Concentrations		
	Colloid-Associated Radionuclide Concentrations		
	In-Package Radionuclide Transport		
	Engineered Barrier System (Invert) Degradation and Performance	Drift Invert	4-217
Delay and Dilution of Radionuclide Concentration by the Natural Barriers	Unsaturated Zone Radionuclide Transport (Advective Pathways; Retardation; Dispersion; Dilution)	Unsaturated rock layers below the repository	4-210 and 4-211
	Saturated Zone Radionuclide Transport	Volcanic tuff and alluvial deposits below the water table (flow path extending from below the repository to point of compliance)	4-212
	Wellhead Dilution		
Low Mean Annual Dose Considering Potentially Disruptive Events	Probability of Volcanic Eruption	N/A	
	Characteristics of Volcanic Eruption		
	Effects of Volcanic Eruption		
	Atmospheric Transport of Volcanic Eruption		
	Biosphere Dose Conversion Factors		
	Probability of Igneous Intrusion		
	Characteristics of Igneous Intrusion		
	Effects of Igneous Intrusion		

NOTE: N/A = not applicable.

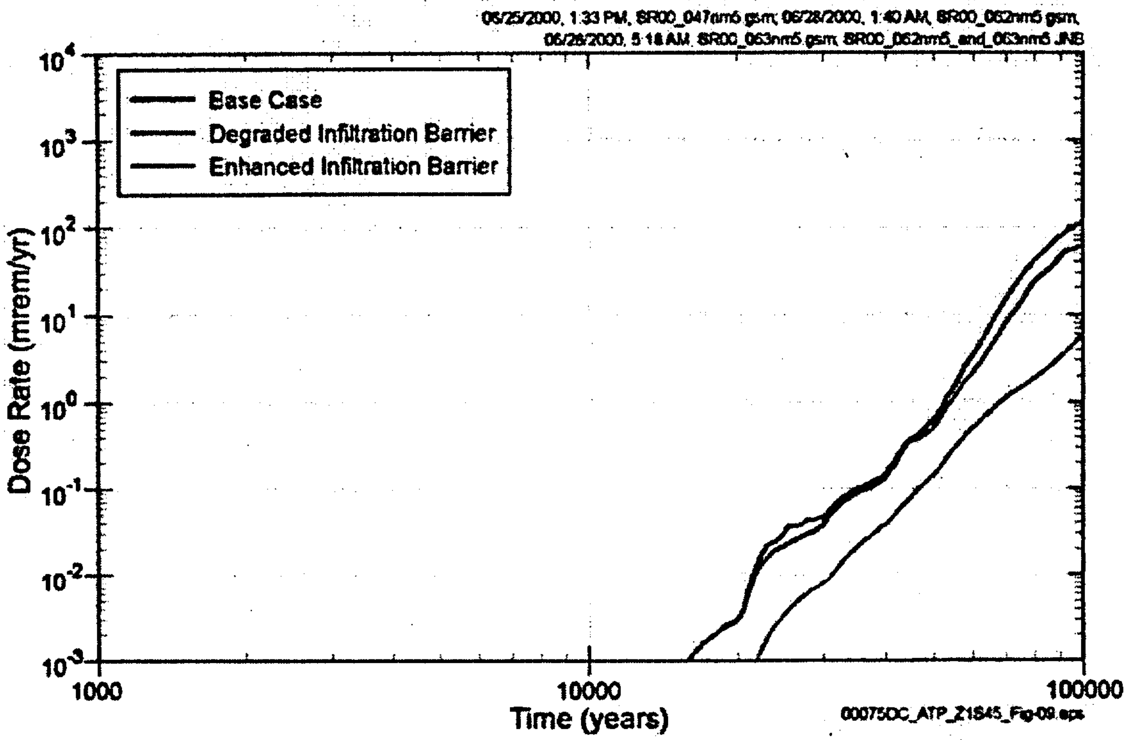


Figure 4-208. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Degraded and an Enhanced Infiltration Barrier
The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. Source: CRWMS M&O 2000a, Figure 5.2-1.

(2) the flow-focusing factor at its maximum to allow a greater fraction of the total percolation to come into contact with the repository drifts, (3) the seepage fraction at its maximum value, and (4) the seepage flow rate at its maximum value. All of these changes tend to force a greater fraction of the total percolation flux to seep into the repository emplacement drifts. To evaluate the consequences of a more positively performing seepage barrier, the above values were also fixed at their minimum values from the distributions presented in Section 4.2.

The effects of this partial degradation of the seepage barrier are illustrated in Figure 4-209. This figure shows that, for the first 40,000 years, the degradation (or improvement) of this barrier has little net consequence. This is because the waste packages are intact for most of this period, and even when waste packages are breached, the drip shield remains intact for a portion of this time. Even if both these barriers are breached, the dominant radionuclide contributing to dose is the mobile technetium-99, which can diffuse through surface water films relatively rapidly (even in the absence of significant advective flux) because of its extremely high solubility. Because the solubility-limited radionuclides like neptunium-237 are the radionuclides most affected by seepage, the significance of seepage to overall performance does not become pronounced until this radionuclide becomes the dominant dose contributor, at about 40,000 years.

There are two additional natural barriers below the repository block: the unsaturated zone rock units and the saturated zone rock units. Both affect the transport of both dissolved and colloidal radionuclides from the engineered barriers to the point where such radionuclides could be extracted with the groundwater.

The significance of the unsaturated zone transport barrier was evaluated by fixing the transport

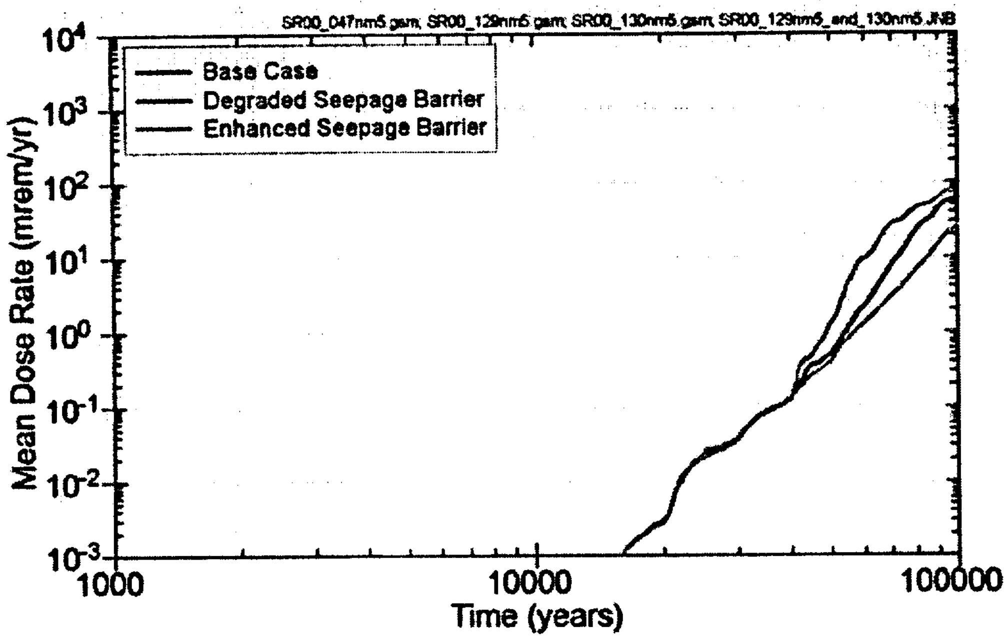


Figure 4-209. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Degraded and an Enhanced Seepage Barrier

The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. Source: CRWMS M&O 2000a, Figure 5.3-1.

parameters within the transport model to minimize retention in the unsaturated zone. This was accomplished by using the 5th percentile on sorption parameters for aqueous and colloid-borne radionuclides (K_d and K_c, respectively), the 95th percentiles on fracture apertures, and the 5th percentile on the matrix diffusion coefficient. To investigate the more optimistic end of the range of uncertainty, a similar analysis was conducted with the distributions at the opposite extremes. The results of this analysis are illustrated in Figure 4-210. Degrading the unsaturated zone transport barrier decreases the transport time of neptunium-237 through the geologic media and, therefore, increases the dose rate at a point 20 km (12 mi) from the repository footprint for a period from 20,000 to 40,000 years.

To evaluate the combined effect of the two unsaturated zone barriers (seepage in unsaturated rock layers overlying the repository and transport in unsaturated rock layers below the repository), a combined degradation analysis was performed. Figure 4-211 presents the results of this analysis; they illustrate that the combined effect of both barriers is approximately additive. This indicates that the contributions of the two unsaturated zone barriers are independent.

The significance of the saturated zone transport barrier is affected by several flow and transport parameters. In addition to the transport parameters described previously for unsaturated zone transport, the saturated zone flow fields are also uncertain, as are the possible flow path lengths through the alluvium and the alluvium transport characteristics. To investigate the significance of the saturated zone transport barrier, all realizations used the 95th percentile breakthrough curve (i.e., that distribution of mass flux that for a given release rate to the water table causes the 95th percentile mass flux at a location 20 km [12 mi] from the repository footprint). The results of this analysis are illustrated in Figure 4-212. These results indicate that the saturated zone barrier has minimal significance when it is degraded, since the mean dose response correlates closely to the more conservative portion of the saturated zone transport characteristics. However, the enhanced barrier

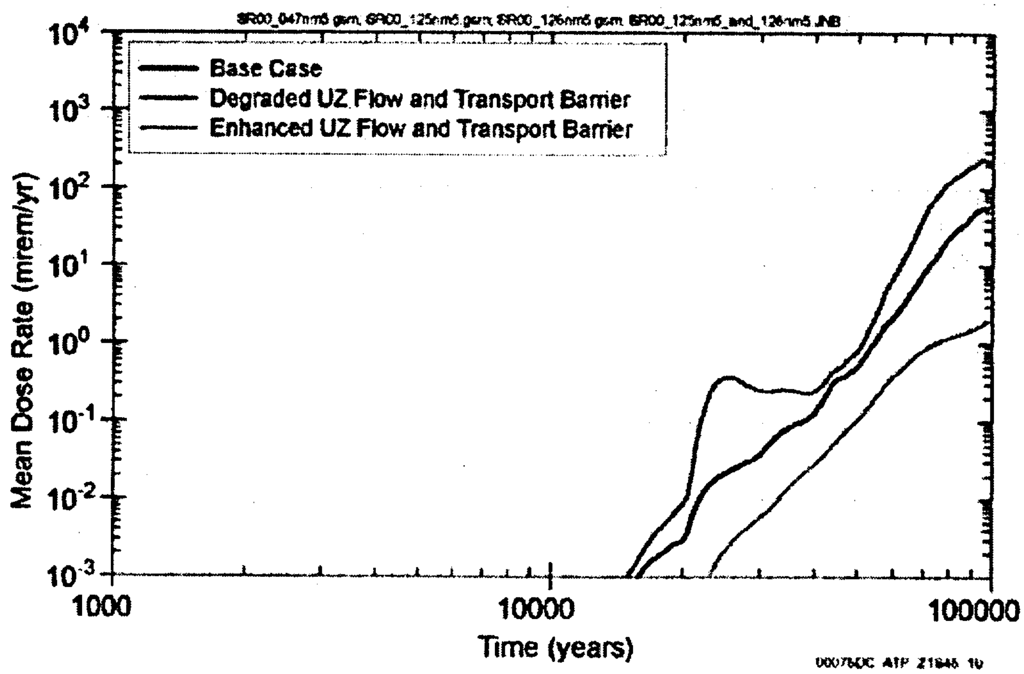


Figure 4-210. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Degraded and an Enhanced Unsaturated Zone Transport Barrier

The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. UZ = unsaturated zone. Source: CRWMS M&O 2000a, Figure 5.3-11.

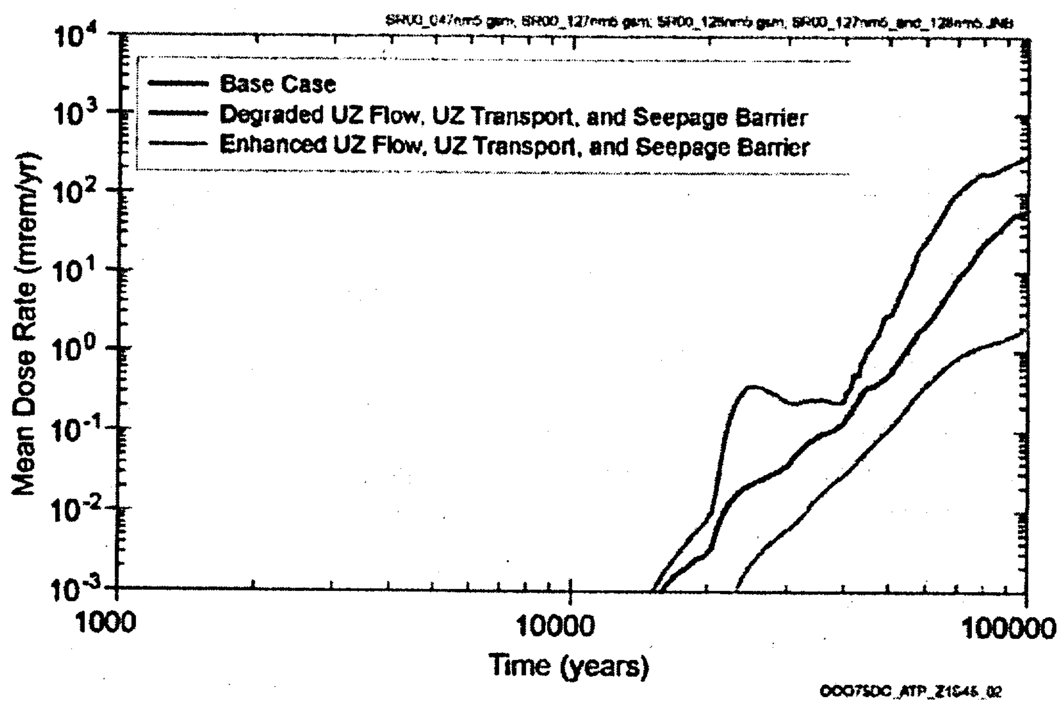


Figure 4-211. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Degraded and an Enhanced Unsaturated Zone Flow, Unsaturated Zone Transport, and Seepage Barrier

The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. UZ = unsaturated zone. Source: CRWMS M&O 2000a, Figure 5.3-12.

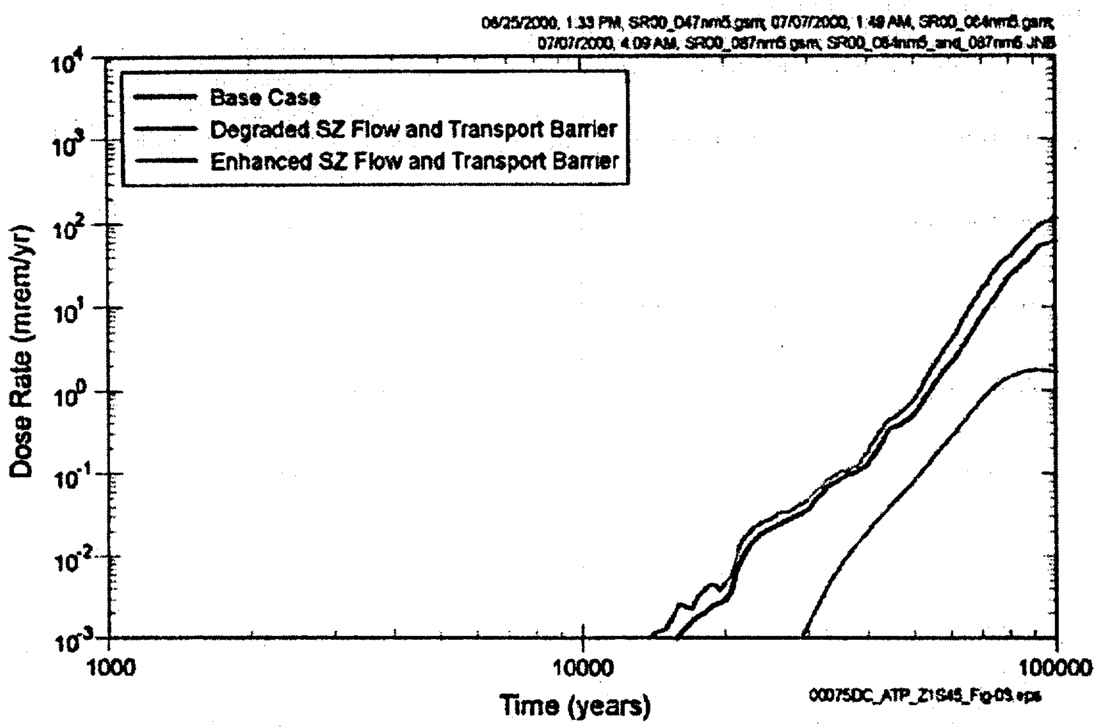


Figure 4-212. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Degraded and an Enhanced Saturated Zone Flow and Transport Barrier
The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. SZ = saturated zone. Source: CRWMS M&O 2000a, Figure 5.3-13.

performance of the saturated zone (which includes parameter assessments that are believed to be more realistic than the base case) is found to significantly delay the transport of neptunium-237 and therefore lower the expected dose.

4.5.4 Evaluation of Engineered Barrier Components

As shown in Table 4-39, the engineered barrier components consist of the drip shield, the waste package, cladding, the waste form, and the drift invert. The principal function of these engineered barriers is to contain the radionuclides as long as possible and, once the containment has been breached, to slow the release of radionuclides to the natural barriers. This section discusses the significance of each of the engineered barriers.

The significance of the drip shield barrier has been examined by fixing the corrosion rate of titanium at the 95th percentile of the distribution. This tends to decrease the lifetime of the drip shield from about 20,000 years to less than 10,000 years. The effect of this change on the calculated dose, however, is minimal (Figure 4-213). This minimal impact is a result of the degradation characteristics of the waste package and the form of the radionuclide release from degraded waste packages.

The titanium drip shield emplaced over the waste package has a modeled lifetime of about 20,000 to 40,000 years (see the results presented in Section 4.4.2 using the models and analyses summarized in Section 4.2). The waste package has a modeled lifetime of about 12,000 to over 100,000 years. As noted in Section 4.4.2, when the waste packages initially are breached, the more mobile radionuclide constituents (notably technetium-99 and iodine-129) can diffuse through the waste package and the invert materials and are not significantly affected by seepage through the degraded drip

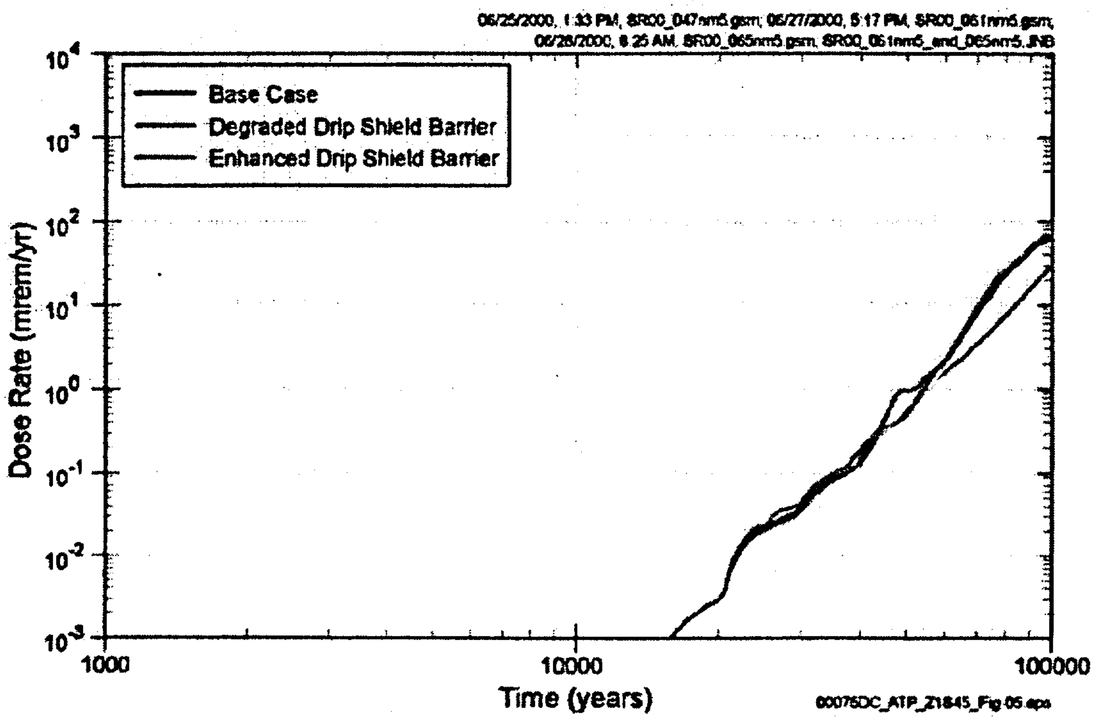


Figure 4-213. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Degraded and an Enhanced Drip Shield Barrier

The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. Source: CRWMS M&O 2000a, Figure 5.3-3.

shields. Once their degradation starts, the drip shields degrade over about a 20,000-year time frame, which minimizes the function they play once the waste packages are breached. As a result, the significance of the drip shield barrier is primarily as a defense-in-depth barrier in that it provides additional assurance in the unlikely event that waste packages may be prematurely breached by some unanticipated process or event as described in *Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations* (CRWMS M&O 2001a, Volume 2, Section 3.4.2).

Uncertainty analysis has determined that the waste package degradation parameters that most significantly affect degradation time are:

- Stress state at the stress-relieved closure welds
- Yield strength of the base metal at the closure welds
- Corrosion rate of Alloy 22, especially at the heat-affected zone associated with the outer and middle lid closure welds
- Accelerated corrosion rates associated with aging and microbially influenced corrosion, especially at the closure welds
- Size and frequency of defect flaws at the closure welds.

A degraded waste package barrier importance analysis was conducted in which each of these parameters was fixed at its 95th percentile value (except for yield strength, where the 5th percentile value leads to more conservative early waste package breaches). The results of this analysis are illustrated in Figure 4-214. The analysis illustrates

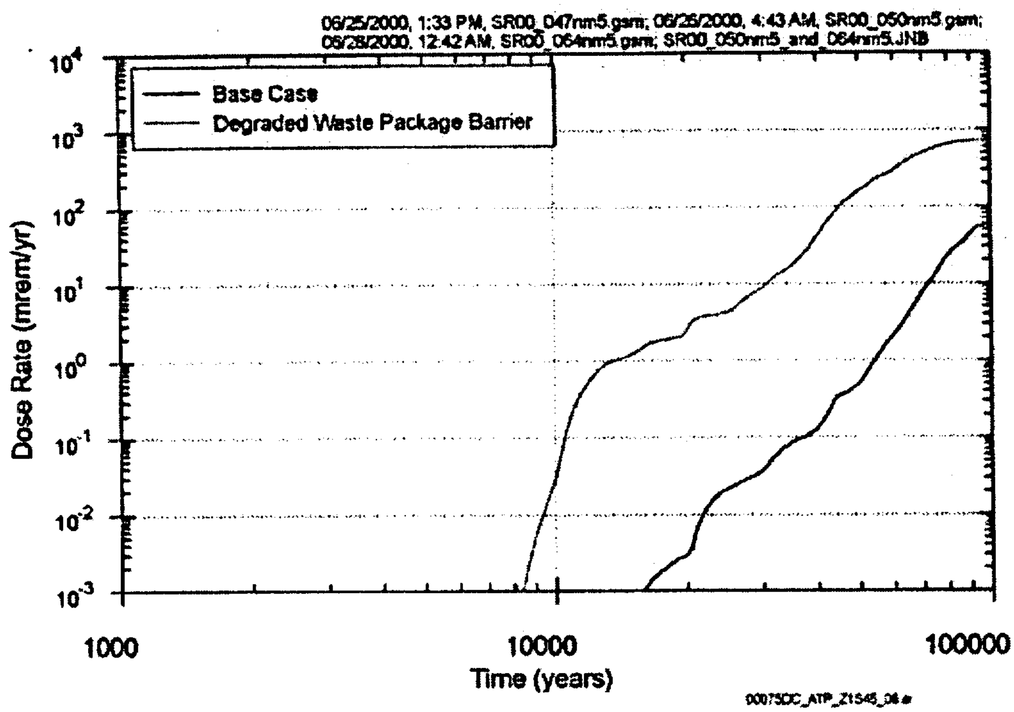


Figure 4-214. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Degraded and an Enhanced Waste Package Barrier

The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. The enhanced waste package barrier resulted in no dose during the first 100,000 years. Source: CRWMS M&O 2000a, Figure 5.3-5.

that earlier breaches of the waste package lead to earlier doses at the point of water extraction and, ultimately, to higher doses. Even the degraded waste package performance analysis, however, calculated doses that are small compared to the standards, due to the performance of the other natural and engineered barriers.

Once the waste packages are breached, the waste form may be exposed to moisture (in the form of humid air if there is no seepage or the drip shield is intact or liquid seepage if the waste is sufficiently cool and liquid water seeps into the waste package). In the case of commercial spent nuclear fuel, Zircaloy cladding provides another barrier, delaying the time at which moisture may directly contact the spent fuel. The degradation rate of the cladding, however, is uncertain. To evaluate the importance of this barrier, a scenario was constructed in which all fuel rods were assumed to be perforated at the time the waste package was breached, and the unzipping rate was set at the maximum value. The results of this analysis are illustrated in Figure 4-215.

Once the waste packages and cladding have been breached, the waste forms themselves must be altered, and the radionuclides must transition to a mobile phase before they can be transported out of the engineered barriers. The mobility of each radionuclide is a function of the geochemical environment and the sorption/desorption characteristics of that radionuclide on any mobile colloidal material (e.g., iron oxide, degraded fuel, silicon, or clay). The significance of this geochemical barrier was evaluated by using near-maximum solubilities in the waste package and invert materials and maximizing the sorption of radionuclides onto colloids. The results of this barrier analysis are shown in Figure 4-216. This figure shows, again, that prior to about 40,000 years, the dominant radionuclides are not controlled by the solubility or colloid concentrations. After 40,000 years, the soluble neptunium-237 and colloidal pluto-

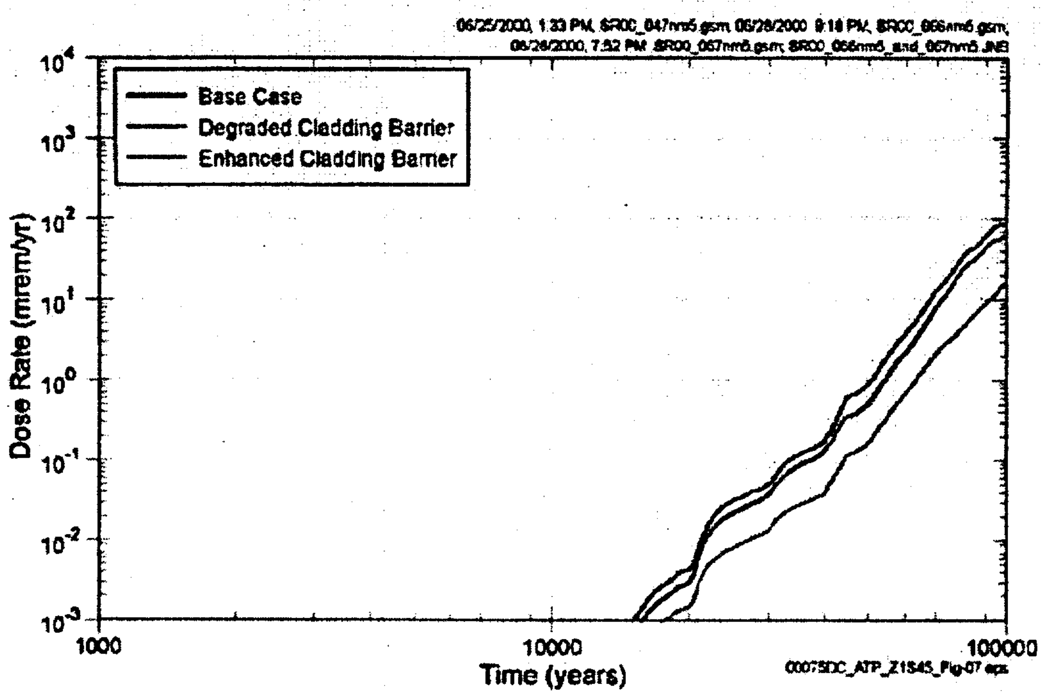


Figure 4-215. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Degraded and an Enhanced Cladding Barrier

The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. Source: CRWMS M&O 2000a, Figure 5.3-7.

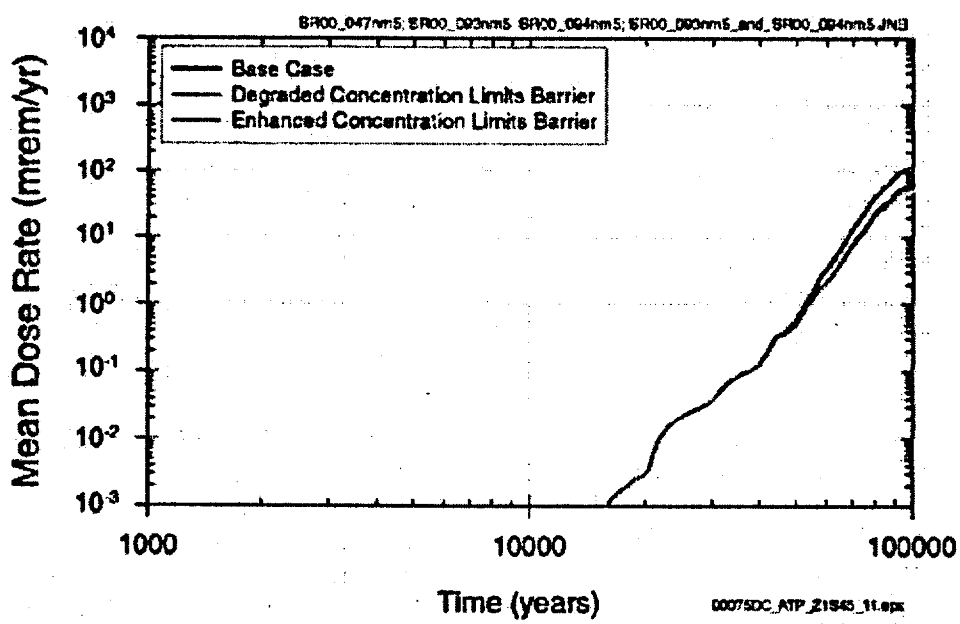


Figure 4-216. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Degraded and an Enhanced Concentration Limits Barrier

The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. Source: CRWMS M&O 2000a, Figure 5.3-8.

nium-239 are important, but degrading this barrier did not have an appreciable effect on overall performance.

In addition to the effect of the waste form and the mobility of different radionuclides, the drift invert materials themselves can significantly affect the release of radionuclides. Figure 4-217 illustrates the combined effect of modifying the in-drift chemistry and colloidal fraction with different diffusion coefficients through the invert. Increasing the diffusion coefficient (i.e., degraded engineered barrier system transport barrier) has the effect of allowing greater diffusive releases through the engineered barrier system, even during the time period that the drip shields are intact (i.e., prior to about 30,000 years). Decreasing the diffusion coefficient (i.e., enhanced engineered barrier system transport barrier) has the effect of allowing essentially no diffusive releases. In this later case, there are essentially no releases from the engineered barrier system at all, until such time as the drip shields have degraded sufficiently to allow advective water into the waste package.

In addition to the partial degradation analyses described previously, a range of barrier neutralization analyses have been conducted to gain additional insight into the defense in depth of the various natural and engineered barriers. An example neutralization analysis is to assume that a waste package has been inadvertently breached at the time it was emplaced. This nonmechanistic early failure of a waste package is outside of the expected degradation modes considered plausible. However, it does provide a useful "what-if" analysis for examining the robustness of the overall system behavior.

The results of the nonmechanistic early failure scenario are illustrated in Figure 4-218. This figure shows the effect of the nonmechanistic early failure waste package, assuming either the nominal cladding degradation model or the degraded cladding degradation model (the same as the model used to generate Figure 4-215). It illustrates that the time to achieve the initial peak dose is about 2,500 years, indicating the advective transport time through the unsaturated and saturated zone. The dominant radionuclide in this analysis is neptunium-237. The second peak of the dose at about 40,000 years occurs after the drip shields have been sufficiently degraded to have an effect on the advective release from the waste package. This peak is dominated by neptunium-237 and plutonium-239 releases.

4.5.5 Summary of Barrier-Importance Analyses

This section has presented a range of different barrier-importance analyses to evaluate the robustness of the repository performance even under extreme sets of assumptions. Although none of these combinations is expected to occur, they illustrate the behavior of the system over time periods beyond a 10,000-year period. These analyses give insights into the system behavior and are useful in evaluating the overall system response and in evaluating the contribution of the different natural and engineered barriers to the postclosure performance. As in Section 4.4.5, these analyses focused on the nominal performance scenario and the degradation of the barriers during the regulatory time period. Analyses beyond 10,000 years were performed as an indicator of long-term performance.

The analyses indicate that the various natural and engineered barriers contribute to the ability of the system to isolate waste. The total system results appear to be most sensitive to the performance of the waste package barrier. However, even degraded waste packages do not appear likely to result in doses higher than applicable standards because of the performance of other natural and engineered barriers.

Additional barrier-importance analyses are presented in the *Total System Performance Assessment for the Site Recommendation* (CRWMS M&O 2000a, Section 5.3) and *Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations* (CRWMS M&O 2001a, Volume 2), as well as in Volume 2, Section 3 of *FY01 Supplemental Science and Performance Analyses* (BSC 2001b).

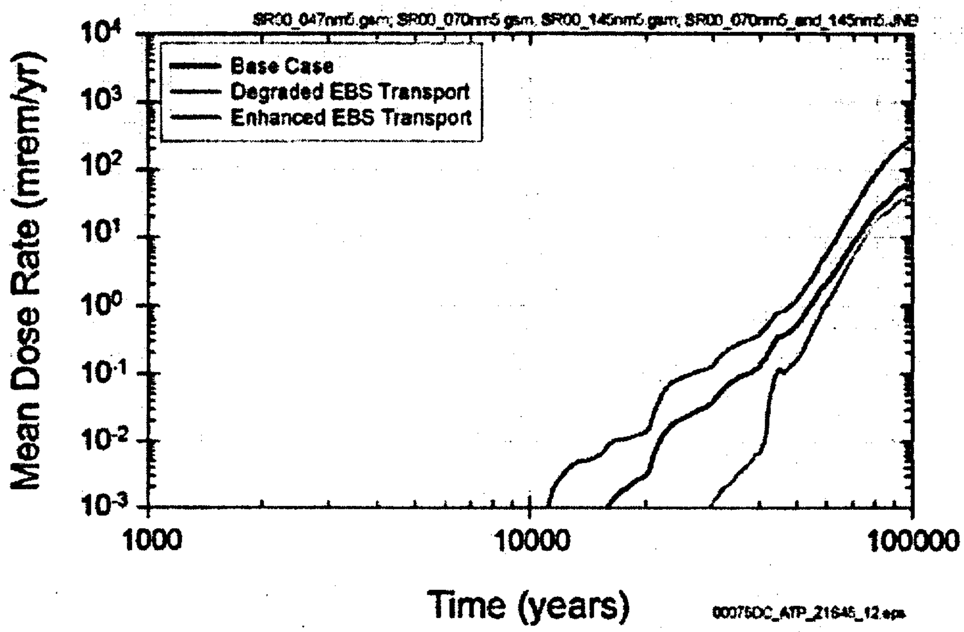


Figure 4-217. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Degraded and an Enhanced Engineered Barrier System Transport Barrier

The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. EBS = engineered barrier system. Source: Modified from CRWMS M&O 2000a, Figure 5.2-13.

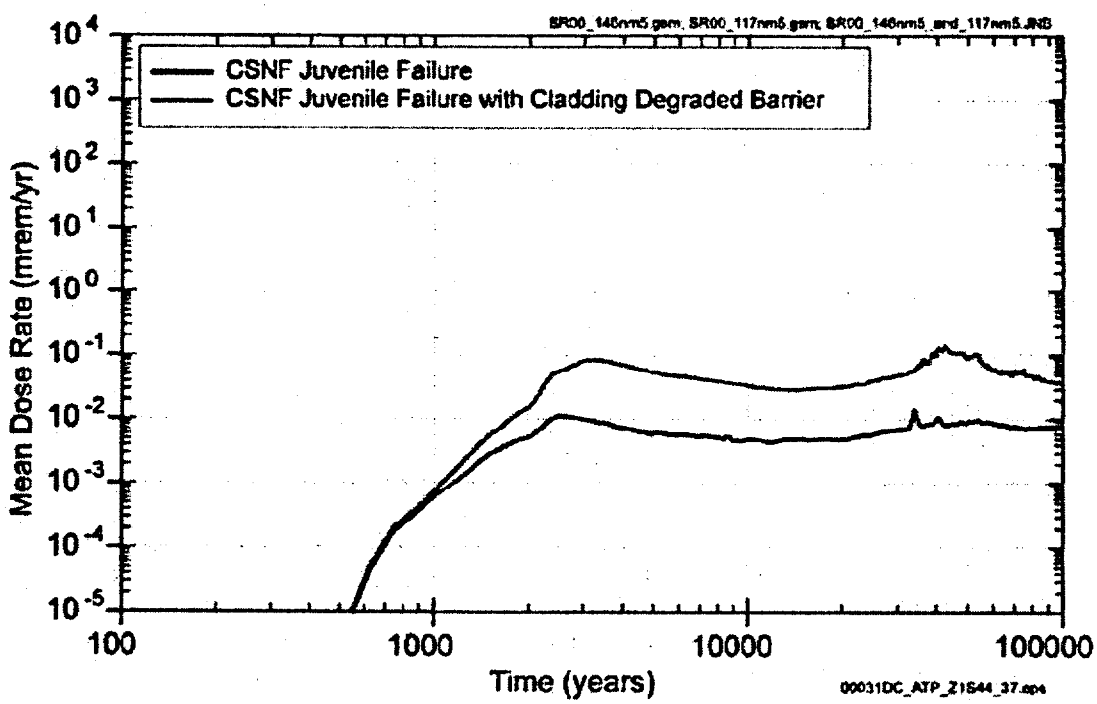


Figure 4-218. Sensitivity of the Mean Dose Rate Projected by the TSPA-SR Base-Case Model Assuming a Juvenile Waste Package Failure with a Degraded Cladding Barrier

The receptor is assumed to be 20 km (12 mi) from the repository footprint. Annual water demand was assumed to be approximately 2,000 acre-ft. CSNF = commercial spent nuclear fuel waste package. Source: Modified from CRWMS M&O 2001a, Figure 3-19.

4.6 PERFORMANCE CONFIRMATION, POSTCLOSURE MONITORING, AND SITE STEWARDSHIP

The quantitative and qualitative analyses of repository performance presented in previous sections are the basis for the DOE's forecast of repository performance. The DOE's confidence in the robustness of these analyses is enhanced by the safety margin and the defense in depth provided by the multiple natural and engineered barriers at Yucca Mountain. Nevertheless, the EPA, NRC, and DOE have all recognized that some uncertainty about repository performance cannot be eliminated. Furthermore, the DOE understands that ensuring public safety requires continued stewardship, including a program for evaluating new information obtained during the construction and operation of the potential repository. Therefore, the final component of the postclosure safety case is a performance confirmation, monitoring, and site stewardship program that accomplishes multiple goals related to the DOE's obligation to protect public health and safety and the environment. Specifically, these programs will include long-term monitoring of the site, maintaining the integrity and security of the repository, and maintaining the ability to retrieve spent nuclear fuel and high-level radioactive waste in the event that new information indicates a need to do so.

The postclosure monitoring and stewardship programs are derived from a variety of sources, including NRC licensing-related regulations and DOE policies. All parts of the program will contribute to the DOE's overall goal of building a repository that will provide for the containment and isolation of waste.

This section briefly describes the key elements of performance confirmation, postclosure monitoring, and site stewardship. These elements include a performance confirmation program to assess the performance of the repository both during operation and before closure. Other activities will include postclosure monitoring, maintenance of institutional control, site stewardship, and safeguards and security.

The objectives of the programs can be summarized as follows:

- Confirm that the repository is performing as expected or take any corrective action necessary, including retrieval of the waste, if warranted
- Provide a sound scientific basis for future generations to decide when to close the repository
- Institute safeguards and security provisions consistent with licensing-related NRC regulations
- Prevent deliberate or inadvertent human intrusion that could adversely affect the performance of the repository.

The components of the programs can be broadly grouped into two categories:

- Performance Confirmation and Monitoring (Section 4.6.1)
- Safeguards and Security (Section 4.6.2).

The following sections describe the activities which will continue or are currently planned or envisioned to be undertaken by the DOE under each of these categories. Data collection, experiments, and testing have been ongoing during site characterization and will continue consistent with NRC regulations.

4.6.1 Performance Monitoring

Performance monitoring will continue and will be conducted under the umbrella of an integrated test and evaluation program (see also Section 5.4) and will take place during all phases of construction and operation of the monitored geologic repository prior to closure. The *Monitored Geologic Repository Test & Evaluation Plan* (CRWMS M&O 2000fj) describes the program designed to ensure that systems are performing as expected to achieve the safety goals of this facility. Performance monitoring is subdivided into the following three categories, which stem from regulatory requirements:

- Preclosure safety monitoring
- Performance confirmation
- Post-permanent closure monitoring.

Preclosure safety monitoring is conducted to provide a safe working environment for onsite personnel and visitors during the preclosure period, which includes the site characterization, construction, operation, and monitoring phases of the program. Preclosure safety monitoring also provides for a variety of safety measures, such as fire sensors and alarms, in surface facilities, as well as radon gas measurement in the subsurface. Section 5.4 describes preclosure monitoring activities in more detail.

Performance confirmation is the set of testing, monitoring, and analysis activities initiated during site characterization and continue until repository closure. The focus of this monitoring is to gather and analyze data on conditions and systems that will affect the performance of the facility after closure and to evaluate their impacts on postclosure performance. These data will be used to confirm both that subsurface conditions are consistent with the assumptions used in performance analyses and that barrier systems and components operate within the bounds expected. The performance confirmation program will test, monitor, and analyze various systems of the monitored geologic repository. The program will include geologic mapping of subsurface conditions, monitoring of the conditions in and around emplacement drifts, monitoring of other factors important to postclosure safety, and assessment of postclosure implications. The program is described in more detail in Section 4.6.1.1.

A post-permanent closure monitoring program will include the monitoring activities that will be conducted around the repository after the facility has been closed and sealed. A license amendment will be submitted for permanent closure of the repository. This amendment will provide an update of the assessment of the repository's performance for the period after permanent closure, as well as a description of the program for post-permanent closure monitoring. The details of this program will be defined during processing of the license amendment for permanent closure. Permanent closure of the repository will include closing the subsurface facilities, decontaminating and decommissioning the surface facilities, reclaiming the site, and establishing institutional barriers. As part of the institutional barriers, provisions may be added for post-permanent closure monitoring, which will be described in the license amendment to obtain authorization to close the facility in the future. Deferring the definition of this program to the closure period will allow for identification of appropriate technology, including technology that may not be currently available.

The *Performance Confirmation Plan* (CRWMS M&O 2000ag) is a subtier document to the *Monitored Geologic Repository Test & Evaluation Plan* (CRWMS M&O 2000fj). The objective of the Monitored Geologic Repository Test and Evaluation Program is to define, design, and conduct testing for the repository to, among other things: (1) support system design and development, (2) evaluate operational suitability and effective-ness of the system, and (3) support the implementation of quality controls. Additional information about this program is provided in Section 5.4. The test and evaluation, performance confirmation, and postclosure monitoring programs will each have specific goals and objectives. However, they will also be designed to be sufficiently flexible to allow the DOE to respond to new information acquired as the repository system is observed both during and after construction and operation. If any unanticipated conditions are encountered or observed, the monitoring programs will provide a means for assessing their impact, if any, on long-term performance. As new insights are gained, existing tests may be modified or new tests defined to ensure that the DOE has a sound basis for future decisions affecting the repository.

4.6.1.1 Performance Confirmation Program

The performance confirmation program is an important part of the strategy for the development of the postclosure safety case for the potential repository (CRWMS M&O 2001a, Volume 2, Section 7.5). It consists of tests, experiments, and analyses that will be implemented in a manner that ensures that the ability of the repository to perform will not be compromised. The performance confirmation program will be periodically updated, for example, to conform to changes in repository design evolving from any licensing process.

In more detail, the aims of the program are to:

- Provide data to verify that subsurface conditions and any changes during construction and waste emplacement will fall within the limits assumed in the license application
- Provide data to verify natural and engineered barrier systems and components required for repository operations and important to postclosure performance (or designed or assumed to operate as barriers after permanent closure) are functioning as intended and anticipated
- Provide consistency with NRC licensing-related requirements for performance confirmation
- Provide information to support the authorization of permanent closure.

These objectives will be pursued in a manner that does not adversely affect the ability of the natural and engineered barrier systems to meet the performance objectives, as demonstrated in site performance protection analyses.

The description of the performance confirmation program will be formally documented in the *Performance Confirmation Plan*; a preliminary version of the plan has been developed (CRWMS M&O 2000ag) based on the repository design and higher-temperature operating mode. As the repository design matures and the thermal operating mode nears selection, the plan will be revised to support the license application and to provide additional details and specifics of performance confirmation activities.

Performance Confirmation Approach—The performance confirmation program will obtain data indicative of the repository's postclosure performance. Information relevant to the program has been developed during site characterization. Performance confirmation will extend until the beginning of repository closure operations. Key geologic, hydrologic, geomechanical, and other physical processes or factors (and related parameters) will be monitored and tested throughout construction, emplacement, and operation to detect any significant changes from baseline conditions.

Briefly, the overall approach to performance confirmation can be divided into eight steps:

1. Identify key performance confirmation processes (factors) and parameters
2. Establish the performance confirmation baseline database and predict the performance of the key factors and parameters
3. Establish tolerances and bounds for the key factors and parameters
4. Establish the completion criteria (which define when there is no longer a need for a test) and guidelines for corrective actions to be used when data exceed tolerances or bounds
5. Plan and set up the performance confirmation test and monitoring program
6. Monitor, test, and collect data
7. Analyze, evaluate, and assess data
8. Recommend corrective actions and implement changes to the performance confirmation program, as required.

The first four steps of the process define the performance confirmation baseline. The baseline includes the data obtained during site characterization and from test startup and initialization, together with parameter predictions and the definition of tolerances and bounds for these data. These four steps establish the basis on which to evaluate whether the data indicate that the related processes are performing as expected.

After this baseline has been established, related testing, monitoring, and analysis activities can be initiated. Testing and monitoring activities include the physical recovery and laboratory testing of representative repository materials and collection of environmental data. Examples of testing and monitoring activities include geologic mapping, subsurface sampling, coupon recovery, dummy waste package testing, and ventilation monitoring. This information is combined and analyzed to provide a comprehensive performance confirmation program.

Sensors in the emplacement and support drifts obtain raw electronic data that are reduced, processed, and stored. During this process, the raw data are converted into engineering numbers, which are transmitted as performance data to be evaluated during the performance confirmation process by comparing the data to predicted bounds for confirmation. Figure 4-219 illustrates this process schematically. The bounds or tolerances vary with time to reflect the effects of construction and operation of the repository, together with effects of waste emplacement.

Identifying Performance Confirmation Factors—An essential step in conducting performance confirmation and establishing the baseline is identifying the parameters to be monitored. Factors to be addressed by the performance confirmation program arise from four sources:

1. **Processes important to repository safety**—To correctly focus resources, the performance confirmation program concentrates on the processes (and related parameters) most important to repository postclosure safety. Identification of these processes (termed "performance confirmation factors") considers available performance assessment analyses and the *Repository Safety Strategy: Plan to Prepare the Postclosure Safety Case to Support Yucca Mountain Site Recommen-*

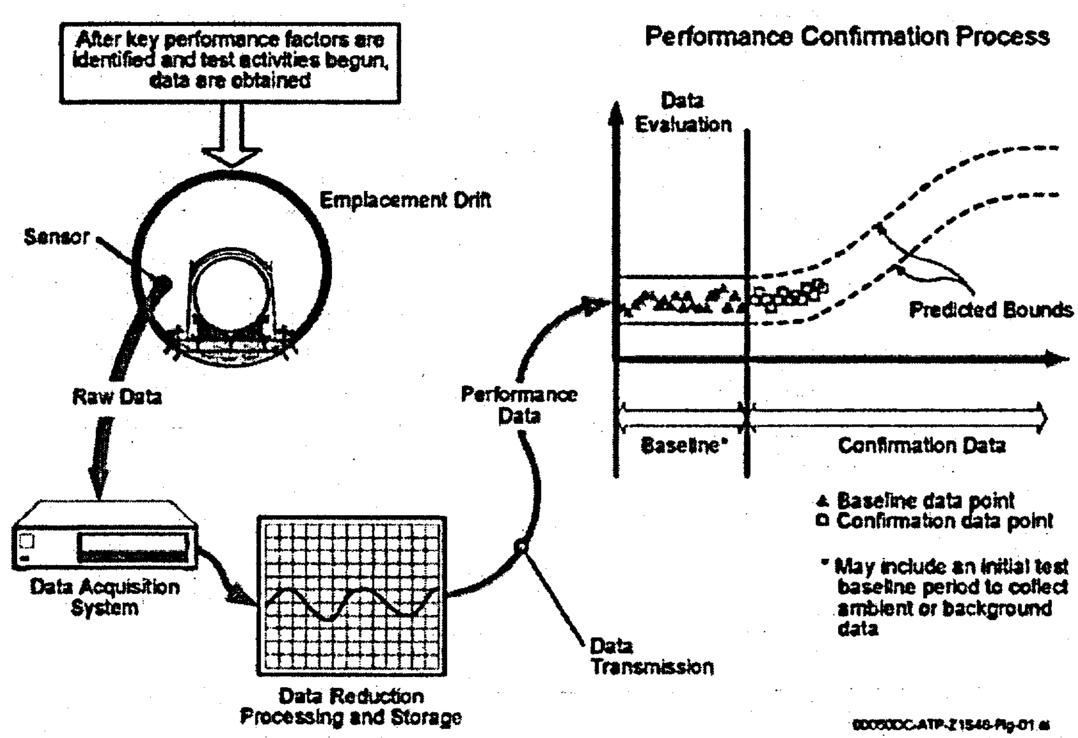


Figure 4-219. Performance Confirmation Process, From Testing to Data Evaluation
Data collected by in situ instrumentation is transmitted, reduced, and processed for analysis. Data observations are compared to predicted values to determine trends and potential deviations from expected performance.

dation and Licensing Considerations (CRWMS M&O 2000fk, pp. 3-7 to 3-10).

2. **Applicable requirements**—The program also must address applicable regulatory and system requirements as part of the licensing process. These requirements identify (or prescribe) particular factors or tests to be included in the performance confirmation program.

3. **Licensing conditions**—As part of the licensing process, requirements or issues that need additional testing may be identified by the NRC or the DOE and may be included in the license as a licensing condition or directive. Testing to address these conditions would be included as part of the performance confirmation program if the issues were important to postclosure safety or were so indicated in the directive.

4. **Data and validation needs of the analysis and process models**—After the completion of the site characterization phase, some additional data related to items that are recognized to have potential postclosure sensitivity may still be required to support the license application. These data needs will be identified in the application with associated test plans to be conducted as part of performance confirmation in accordance with regulatory guidance.

Based on these four sources, analyses have been conducted to identify the key performance confirmation factors to be tested or monitored. Table 4-41 lists the key performance confirmation factors identified in considering the processes important to repository postclosure safety. Table 4-42 shows additional factors that have been identified from evaluations of applicable system requirements and regulations (CRWMS M&O 2000ag, Table 3-7 and Appendix E). Table 4-43 shows the added key performance confirmation factors that have been identified in considering potential data needs in support of process model development. No factors have been identified to date based on licensing conditions. Specific tests that address these factors are discussed in the following section.

Testing Program—Performance confirmation activities will be conducted as part of the repository's overall test and evaluation program. They will be performed under several different test categories of the program. The preliminary list of test activities (and associated test categories) is presented in Table 4-44, and each performance confirmation activity is described briefly in the following sections (CRWMS M&O 2000ag). It is important to note that the performance confirmation program will continue to evolve consistently with the postclosure safety case, applicable licensing-related regulations, and applicable licensing requirements. Any or all of these may change and necessitate revisions to the plan, either before or after license application submittal.

Table 4-41. Performance Confirmation Factors Based on Processes Important to Safety

Performance Confirmation Key Factors	Type of Testing Indicated
Ambient flow through the repository horizon and seepage into unventilated drifts	In situ testing during construction and operation
Failure rate of drip shields	Laboratory testing of drip shield materials over a range of expected repository conditions combined with process modeling of drip shield failure
Amount and chemistry of water contacting waste packages	Process modeling and drift-scale testing to validate process model results
Corrosion rate of waste package materials and failure rate of waste packages	Laboratory testing of waste package materials over a range of expected repository conditions and process modeling of waste package failure
Potential factor of cladding degradation	Potential long-term laboratory testing of Zircaloy failure rate
Potential factor of solubility of actinides	Potential laboratory testing of solubility of actinides

Source: CRWMS M&O 2000ag, Table 3-5.

Table 4-42. Performance Confirmation Factors Consistent with Potential Licensing Requirement

Performance Confirmation Key Factors	Type of Testing Indicated
Waste package surface temperature to assess cladding condition	Monitoring waste package surface
Emplacement drift air temperatures to assess heat removal	Monitoring temperatures of both incoming and exhaust air of emplacement drifts
pH of water within emplacement drift (seepage) to confirm waste package environment	Monitoring the pH of water (as observed) within emplacement drifts and special test alcoves
Emplacement drift air temperatures to confirm waste package environment	Monitoring temperature within emplacement drifts
Emplacement drift humidity to confirm waste package environment	Monitoring humidity within emplacement drifts
Colloid content of water within emplacement drift to define waste package environment	Monitoring colloid content of water (as observed) within emplacement drifts
Drainage of invert to confirm free-draining conditions	Visually monitoring water accumulations within emplacement drifts
Amount of seepage to confirm expected waste package environment	Monitoring seepage in special test alcoves
Observation of the encountered subsurface (geologic) conditions of the repository horizon	Geologic observation, mapping, and index laboratory testing
In situ rock mass response due to repository construction and waste emplacement	Rock mass monitoring (temperature and displacement) near emplacement drifts
Performance and constructibility of borehole, ramp, and shaft seals	Field testing of borehole, ramp, and shaft seals
Engineered barrier system interaction response of waste packages, backfill (if used), rock, and groundwater	Field testing of engineered barrier system postclosure configuration
In situ waste package monitoring	Remotely monitoring waste package in emplacement drifts
Laboratory investigations of internal waste package material testing	Laboratory materials testing
Groundwater quality measurements	Well monitoring both downgradient (at accessible environment) and upgradient
Potential for disruptive events	Precise leveling surveys over repository Subsurface seismic monitoring Monitoring groundwater temperature and level

Source: Modified from CRWMS M&O 2000ag, Table 3-7 and Appendix E.

Table 4-43. Performance Confirmation Factors Based on Potential Data Needs

Performance Confirmation Key Factors	Type of Testing Indicated
Sorptive properties of the Calico Hills nonwelded hydrogeologic unit immediately below the repository	Sampling and laboratory testing of sorptive properties of the Calico Hills nonwelded hydrogeologic unit
Unsaturated flow and transport within the Calico Hills nonwelded hydrogeologic unit formation to validate conceptual models	Sampling and laboratory and field testing
Rock mass response to cooling	Monitoring rock mass thermal-hydrologic-mechanical response as in cooling of Drift Scale Test
Coupled thermal-mechanical-hydrologic-chemical processes	Field testing around heated test drift
Geochemical interactions as part of coupled processes	Laboratory testing
Stress corrosion cracking of barrier materials: Alloy 22; Titanium Grade 7; and Stainless Steel Type 316NG Long-term phase stability of Alloy 22 and Stainless Steel Type 316NG Long-term stability of passive film on barrier materials	Laboratory materials testing
Dissolved radionuclide concentration limits Colloidal radionuclide concentration and transportation limits Clad performance In-package chemistry Spent nuclear fuel degradation and high-level radioactive waste glass degradation	Laboratory materials testing

Source: Modified from CRWMS M&O 2000ag, Table 3-6.

Table 4-44. Identified Performance Confirmation Testing and Monitoring Activities

Test Category	Performance Confirmation Activity
Core Performance Confirmation	Seepage Monitoring
	In Situ Waste Package Monitoring
	Long-Term Materials Testing
	Ventilation Monitoring
	Rock Mass Monitoring
	In-Drift Monitoring
	Introduced Materials Monitoring
	Recovered Material Coupon Testing
	Dummy Waste Package Testing
	Recovered Waste Package Testing
	Postclosure Simulation Testing
Development Testing	Geologic Observations and Mapping
	Subsurface Sampling and Index Testing
	Baseline Analyses and Evaluations[a]
	Unsaturated Zone Testing
	Near-Field Environment Testing
	Waste Form Testing
	Waste Package Testing
Prototype Testing	Borehole Seal Testing
	Ramp and Shaft Seal Testing
Technical Specifications and Monitoring	Groundwater Quality Monitoring
	Groundwater Level and Temperature Monitoring
	Surface Uplift Monitoring
	Subsurface Seismic Monitoring

NOTES: [a]This activity supports all other performance confirmation activities and is included for completeness.
Source: CRWMS M&O 2000ag, Table G-1.

4.6.1.1.1 Core Performance Confirmation

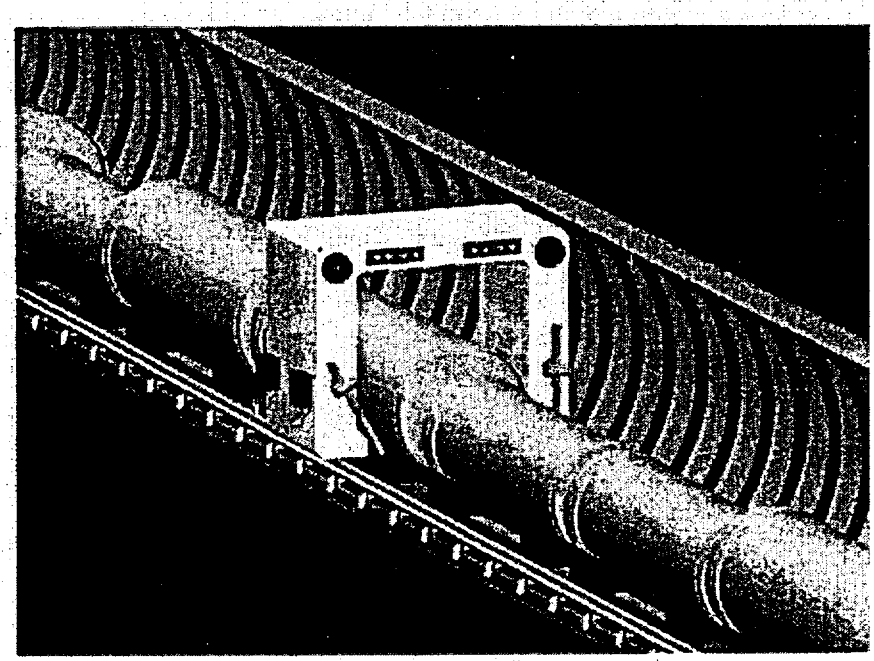
Core performance confirmation activities, as defined in the *Performance Confirmation Plan* (CRWMS M&O 2000ag), are the monitoring and testing activities focused solely on the postclosure performance of the repository. These core activities generally begin with the start of waste emplacement. Conceptually, activities under this test category include the testing and monitoring activities discussed in the following paragraphs.

Seepage Monitoring—This monitoring evaluates the ambient flow of water into excavations (i.e., seepage). Hydrologic testing and monitoring will be performed in closed alcoves along access or monitoring drifts. The location of the alcoves will be determined by applying such criteria as areas of relatively high and low infiltration; overlying bedrock/alluvium contact; faults and fracture zones; geographic variation (north to south in the Exploratory Studies Facility); and presence of extreme (high or low) lithophysal cavity densities and welded or nonwelded rock units. This activity will also include confirmation testing of the seepage threshold concept arising from the capillary barrier mechanism.

In Situ Waste Package Monitoring—Remote inspection technology will be employed to conduct inspections of the waste packages in the emplacement drifts. In particular, remotely operated inspection gantries that can conduct remote visual, thermal, and radiological inspection of the waste packages, as well as collect and place material samples or coupons, will be used. These vehicles will be remotely operated from a control center and will periodically inspect each emplacement drift and all accessible waste packages. They will be able to inspect the upper and lower surfaces of the waste packages and will operate within the high-temperature and high-radiation environment of the emplacement drifts. A conceptual view of a remotely operated inspection gantry is shown in Figure 4-220.

Long-Term Materials Testing—Long-term laboratory studies of the waste form, waste package, and drip shield materials have been and continue to be conducted to obtain data on various degradation phenomena. Laboratory testing employs a range of environmental conditions that bound and simulate repository conditions at future times. Key parameters affecting drip shield and waste package container performance have been and will continue to be investigated, including parameters associated with oxidation and aqueous corrosion.

Ventilation Monitoring—The emplacement drift environment will also be monitored by sampling the ventilation air that goes into and comes out of each emplacement drift. Conceptually, monitoring instruments can be installed within the air regulator at the isolation door and within the exhaust mains or raises. Some parameters that can be measured as part of ventilation monitoring include wet and dry



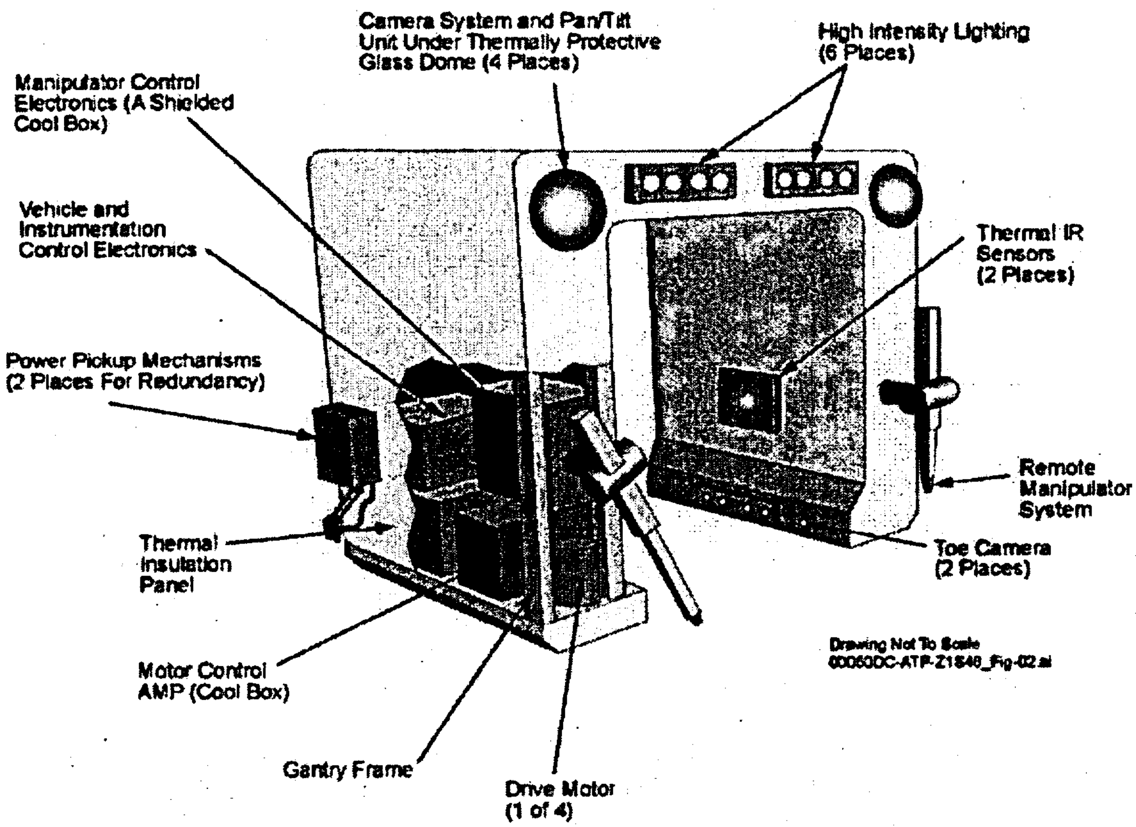


Figure 4-220. Conceptual Illustration of a Performance Confirmation Inspection Gantry
The inspection gantry will have onboard monitoring equipment capable of inspecting waste packages, emplacement drift structures, and environmental conditions. Source: CRWMS M&O 2000ag, Figure 5-1.

bulb temperatures; relative humidity; radioactive gas content (e.g., tritium, krypton-85, and radon); air pressure; percentage of oxygen and carbon dioxide; and dust content.

Rock Mass Monitoring—The coupled thermal-hydrologic-mechanical response of the rock mass around emplacement drifts will be monitored through the performance confirmation period to confirm the conceptual understandings and numerical simulations of coupled processes considered in performance assessments. This monitoring will be conducted at appropriate locations or areas to provide sufficient coverage of the emplacement horizon. Conceptually, this could entail measurement at three locations across the upper block of the potential repository horizon (i.e., at the northern, central, and southern reaches of the horizon). Each test area will monitor approximately 150 m (500 ft) of emplacement drift.

In-Drift Monitoring—To continually monitor the conditions within emplacement drifts, a limited number of in-drift instrument areas will be installed in the drifts. These instruments will provide continual readings at specific locations, at times between the monitoring of the drifts by the inspection gantry, and as a supplement to the indirect monitoring of the drifts (e.g., through ventilation monitoring). The sensors would be similar to those used for the inspection gantry. Several stations would be located along the emplacement drift axis of each test area to provide an estimate of the change in values with distance along the drift. Conceptually, to provide access to these instruments, boreholes from adjacent observation drifts or cross-block drifts could be extended into the emplacement drifts.

Introduced Materials Monitoring—To monitor and analyze changes from the baseline condition of parameters that could affect the performance of a geologic repository, it will be necessary to monitor fluids and other materials introduced into the repository horizon as a result of construction and operations. This monitoring will be done to evaluate the impact that introduced materials (e.g., water from construction activities, fire suppression and ground support materials, hydrocarbons, concrete, steel, and railcars) could have on the postclosure performance of the repository if they remained after closure.

Recovered Material Coupon Testing—This testing involves the placing of nonradioactive waste package material specimens at different locations in the emplacement drifts to expose them to different environmental conditions. After a defined period, the inspection gantry will retrieve the specimens for laboratory examination at surface or offsite facilities. Specimens will be placed in a variety of exposure locations that will cover a reasonable range of the geological and geochemical variations expected to occur in the repository. Testing with some specimens, including welds, will be focused on container and cladding materials.

Dummy Waste Package Testing—This activity entails the construction of dummy waste packages that have the same external materials, dimensions, and configuration as a real waste package, but do not contain any radioactive waste. Instead of waste, each dummy package will house an electrical heater and will be used in test drifts under simulated postclosure conditions.

Recovered Waste Package Testing—If a waste package must be recovered for remediation, activities will be defined on a contingency basis to examine and test it for any potential surface or weld degradation. However, an active program of waste package recovery and inspection for performance confirmation purposes alone is not proposed in the program.

Postclosure Simulation Testing—Postclosure simulation testing will confirm whether the measured conditions within a simulated postclosure drift are within ranges consistent with those assumed in the license application. The testing would be conducted after the start of waste emplacement and would allow for the use of actual waste in addition to dummy waste packages equipped with heaters. A single test drift will be separated into test sections, allowing for the simulation of several different test cases within the drift. The postclosure configuration will be constructed in the section with either dummy or actual waste packages, a drip shield, and backfill (if employed) and will employ expected postclosure technology and equipment. The test sections will be monitored for decades, then deconstructed to evaluate barrier response.

4.6.1.1.2 Development and Licensing Testing

Performance confirmation activities under this test category include baseline testing through completion of subsurface construction. Preemplacement testing to support the license application submittal, licensing interactions, and preemplacement licensing conditions, will also be performed under this category.

The performance confirmation testing and monitoring activities under development and licensing testing are described in the paragraphs below.

Geologic Observations and Mapping—Geologic mapping of repository excavations will be conducted during construction to provide information to confirm and document the geologic structure (stratigraphy) of the emplacement horizon and the characteristics of major fracture sets and faults. This mapping will be conducted in accordance with applicable standard engineering practices and procedures. Conceptually, mapping will be performed for all excavations using a combined approach of stereoscopic imaging and digital mapping to obtain a digital record of the excavation surface. This recording will be supplemented with localized, detailed, full-periphery geologic mapping and the collection of discontinuity statistics in selected areas.

Subsurface Sampling and Index Testing—A sampling and laboratory index test program will be implemented to support seepage testing, together with thermal testing and monitoring activities. Rock samples will be collected at sites corresponding to testing and instrumentation locations, including alcoves and observation drifts. Parameters to be tested in the laboratory will include rock chemistry, hydrologic properties, and mechanical and thermal properties. The data will be incorporated into the performance confirmation baseline and performance predictions and are also expected to provide a basis for evaluating the areal variability of properties across the repository horizon.

Baseline Analyses and Evaluations—The activities for baseline development include developing the performance confirmation database and predicting performance for all key parameters, establishing tolerances and bounds for the parameters, and identifying completion criteria and guidelines for corrective actions.

Unsaturated Zone Testing—Based on a preliminary assessment of data needs, additional testing may be conducted to obtain data on the effects of construction on ambient moisture and seepage, as well as on the sorptive and the unsaturated flow and transport properties of the Calico Hills nonwelded hydrogeologic unit.

Near-Field Environment Testing—Based on a preliminary assessment of data needs, additional testing may be conducted to obtain both data on the coupled thermal-mechanical-hydrologic-chemical response of the rock mass due to cooling and additional field and laboratory testing data from ongoing heater tests to investigate coupled processes necessary to confirm the near-field environment.

Waste Form Testing—Based on a preliminary assessment of data needs for the waste form, additional testing may be conducted to obtain data on dissolved radionuclide concentrations, colloidal concentration and transportation limits of waste form materials, and cladding performance data over the range of expected in situ conditions. Laboratory testing has been identified to obtain additional data on in-package chemistry and its effects on cladding performance and on colloidal and dissolved radionuclide concentrations.

Waste Package Testing—Based on a preliminary assessment of data needs for the waste package, additional testing may be conducted to obtain data on barrier materials (such as Alloy 22, Titanium Grade 7, and Stainless Steel Type 316NG), stress corrosion cracking, material phase stability, and the phase stability of passive films that can form on the surface of waste package materials.

4.6.1.1.3 Prototype Testing

Performance confirmation testing activities will also be conducted to examine the performance of full-scale prototypes to demonstrate constructibility and the effectiveness of systems considered important to safety. These activities are described in the following paragraphs.

Borehole Seal Testing—Prototype testing of borehole seals will be performed in surface boreholes, using appropriate drilling technology to install and test the seals. This testing will involve seals emplaced in shallow boreholes to allow for seal recovery and will include hydraulic conductivity and nondestructive testing of the seal. After testing, the seals will be removed from the site for additional materials testing in the laboratory.

Ramp and Shaft Seal Testing—Prototype ramp and shaft seal tests will be conducted to examine seal performance and to resolve design issues associated with large-scale construction not previously addressed by laboratory or small-scale in situ testing. These tests will be performed within representative geologic units and will use construction techniques similar to those identified for repository closure.

4.6.1.1.4 Technical Specifications and Monitoring

Performance confirmation testing, and other testing and evaluation activities, will be conducted within the Geologic Repository Operations Area and in the immediate region around the repository, as necessary. Specific environmental testing and monitoring activities related to performance confirmation are described in the following paragraphs.

Groundwater Quality Monitoring—Active monitoring of the uppermost aquifer will be performed using a series of both upgradient and downgradient wells. The wells will be installed and periodically sampled to evaluate the chemistry and radioactivity of groundwater adjacent to the repository. Conceptually, one upgradient and four downgradient wells will be employed for performance confirmation.

Groundwater Level and Temperature Monitoring—In coordination with groundwater quality measurements, the in situ temperature and the elevation of the groundwater in wells will be measured and compared to prior measurements to determine whether there have been hydrogeologic or other changes that could modify groundwater flow patterns.

Surface Uplift Monitoring—Using periodic, precise measurements, uplift monitoring will be conducted for the elevation grid of reference points on the surface above the repository horizon. Elevation changes over time will be used to determine whether significant local surface movement is occurring above the repository.

Subsurface Seismic Monitoring—Subsurface seismic monitoring will be conducted to measure the occurrence and magnitude of seismic events at repository depth. Measurements will be compared to the seismic design bases to confirm that a sufficient safety margin is being maintained.

4.6.2 Safeguards and Security

A repository at Yucca Mountain would be the first permanent geologic repository for the disposal of spent nuclear fuel and high-level radioactive waste, including immobilized plutonium encased in high-level radioactive waste. The DOE will implement appropriate safeguards, security, and reporting measures consistent with 10 CFR 63.102(k) and 10 CFR 63.21(b)(3) (66 FR 55732).

The DOE will establish a system for verifying, tracking, and mapping each item of waste—both civilian and defense—that is accepted, transported, and eventually emplaced in the repository.

After closure, the DOE will also have the responsibility of maintaining institutional control over the repository. The DOE will maintain appropriate institutional controls consistent with plans to be developed to support a license application if the site is designated.

5. DESCRIPTION OF THE PRECLOSURE SAFETY ASSESSMENT

This section describes the analytical methods and summarizes the results of the preclosure safety assessment for a potential repository at Yucca Mountain. Section 5.1 describes how facilities and systems for the potential repository would use established commercial technologies and nuclear industry technologies to reduce the risk of Category 1 and Category 2 event sequences, since these technologies are well understood. Section 5.2 describes the approach used in assessing the preclosure operational safety of a potential repository at Yucca Mountain. It also discusses event identification, event sequence categorization, event sequence consequence analysis, use of features and controls important to radiological safety, and quality assurance classification. Section 5.3 provides a description of events and the results of consequence analyses and evaluations. Section 5.4 describes the testing and evaluation program planned for the potential repository's preclosure period.

5.1 KNOWN TECHNOLOGY AND OPERATING SYSTEMS

A repository at Yucca Mountain would use commercial and nuclear industry technologies for preclosure construction and operations. The methods these technologies use to reduce the risk of event sequences are well understood.

Over the past 50 years, large nuclear facilities have been designed, constructed, and operated by the commercial nuclear industry and the U.S. government. Incorporated into the design of these facilities are features and controls that prevent or reduce the consequences of accidents. The repository design draws upon this extensive experience and is based on proven technology in use at nuclear installations worldwide. For example, high-efficiency particulate air filters have been used for many years to reduce atmospheric emissions from nuclear facilities. Monitoring systems have also been used for many years to measure atmospheric effluents. Computer codes to estimate exposure from effluents have been developed and are widely used. The principles of radiation shielding are well known, and computer codes are available to aid in shielding design. The principles of time, distance, and shielding are used to keep radiation doses as low as is reasonably achievable (ALARA) (e.g., *Health Physics Manual of Good Practices for Reducing Radiation Exposure to Levels that are As Low As is Reasonably Achievable* [Munson et al. 1988]).

Spent nuclear fuel transportation casks are routinely loaded and unloaded in the United States. Heavy loads are routinely moved by bridge cranes at nuclear facilities, as they would be at a repository at Yucca Mountain. Across the United States, commercial nuclear power reactors currently operate spent nuclear fuel pools. At all operating nuclear plants, handling spent nuclear fuel is a routine activity. For example, from 1968 to 1994, about 105,000 spent nuclear fuel assemblies were discharged from commercial nuclear power reactors (DOE 1996b, Table 5). The lessons learned from these experiences would be incorporated into the design and concept of operations for any repository.

5.2 BASIC SAFETY ASSESSMENT METHOD

The two basic elements of any safety assessment are event identification and consequence analysis. The first element involves performing a systematic review of relevant site and facility features and processes in order to define the types of events that can occur. Events identified include the full range of probable events, from normal operational events that might occur to very low-probability events. Events are identified by first evaluating potential hazards applicable to the site and facility design, then developing a detailed site- and design-specific event scenario in which event sequences are defined and the anticipated frequency of occurrence of events is established. Based on the frequency of occurrence, events are categorized as Category 1 or Category 2 event sequences. Event sequences with lower frequencies of occurrence are considered beyond Category 1 and Category 2 event sequences and were not analyzed further. The second element of the safety assessment involves estimating the consequences of the event sequences that are identified as a Category 1 or Category 2 event sequences in the first process.

The safety assessment performs an important role in the design process. It plays a key role in the identification of facility design features and controls important to safety and is a primary input to the quality assurance classification process. In some cases, alternative design approaches or additional design features may be identified based on safety assessment results, which are then considered as part of an iterative design process. Based on the insights and results obtained from the safety assessment, the acceptability of the design can be established.

5.2.1 Event Identification Process

Events are identified based on a review of repository site characteristics, facility design features, and operational processes to be performed. An analysis of the internal and external hazards associated with preclosure operations is performed. Internal hazards are presented by the operation of the facility and associated processes. External hazards involve natural phenomena and outside man-made hazards, such as those posed by aircraft and nearby government or industrial facilities. The methodology used in the event identification analysis provides a systematic means to identify facility hazards and associated events that may result in radiological consequences to the public and workers during the repository preclosure period.

The first step in the hazard identification process is to develop a list of generic internal and external events that could result in radiological consequences to the public or workers. This generic list is not facility-specific and attempts to identify potentially hazardous events by providing a comprehensive list of possible events. The generic lists developed for the internal and external hazard analyses are based on established hazard evaluation techniques (Stephans and Talso 1997; American Institute of Chemical Engineers 1992). Tables 5-1 and 5-2 list these generic internal and external events.

Table 5-1. Generic Internal Events

Internal Event
Collision/Crushing
Chemical Contamination/Flooding
Explosion/Implosion
Fire
Radiation/Magnetic/Electrical/Fissile
Thermal

Source: BSC 2001f, Table 5-1.

Once the site characteristics, facility design, and operational processes are defined, they are evaluated against specified criteria to determine the credibility of generic hazard events that could result in radiological consequences. Event applicability criteria are developed for the generic events to support the applicability determination. If the criteria are satisfied, the generic event has the potential for a radiological consequence and is added to a list of specific initiating events to be considered in the design and safety analysis.

The criteria used to determine the applicability of internal hazards as initiators of event sequences are listed below for each event category. Applicability to a functional area of design is determined by a positive response to all questions within a hazard category or subcategory, as appropriate:

A. Collision/Crushing

   1. Is kinetic or potential energy present?

   2. Can the kinetic or potential energy be released in an unplanned way?

   3. Can the release of kinetic or potential energy interact with the waste form?

B. Chemical Contamination/Flooding

   1. Reactions

      a. Are corrosive/reactive chemicals or materials present?

      b. Can these chemicals or materials be released?

      c. Can the chemicals or materials interact with the waste form?

Table 5-2. Generic External Events

External Events		
Aircraft crash	High river stage	Seismic activity, uplifting (tectonic)
Avalanche	Hurricane	Seismic activity, earthquake
Coastal erosion	Inadvertent future intrusions (man-made)	Seismic activity, surface fault displacement
Dam failure	Industrial activity-induced accident	Seismic activity, subsurface fault displacement
Debris avalanching	Intentional future intrusions (man-made)	Static fracturing
Denudation	Landslides	Stream erosion
Dissolution	Lightning	Subsidence
Epeirogenic displacement	Loss of offsite/onsite power	Tornado
Erosion	Low lake level	Tsunami
Extreme wind	Low river level	Undetected past intrusions (man-made)
Extreme weather fluctuations	Meteorite impact	Undetected geologic features
Range fire	Military activity-induced accident	Undetected geologic processes
Flooding (storm, river diversion)	Orogenic diastrophism	Volcanic eruption
Fungus, bacteria, and algae	Pipeline accident	Volcanism, intrusive magmatic activity
Glacial erosion	Rainstorm	Volcanism, ashflow (extrusive magmatic activity)
Glaciation	Sandstorm	Volcanism, ashfall
High lake level	Sedimentation	Waves (aquatic)
High tide	Seiche	

Source: BSC 2001f, Table 5-2.

2. Off-Gassing
   a. Are volatile/condensable materials present?
   b. Can these materials be released?
   c. Can these materials interact with the waste form?
3. Venting
   a. Is there a potential for venting materials in the area?
   b. Can the materials interact with the waste form?
4. Debris/Leaks
   a. Is there a potential for debris or leaks in the area?
   b. Can the debris or fluids interact with the waste form?
5. Flooding
   a. Are sources of water present in the area?
   b. Is there a potential to release the water?
   c. Can the released water interact with the waste form with the potential for criticality?

C. Explosion/Implosion

1. Are pressure and electrical, chemical, or mechanical energy present?
2. Can an event occur that results in an explosion or implosion energy release?
3. Can the released energy impact the waste form directly?

D. Fire

1. Are fuel, oxidizers, and ignition sources present?

2. Is there sufficient fuel and oxidizer to sustain fire?

3. Can fire interact with the waste form?

E. Radiation/Magnetic/Electrical/Fissile

1. Are radiation/magnetic/electrical energy sources present external to the waste form? Is fissile material present?

2. Is a mechanism present to release radioactive/magnetic/electrical energy?

3. Can the release of radiation/magnetic/electrical energy interact with the waste form? Can fissile material be arranged, through operational processes, in a way that will result in criticality?

F. Thermal

1. Are external heat energy sources present?

2. Can heat energy be released?

3. Can the heat energy affect the waste form?

The criteria used to determine the applicability of external events as initiators of event sequences are listed below. The external event is considered a potential initiator of an event sequence if all of the following are determined to be true:

A. The potential exists and is applicable to the Yucca Mountain site.

B. The rate of the process is sufficient to affect the 100-year operational period. (Example: Is erosion expected to occur at the repository during the 100-year operational phase?)

C. The consequence of the process is significant enough to affect the 100-year operational period. (Example: Can the consequences of erosion lead to a radiological release at the repository during the 100-year operational phase?)

D. The event frequency is greater than or equal to 0.000001 events per year. (Example: Is any event associated with erosion expected to occur at a rate greater than or equal to once in a million years?)

If all the above statements are true for any external event, then the event is considered applicable. If any one of the above statements is false for any external event, the event is not considered applicable. If any statement is indeterminate (i.e., its validity cannot be determined at this time), the statement is treated as true and cannot be screened out at this point.

To evaluate the design and operations for the preclosure period, the period to be evaluated must be defined. The process described above used a 100-year operational phase for the higher-temperature operating mode, but the same process is valid for lower-temperature operating modes that have longer preclosure operational phases. Depending on the thermal operating mode, the preclosure period could be longer (see Section 2.1.5.2, Table 2-2) (BSC 2001f). An operational period of 100 years was selected as the duration to be used in the evaluation since it bounds the emplacement period for the range of thermal operating modes. The handling of waste in the surface and subsurface facilities is expected to last a minimum of 24 years (see Section 2.3.4.5). A 100-year preclosure period bounds surface and subsurface facilities operations and is conservative for classifying events as Category 1 and Category 2 event sequences. For example, using a 24-year period would result in a Category 1 cutoff at 4.2×10^{-2} per year and potentially allow more event sequences to be compared with the less restrictive Category 2 dose limits. After the operational phase, when the waste has been emplaced in the subsurface facility, the potential for internal and external events is still possible. Assuming a preclosure period of 325 years would lower the Category 1 cutoff to 3.1×10^{-3} per year and lower the Category 2 cutoff to 3.1×10^{-7} per year. However, it was determined that with a 325-year preclosure period, no new events would be included (BSC 2001f, Section 4.4.1.2.1). Note that *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f) describes Category 1 and Category 2 design basis events. There is no difference between Category 1 and Category 2 design basis events and Category 1 and Category 2 event sequences.

5.2.2 Event Sequence Categorization Process

The result of the event sequence identification process is a list of event sequences with a corresponding frequency of occurrence. The frequency of occurrence for each event sequence is determined using fault tree analysis or data from historical events. The frequency of occurrence is usually expressed in terms of the chance of the particular event sequence occurring during facility operations, for example, "3 chances in 100 of occurring before permanent closure of the repository." In this example, if the repository operates for 100 years and the event sequence frequency is uniform over the entire period, it can be expressed as 0.0003 per year or 3.0×10^{-4} per year. Initially, when postulating the event sequence, no credit is given to design features that could prevent or mitigate the event (i.e., the most severe consequences are evaluated). If the radiation dose consequences of an event sequence are unacceptable, design features are added to prevent or mitigate the event.

Based on frequency of occurrence, event sequences are categorized as a Category 1 or Category 2 event sequence or beyond Category 1 and Category 2 event sequences, consistent with 10 CFR 63.2 (66 FR 55732). Category 1 event sequences are expected to occur one or more times before permanent closure. This is about equal to an annual frequency of one chance in one hundred (0.01 per year)[1], based on a 100-year preclosure operational period (BSC 2001f, Section 4.4.1.2.1). Category 2 event sequences are other event sequences that have at least 1 chance in 10,000 of occurring before permanent closure. This is about equal to an annual frequency of one chance in one million (0.000001 per year), based on a 100-year preclosure operational period (BSC 2001f, Section 4.4.1.2.1). Event sequences that have less than 1 chance in 10,000 of occurring before permanent closure of the repository are considered beyond Category 1 and Category 2 event sequences. The consequences of some of these types of events are presented in *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE 2002, Tables H-6 and H-7).

Event sequences are developed using event trees, which are diagrams that depict the chronological sequence of events. Figure 5-1 shows an example of a typical event tree used to define event sequences and quantify frequency of occurrence. In this example, Event Sequence 1 begins with an unsealed disposal container drop as the initiating event. The second event represents a breach of the spent nuclear fuel assembly. The last event in this branch represents a fully functional ventilation system and associated high-efficiency particulate air filtration. This event sequence has a frequency of 0.0084 per year, classifying it as a Category 2 event sequence. Event Sequence 2 represents a release scenario in which the ventilation system and the high-efficiency particulate air filtration system are nonfunctional. This event sequence has a frequency of 1.4×10^{-9} per year, which is considered to be a beyond Category 1 and Category 2 event sequence; this would be the case even if the probability of the high-efficiency particulate air filtration system not functioning were increased a hundredfold. Event Sequence 3, with zero probability, represents an unsealed disposal container drop that does not breach the enclosed spent nuclear fuel assemblies and does not result in a release.

As illustrated in Figure 5-1, the scenario development process involves the analysis of all facility features or controls that can affect the progression of an event sequence, including the effects of successful operation or failure of the heating, ventilation, and air conditioning systems with

[1] All event sequence frequencies are assumed to be constant over the 100-year preclosure operating period.

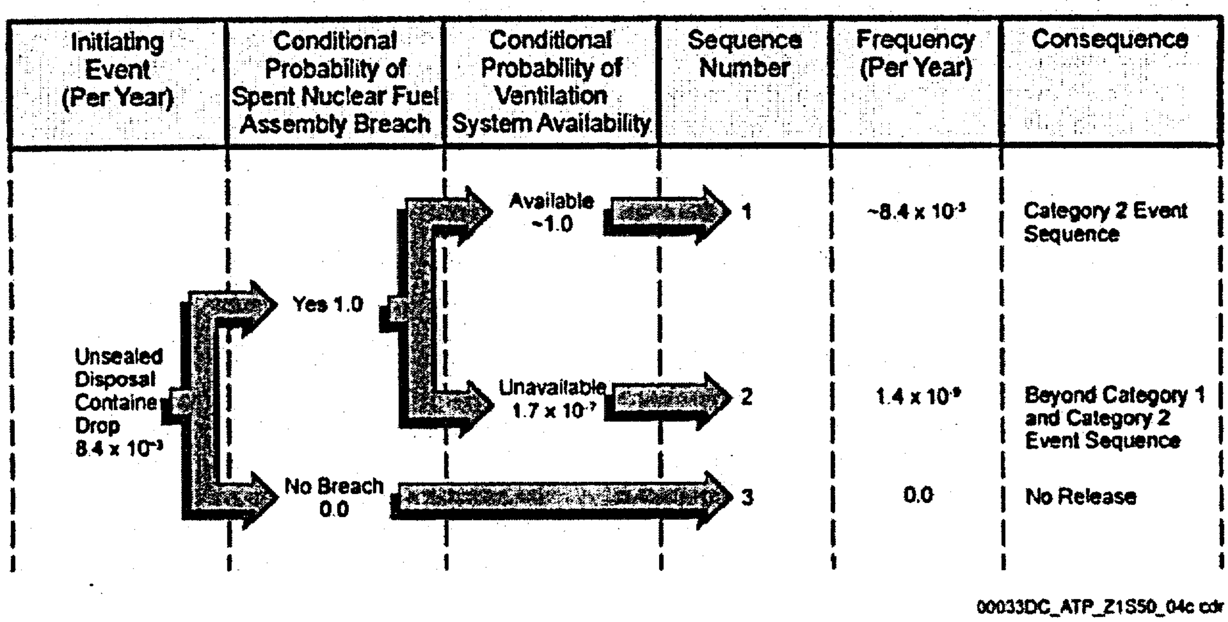


Figure 5-1. Sample Event Tree
An event tree depicts the progression of an event sequence, starting at an initiating event and including successful operation or failure of facility features or controls. In this example, the initiating event is an unsealed disposal container drop and the facility feature or control is the ventilation system. The probability of breaching the spent nuclear fuel assembly in this example, given an unsealed disposal container drop, is 100 percent.

high-efficiency particulate air filters, where appropriate. Insights gained from evaluating the frequency and consequences of such failure sequences are especially useful as inputs to the design and quality assurance classification processes.

5.2.3 Event Sequence Consequence Analysis Process

Category 1 Event Sequences—Three sources are expected to contribute to the annual radiation dose to the public or repository workers from Category 1 event sequences during the facility's preclosure operational lifetime: (1) operational effluents from the Waste Handling Building, (2) operational effluents from the subsurface areas of the repository, and (3) event sequences anticipated to occur at a frequency of 0.01 per year or higher. Section 5.3.5.4.1 in *Preliminary Preclosure Safety Assessment for the Monitored Geologic Repository Site Recommendation* (BSC 2001f) describes the models used to estimate the radiation doses from Category 1 event sequences. Appendix A of *Preliminary Preclosure Safety Assessment for the Monitored Geologic Repository Site Recommendation* (BSC 2001f) considers the influence of flexible thermal operating modes with preclosure periods of up to 325 years on Category 1 event sequence selection.

Category 2 Event Sequences—The radiation doses from Category 2 event sequences come from event sequences anticipated to occur with frequencies between 0.01 and 0.000001 per year. This frequency range assumes a 100-year preclosure period that is associated with the higher-temperature repository operating mode. The Category 2 event sequences all involve drops or collisions while handling fuel assemblies, disposal containers, and transportation casks. Section 5.3.5.4.2 in *Preliminary Preclosure Safety Assessment for the Monitored Geologic Repository Site Recommendation* (BSC 2001f) describes the models used to estimate the radiation doses from Category 2 event sequences. The influence on the selection of Category 2 event sequences of the flexible thermal operating modes with preclosure periods of up to 325 years is discussed in Appendix A of *Preliminary Preclosure Safety Assessment for the Monitored Geologic Repository Site Recommendation* (BSC 2001f).

Several dosimetric quantities were calculated for Category 1 and Category 2 event sequences: (1) the total effective dose equivalent; (2) the radiation dose for various organs and tissues, such as the thyroid, lungs, and bone marrow; and (3) the radiation dose for the skin. Consistent with *Standard Review Plan for Spent Fuel Dry Storage Facilities* (NRC 2000a, Section 9.5.2.2), the sum of the skin dose equivalent and the total effective dose equivalent was used to indicate the lens of the eye dose.

5.2.4 Use of Features and Controls Important to Radiological Safety

The repository design incorporates a combination of prevention and mitigation features and operational controls. Prevention is the use of design features to reduce the frequency of events that result in radiological release. Mitigation involves the use of design features to reduce the consequences of a postulated radiological release event sequences, and includes those features intended to reduce releases from routine operations that are included in the Category 1 event sequences annual dose summation. The safety assessment is used to identify preventive and mitigative features.

The repository design emphasizes prevention features because prevention provides design and operational benefits. From an operations perspective, surveillance and maintenance of active safety features have been demonstrated to add to the operational complexity of existing nuclear facilities. Prevention features are incorporated in the design by performing the safety assessment as an integral part of the design process in a manner consistent with a performance-based, risk-informed philosophy. A risk-informed approach uses risk insights, engineering analysis and judgment, and equipment performance history to focus attention on the most important facility activities and to establish design criteria and management controls based upon these risk insights. This approach ensures that design features and operational controls important to radiological safety are selected in a manner that ensures safety while minimizing operational complexity through the use of proven technology.

The repository would be designed, constructed, and operated to withstand external events and natural phenomena for Category 1 and Category 2 event sequences. For example, Section 2.2.4.2.2 of this report discusses requirements for designing the surface facilities to withstand the vibratory motion associated with earthquakes. As an example, in the assembly transfer system and canister transfer system, overhead cranes and assembly transfer machines would be designed so that they would not become dislodged from their rails during a Category 1 or Category 2 event sequence earthquake. Section 2.2.5 also discusses the design processes used to keep radiation doses to workers ALARA.

For accidents involving internal events, the analysis in *Design Basis Event Frequency and Dose Calculation for Site Recommendation* (BSC 2001u, Table 9) shows that drops of a spent nuclear fuel assembly or canister were important contributors to event sequences. To prevent these types of accidents, the assembly transfer system would be designed, constructed, and operated so that the probability of the dry assembly transfer machine dropping an assembly is low (CRWMS M&O 2000v, Section 1.2.2.1.1). In addition, to reduce the probability that the assembly or canister would be breached because of a drop, the lift heights for fuel assemblies and canisters would be limited, as is standard practice in nuclear facility design and operations.

The analyses in *Design Basis Event Frequency and Dose Calculation for Site Recommendation* (BSC 2001u, Section 5.2.5) show that the availability of the Waste Handling Building heating, ventilation, and air conditioning system with high-efficiency particulate air filters plays a large role in mitigating the consequences of accidents. Therefore, the ventilation system would be designed to be highly reliable. For example, it would be designed to withstand earthquakes, impacts from flying debris (referred to as missiles); fires, or loss of offsite electrical power and still perform its intended safety functions.

The key prevention and mitigation methods rely on the use of:

- Designs that accommodate potential natural phenomena (e.g., fault avoidance, placement, layout, design basis to withstand seismic events, backup power)
- Designs that incorporate safety features for normal operations and Category 1 and Category 2 event sequences (e.g., limits of lift heights, air filtration and confinement systems, redundant systems, limit switches)
- Administration controls (e.g., trained and certified personnel, approved procedures).

5.2.5 Quality Assurance Classification Process

The safety assessment provides valuable input to the quality assurance classification process. Repository features credited as event prevention or mitigation features in the safety assessment are "important to safety," and the safety assessment is useful in determining an item's functional role as part of the repository preclosure safety strategy. Classification is performed in a separate analysis, in accordance with formal quality assurance classification procedures. Structures, systems, and components important to safety are classified in a graded fashion to ensure quality assurance controls are implemented over the facility life cycle commensurate with an item's importance to safety.

The classification process consists of establishing the configuration and function of structures, systems, and components and their effect on repository radiological safety. It is limited to structures, systems, and components procured as a part of the repository system (e.g., transportation casks are not included). This information is then evaluated against criteria provided in the classification procedure to determine the quality assurance classification of the particular item. The following classification categories are specified by Section 3.1.3 of QAP-2-3, *Classification of Permanent Items*, consistent with Section 2 of *Quality Assurance Requirements and Description* (DOE 2000a).

Quality Level (QL)-1—Structures, systems, and components whose failure could directly result in a condition adversely affecting public safety are classified as QL-1. These items have a high safety or waste isolation significance.

QL-1 structures, systems, and components include those items, which:

- Maintain containment and criticality control for spent nuclear fuel and high-level radioactive waste
- Prevent or mitigate a Category 1 or Category 2 event sequence.

QL-1 structures, systems, and components are listed in Table 5-3 with a brief summary of their functions that are important to safety.

QL-2—Structures, systems, and components whose failure or malfunction could indirectly result in a condition adversely affecting public safety, or whose failure would result in doses in excess of normal operational limits, are classified as QL-2. These items have a lower safety or waste isolation significance.

QL-2 structures, systems, and components include those items, which:

- Provide control and management of site-generated radioactive waste
- Provide fire protection/suppression to protect the function of a QL-1 structure, system, or component important to safety
- Maintain structure, system, or component integrity so that a QL-1 structure, system, or component is not prevented from performing its intended function if an event sequence occurs
- Prevent or mitigate a Category 1 event sequence

Table 5-3. QL-1 Structures, Systems, and Components

Structure, System, or Component	Monitored Geologic Repository System	Function Important to Safety
Assembly transfer baskets	Assembly transfer system	Provide criticality control for spent nuclear fuel assemblies
Basket staging racks	Assembly transfer system	Provide criticality control for spent nuclear fuel assemblies
Control and tracking system	Waste emplacement/ retrieval system	Provide operational information to the operations, monitoring, and control system; minimize the likelihood of uncontrolled descent of the waste package transporter and the possible impact of a waste package with the subsurface facility structure or other facility equipment resulting in radiological release
Disposal containers	Waste package designs	Provide containment and criticality control for waste
Drip shield	Emplacement drift system	Provide containment, waste package protection, and heat transfer
Locomotives	Waste emplacement/ retrieval system	Minimize the likelihood of uncontrolled descent of the waste package transporter and the possible impact of a waste package with the subsurface facility structure or other facility equipment, resulting in a radiological release
Modified waste package transporter	Waste emplacement/ retrieval system	Minimize the likelihood of uncontrolled descent of the waste package transporter and the possible impact of a waste package with the subsurface facility structure or other facility equipment resulting in radiological release
Small canister staging racks	Canister transfer system	Provide criticality control for DOE high-level waste canisters
Waste package transporter	Waste emplacement/ retrieval system	Minimize the likelihood of uncontrolled descent of the waste package transporter and the possible impact of a waste package with subsurface facility structure or other facility equipment, resulting in a radiological release
Waste Handling Building structure	Waste Handling Building system	Provide containment of radioactive materials, radiation shielding, and protection of equipment from internal and external hazards

Source: BSC 2001f, Table 4-1.

- In conjunction with an additional item or administrative control, prevent or mitigate a Category 2 event sequence.

QL-2 structures, systems, and components are listed in Table 5-4 with a brief summary of their functions that are important to safety.

QL-3—Structures, systems, and components whose failure or malfunction would not significantly impact public or worker safety, including those defense-in-depth design features intended to keep radiation doses ALARA, are classified as QL-3. These items have a minor impact on public and worker safety and on waste isolation.

QL-3 structures, systems, and components include those items, which:

- Warn of significant increases in radiation levels or concentrations of radioactive materials
- Monitor variables to verify that operating conditions are within technical specification limits
- Support repository emergency response actions
- Assess radionuclide release or dispersion following an event sequence
- Maintain levels of radioactive effluents
- Limit worker doses from normal operations and Category 1 event sequences.

Examples of structures, systems, and components classified as QL-3 include the meteorological monitoring system, area radiation monitoring system, and exhaust stack radiation monitors (BSC 2001f, Table 4-3).

Table 5-4. QL-2 Structures, Systems, and Components

Structure, System, or Component	Monitored Geologic Repository System	Function Important to Safety
Assembly drying system	Assembly transfer system	Collect and manage site-generated radioactive waste produced in the assembly drying process
Backfill emplacement system	Backfill emplacement system	Maintain structural integrity
Bridge cranes	Assembly transfer system Carrier/cask handling system Canister transfer system Disposal container handling system Waste package remediation system	Maintain structural integrity Prevent interactions with QL-1 structures, systems, and components
Control and tracking system	Assembly transfer system	Minimize the likelihood of drop of assembly transfer basket during transfer of spent nuclear fuel assemblies
Control and tracking system	Carrier/cask handling system	Provide operations support necessary for waste handling safety by controlling crane movement during handling of transportation casks
Control and tracking system	Canister transfer system Disposal container handling system Waste package remediation system	Support site-generated radiological waste collection and management functions
Cooling system	Assembly transfer system	Collect and manage the site-generated radioactive waste generated in the spent nuclear fuel container cooling process
Covered shuttlecars	Waste emplacement/retrieval system	Provide for radioactive particulate confinement
Disposal container inerting system	Disposal container handling system	Collect and manage the site-generated radioactive waste generated in the disposal container inerting process
Disposal container loading port mating device	Assembly transfer system	Maintain structural integrity
Disposal container weld station jib crane	Disposal container handling system	Maintain structural integrity
Decontamination systems	Assembly transfer system Canister transfer system Disposal container handling system Waste Handling Building system Waste package remediation system Waste emplacement/retrieval system	Collect and manage the site-generated radioactive waste generated in the process of facility and equipment decontamination
Dry assembly transfer machine	Assembly transfer system	Maintain structural integrity Prevent drop of assembly transfer basket during transfer of spent nuclear fuel assemblies
Dual-purpose canister lid severing tool	Assembly transfer system	Collect and manage radiologically contaminated metal chips generated during dual-purpose canister lid removal operations
Emergency power source and distribution system	Waste Handling Building electrical system	Support the Waste Handling Building primary ventilation system in mitigating the consequences of an event sequence
Emplacement drift ground control	Ground control system	Minimize the likelihood of breach of waste package in emplacement drift due to rockfall
Fire detection systems	Waste Handling Building fire protection system	Protect QL-1 structures, systems, and components from the effects of fire
Fire suppression systems	Waste Handling Building fire protection system	Protect QL-1 structures, systems, and components from the effects of fire
Invert	Emplacement drift system	Provide support for mobile equipment in the drifts and for the drip shield and waste package/pallet combination
Lifting fixtures, cask and dual-purpose canister preparation system Lifting fixtures, disposal container handling system	Assembly transfer system Disposal container handling system	Maintain structural integrity

Table 5-4. QL-2 Structures, Systems, and Components (Continued)

Structure, System, or Component	Monitored Geologic Repository System	Function Important to Safety
Lifting fixtures, dry assembly handling system	Assembly transfer system	Minimize the likelihood of drop of assembly transfer basket during transfer of spent nuclear fuel assemblies
Liquid low-level waste system	Site-generated radiological waste handling system	Collect and manage site-generated radioactive waste generated in the operation of monitored geologic repository facilities
Mixed low-level waste system	Site-generated radiological waste handling system	Collect and manage site-generated mixed waste generated in the operation of monitored geologic repository facilities
Monitored geologic repository operations monitoring and control system	Monitored geologic repository operations monitoring and control system	Mitigate the consequences of an event sequence
Multipurpose hauler	Waste emplacement/retrieval system	Provide for radioactive particulate confinement for breached waste packages
Pool water treatment	Pool water treatment and cooling system	Collect and manage site-generated radioactive waste generated in the process of pool water treatment
Site fire protection system	Site fire protection system	Protect QL-1 structures, systems, and components from the effects of fire
Solid low-level waste system	Site-generated radiological waste handling system	Collect and manage site-generated radioactive waste generated in the operation of monitored geologic repository facilities
Waste Handling Building primary, secondary, and tertiary confinement area ventilation system	Waste Handling Building ventilation system	Mitigate the consequences of an event sequence
Waste package/disposal container weld preparation and opening system	Waste package remediation system	Collect and manage radiologically contaminated metal chips generated during lid removal operations
Waste package horizontal lifting system	Disposal container handling system	Maintain structural integrity
Waste package emplacement pallet	Emplacement drift system	Prevent the waste package from shifting and impacting the drip shield
Waste Treatment Building confinement area ventilation system	Waste Treatment Building ventilation system	Collect and manage site-generated radioactive waste generated in the operation of monitored geologic repository facilities
Waste Treatment Building system	Waste Treatment Building system	Collect and manage site-generated radioactive waste generated in the operation of monitored geologic repository facilities
Wet assembly transfer machine	Assembly transfer system	Maintain structural integrity

Source: BSC 2001f, Table 4-2.

Conventional Quality (CQ)—Those structures, systems, and components not meeting any of the criteria for QL-1, QL-2, or QL-3. Examples of structures, systems, and components classified as CQ include materials for balance-of-plant buildings, utilities, and commercial off-the-shelf materials and equipment.

This classification process is implemented in an iterative fashion, where each analysis iteration is considered for that phase of design. Classifications of repository structures, systems, and components will, therefore, be reevaluated as the design is developed. This approach is consistent with *Technical Position on Items and Activities in the High-Level Waste Geologic Repository Program Subject to Quality Assurance Requirements* (Duncan et al. 1988, Section 4.2(a)), which allows engineering judgment and conservative bounding assumptions to be used in cases where data are limited.

5.3 PRELIMINARY DESCRIPTION OF POTENTIAL HAZARDS, EVENT SEQUENCES, AND CONSEQUENCES

This section presents the preliminary description of potential hazards, event sequences, and consequences of event sequences. Section 5.3.1 identifies the external events and natural phenomena that are the initiating events that could lead to a radiological release. Section 5.3.2 describes internal initiating events, including those that could result in a potential radiological release, no release, or a beyond Category 1 and Category 2 event sequence. Section 5.3.3 presents the consequence evaluations for Category 1 and Category 2 event sequences.

5.3.1 Preliminary Description of External Events

The general strategy for managing external initiating events is to design those structures, systems, and components important to safety to withstand the initiating events so that no release scenarios are initiated and no loss of isolation of radioactive material results. Table 5-5 lists the external events and natural phenomena initiating events considered in this evaluation. The events in Table 5-5 are appropriate for preclosure period of 100 years as well as 325 years (BSC 2001f, Appendix A).

Loss of Offsite Power—This event results in the total loss of external alternating current power to the potential repository for any period of time. It is postulated to occur as a result of an external event (e.g., lightning or downed power line) or an internal event (e.g., fire or random equipment failure). Loss of offsite power would, at a minimum, temporarily halt the transfer of waste. Loss of offsite power at the potential repository is assumed to occur one or more times during preclosure operations; therefore, it is a Category 1 event sequence.

The strategy for this event is to prevent Category 1 or Category 2 release scenarios by providing reliable power through redundant standby power sources (onsite), uninterruptible power, redundant emergency equipment where needed, redundant distribution systems, and mechanical backup controls for components important to safety. Structures, systems, and components important to safety are designed to prevent load drops during a loss of offsite power. Onsite backup power sources with staged loading controls and potential redundant offsite power lines and sources may be used to ensure continuous power is supplied to structures, systems, and components important to safety. The potential repository design would also include such features as external lightning rods to protect against a lightning-initiated loss of offsite power.

Earthquake—Vibratory Ground Motion—An earthquake is the result of sudden relative motions, or slip, between two adjacent rock surfaces in the earth's crust. The sudden slip results in the release of seismic energy, in the form of vibratory ground motion, that propagates from the location of the

Table 5-5. External Initiating Events and Natural Phenomena

Initiating Event/Natural Phenomenon	Location	Frequency (per year)	Initiating Event Frequency Category[a]
Loss of offsite power	Surface and subsurface facilities	<1	1
Earthquake—vibratory ground motion	Surface and subsurface facilities	0.001 0.0001	1 2
Earthquake—fault displacement	Surface and subsurface facilities	0.0001 0.00001	1 2
Probable maximum flood	Surface and subsurface facilities	<<0.01	2
Tornado missiles	Surface facilities	<<0.01	2
Tornado wind	Surface facilities	<<0.01	2

NOTES: [a]For external events, the initiating event (e.g., earthquake) frequency is considered instead of the event sequence.
Source: BSC 2001f, Table 5-4

earthquake to the earth's surface. This ground motion can impact structures, systems, and components in the surface and subsurface facilities and lead to a radiological release. The possible consequences of an earthquake include a collapse of structures, concrete cracking, loss of offsite power, ground displacement, and subsurface rockfall.

The U.S. Department of Energy (DOE) would use proven engineering techniques to design structures to withstand potential earthquakes in the site area. The repository surface facilities, where waste would be received, prepared for emplacement, and moved into the repository, would be subject to stronger earthquake ground shaking than subsurface facilities, where waste would be emplaced.

Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain (YMP 1997, Section 3.1) establishes seismic hazard probability reference values to be used in determining two levels of design basis vibratory ground motion. The two reference values correspond to Category 1 and Category 2 event sequences and are defined as mean annual exceedance probabilities of 10^{-3} and 10^{-4}, respectively. The mean annual probabilities were used in the disaggregation of probabilistic seismic hazard estimates (CRWMS M&O 2000fd, Section 6.5.3) to identify those earthquakes that control the seismic hazard at the reference probabilities.

Ground motion inputs used for preclosure design analyses are described in Section 4.3.2.2.3 (Figure 4-165). These inputs are based on a mean annual exceedance probability of 10^{-4} and were developed for generic locations at the repository elevation (i.e., a depth of 300 m [1,000 ft]) and at a hard-rock outcrop directly above the potential repository.

The safety strategy for the surface facilities is to design the structures, systems, and components important to safety to withstand the effects of a design basis earthquake. The design and construction attributes necessary to ensure that structures and systems are not compromised during a seismic event are well understood and would be applied to the repository facilities.

The following NRC documents related to design basis seismic events were among the sources considered in the repository design process:

- Regulatory Guide 1.29, *Seismic Design Classification*
- Regulatory Guide 1.32, *Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants*
- Regulatory Guide 1.60, *Design Response Spectra for Seismic Design of Nuclear Power Plants*
- Regulatory Guide 1.61, *Damping Values for Seismic Design of Nuclear Power Plants*
- Regulatory Guide 1.92, *Combining Modal Responses and Spatial Components in Seismic Response Analysis*
- Regulatory Guide 1.122, *Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components*
- Regulatory Guide 1.165, *Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion.*

Earthquake—Fault Displacement—A fault is a fracture or zone of weakness in the earth's crust along which there is the potential for relative motion of rocks on opposing sides of the fracture. Scientists have used the data from site characterization studies to assess the potential for fault rupture related to earthquakes.

Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain (YMP 1997) establishes the probabilistic criteria for fault displacement initiating events appropriate for structures, systems, and components important to safety. Specifically, the mean annual exceedance probabilities of 10^{-4} and 10^{-5} are used for Category 1 and Category 2 initiating event fault displacements, respectively. These values are a factor of 10 lower than the exceedance probabilities of the corresponding Category 1 and Category 2 initiating event vibratory ground motion, reflecting the more limited experience with engineering designs for facilities that are subject to fault displacement and with assessments of fault displacement hazard.

An evaluation of the fault displacement hazard at nine locations in the Yucca Mountain vicinity was part of the probabilistic seismic hazard analyses (CRWMS M&O 2000fd, Section 6.6.3). The nine locations span the range of known faulting conditions in the area, which include recognized faults, small fractures, and unfaulted (intact) rock. Results of the hazard assessment indicate that mean displacements on the block-bounding Bow Ridge and Solitario Canyon faults are 7.8 cm (3.1 in.) and 32 cm (12.6 in.), respectively, at the 10^{-5} annual exceedance probability level. In contrast, in areas where waste would be emplaced, displacements of 0.1 cm (0.04 in.) have less than one chance in 100,000 of being exceeded each year during the preclosure period.

Unlike vibratory ground motion hazard, fault displacement hazard is concentrated at the location of faults. Consequently, the exposure of structures, systems, and components to fault displacement hazard can be limited by avoiding locations near faults that have a significant potential for fault displacement. Fault avoidance is the DOE's preferred approach to mitigating fault displacement hazards.

The NRC's *Staff Technical Position on Consideration of Fault Displacement Hazards in Geologic Repository Design* (McConnell and Lee 1994) was considered in the repository design process.

Flood—An external flood may be initiated by intense precipitation, runoff, or a landslide. As defined by Section 2 of ANSI/ANS-2.8-1992, *American National Standard for Determining Design Basis Flooding at Power Reactor Sites*, the probable maximum flood is the hypothetical flood (peak discharge, volume, and hydrograph shape) considered to be the most severe reasonably possible flood, based on a probable maximum precipitation and other hydrologic factors favorable for maximum flood runoff, such as sequential storms and snowmelt. A 100-year flood is defined as the magnitude of peak discharge at any point on a river or drainage channel that can be expected to occur or be exceeded, on average, once in 100 years. Since the Yucca Mountain area is located inland and has no significant surface-water bodies or water-control structures near the site, there is no potential for such events as surges, seiches, tsunamis, dam failures, or ice jams that could affect the site nor is there any potential for future dam development. No evidence for past flooding induced by landslides in the vicinity of the site has been reported. However, floods can produce heavy loads on structures and equipment. The consequences of a flood initiating event are expected to bound the rainstorm, landslide, and debris avalanche events (BSC 2001f, Section 5.2.1.4).

The primary safety strategy for the flood event is to locate facilities outside of flood-prone areas and provide diversion channels to divert runoff away from structures. Taking into account the effects of sediment and debris transported during flood events, a series of worst-case flood studies was completed. The North Portal site is adjacent to the Midway Valley Wash. The maximum depth of water in this wash was estimated to be about 3 to 4 m (9 to 12 ft) during a probable maximum flood, with consideration given to the presence of sediment and debris. Although it was determined that a portion of the North Portal pad is in the flood-prone area, the flood waters would stop at or flow around the boundary of the pad because the pad would be higher than the maximum flood levels. Since water would rise in response to flow restrictions caused by the pad, the Waste Handling Building and Waste Treatment Building would be set approximately 0.5 m (1.5 ft) above the maximum flood elevation. The pad for the balance-of-plant area would be set about 1 m (3 ft) below the floor elevation of the Waste Handling Building to account for its dock height at the southeast corner. The drainage of the Radiologically Controlled Area would protect this pad from a probable maximum flood. An underground storm drainage collection system would contain the runoff from this area and prevent spillage into the balance-of-plant area, protecting the pad from the flood. Two open channels constructed for the Exploratory Studies Facility would protect the

North Portal from the probable maximum flood (BSC 2001f, Section 5.2.1.4).

The Waste Handling Building, Waste Treatment Building, and Carrier Preparation Building are all designed to withstand the probable maximum flood. Other surface facilities are designed to withstand the 100-year flood, based on standard industrial practice (BSC 2001f, Section 5.2.1.4).

For defense in depth, the following additional surface facility characteristics or design features may also be used for flood protection:

- Hardened foundations and structures
- Sandbags, flood doors, and bulkheads
- Waste Handling Building cells located within an interior wall
- Administrative controls during severe weather conditions.

Regulatory Guide 1.59, *Design Basis Floods for Nuclear Power Plants*, was among the sources considered in the repository design process.

Tornado Missiles—This event involves the impact of a tornado-generated missile (flying debris). The tornado initiating event is classified as a Category 2 event sequence (BSC 2001f, Section 5.2.1.5).

The primary safety strategy is to preclude a radiological release by designing structures, systems, and components important to safety that could be vulnerable to a tornado missile to withstand the design basis tornado.

Structures, systems, and components that are vulnerable to tornado missile impacts are either protected from the missiles, designed to withstand a missile impact, or shown to not interact with a missile by a probabilistic analysis. The waste package transporter is designed to prevent any penetration that could breach a waste package as a result of the impact of a tornado missile, the surface facility foundations and structures would be designed to protect the waste forms inside from a tornado missile initiating event, and the Waste Handling Building ventilation system would be designed to continue functioning after a tornado missile initiating event impact.

Sections 3.5.1.4 and 3.5.2 of *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* (NRC 1987) provide NRC guidance on missiles generated by natural phenomena and externally generated missiles, respectively. Additional defense-in-depth safety features may include administrative controls in the event of a tornado warning or extreme weather conditions, hardened buildings, and the installation of underground utilities.

Tornado Wind—This event is associated with the effects produced by high winds during a tornado (i.e., pressure drop and wind loading). The consequences of this event are pressure loads on the surface facilities, waste package transporter, and transportation cask surfaces. The design basis tornado wind is classified as a Category 2 event sequence (BSC 2001f, Section 5.2.1.6).

Structures, systems, and components that are important to safety and potentially vulnerable to a tornado would be designed to withstand the static loading and pressure drops associated with the design basis tornado. This strategy includes designing the Waste Handling Building foundations and structures to withstand the design basis tornado and designing the Waste Handling Building ventilation system to confine and filter particulates following a design basis tornado.

The following NRC documents related to design basis tornadoes were among the sources considered in the repository design process:

- Regulatory Guide 1.76, *Design Basis Tornado for Nuclear Power Plants*
- Regulatory Guide 1.117, *Tornado Design Classification*
- Sections 2.3.1 (Regional Climatology), 3.3.1 (Wind Loadings), and 3.3.2 (Tornado Loadings) of *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* (NRC 1987)
- *Tornado Climatology of the Contiguous United States* (Ramsdell and Andrews 1986).

The design basis tornado wind for the Yucca Mountain region is 304 km/hr (189 mi/hr), with a 0.000001 probability of occurrence and a 90 percent strike probability confidence interval (BSC 2001f, Section 5.2.1.6). This wind speed bounds both the 100-year return period fastest mile wind (100-year, 1-minute gust) referenced in *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants* (NRC 1987, Section 2.3.1) and the basic wind (50-year, 3-second gust) calculated from the methodology in Section 6 of ASCE 7-98, *Minimum Design Loads for Buildings and Other Structures*.

As with the tornado-generated missile event, potential defense-in-depth safety features to protect against tornado winds may include administrative controls in the event of a tornado warning or extreme weather conditions, hardened buildings, and the installation of underground utilities.

In summary, the repository structures, systems, and components deemed important to safety would be designed to withstand or to be protected from bounding external events and natural phenomena to prevent the release of radioactive material.

5.3.2 Preliminary Description of Internal Event Sequences

Radiological consequences for the bounding internal event sequences were evaluated. Bounding event sequences include groups of similar event sequences that result in the maximum radiological consequences to a member of the public at the preclosure controlled area boundary or to a worker onsite. Collectively, the bounding event sequences establish constraints on the facility design to ensure that structures, systems, and components important to safety would perform their intended function during an event sequence, and that any radiological releases would remain within established dose limits.

Internal event sequences were screened into one of three groups, based on their frequency of occurrence and potential to result in a radiological release:

- Internal event sequences with potential releases
- Internal event sequences with no releases
- Beyond Category 1 and Category 2 event sequences.

5.3.2.1 Internal Event Sequences with Potential Releases

These events could potentially result in a release of radionuclides, and would therefore be mitigated by the facility design. These events have been classified as Category 1 or Category 2 event sequences.

In *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f, Section 4.4.1.2.1), the impact of preclosure operational periods of up to 325 years on the internal events screening frequency thresholds (see Section 5.2.2) were investigated. For internal events that could impact the surface facility, the conclusion was that the results of using a 100-year preclosure period to screen internal event sequences would be unchanged by extending the period to 325 years since surface fuel handling operations would be completed after approximately 24 years. There would be no waste forms in the surface facility once the waste package subsurface emplacement operations are completed. *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f, Appendix A2.1) considered the increased number of waste packages for the lower-temperature thermal operating mode with de-rated or smaller waste packages (see Section 2.1.5.2, Table 2-2) and judged that the effect of additional waste package handling could increase the likelihood of some event sequences but would not change the selection of bounding event sequences that result in radionuclide releases. One potential approach to lowering the thermal output of waste packages is to age fuel by placing it into the fuel blending inventory (see Section 2.1.4). *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f, Appendix A6) judged that the handling and storage of fuel in this scenario is not expected to change the selection of bounding event sequences that result in radionuclide releases.

For the subsurface facility, extension of preclosure operations to 325 years does impact the screening criteria. However, *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f) examined the selection of internal event sequences based on an extended preclosure period and found no new internal events that would impact the selection of bounding event sequences. For example, the extended forced circulation ventilation activities in the subsurface facility after emplacement is completed, but before permanent closure, would not be expected to result in a loss of waste package containment.

All the thermal operating modes evaluated periods of forced ventilation (see Section 2.1.5.2, Table 2-2). Forced ventilation system failures are not expected to prevent the waste package from providing containment during the preclosure period. After waste emplacement is completed, it would take about 3 weeks without forced cooling before emplacement drift wall temperature limits are approached. Therefore, temperature goals supporting postclosure performance can be maintained by repairing and restarting the forced circulation equipment within about 3 weeks (see Section 2.3.4.3.1.3).

5.3.2.1.1 Category 1 Event Sequences—Internal

The Category 1 event sequences evaluated in *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f, Section 5.3.2) occurred during the handling of uncanistered commercial spent nuclear fuel assemblies or spent nuclear fuel assembly baskets in the assembly transfer system.

Table 5-6 identifies the Category 1 event sequences that could potentially result in radiological releases.

Sequences Involving Individual Spent Nuclear Fuel Assemblies—Unconfined spent nuclear fuel assemblies (i.e., assemblies not in containers) will be handled remotely, underwater and individually, during transfer from the cask to the assembly transfer system basket staging rack. Then they will be handled in a dry environment during transfer from the assembly transfer system dryer to the disposal container.

Table 5-6. Category 1 Internal Event Sequences

Event Sequence Number	Event Description	Location	Frequency (per year)
1-01	Spent fuel assembly drop onto another spent fuel assembly in cask	Assembly transfer system pool	0.2
1-02	Spent fuel assembly collision	Assembly transfer system pool	0.04
1-03	Spent fuel assembly drop onto empty basket	Assembly transfer system pool	0.04
1-04	Spent fuel assembly drop onto another spent fuel assembly in basket staging rack (lowering into)	Assembly transfer system pool	0.2
1-05	Basket drop onto another basket in basket staging rack (lifting out)	Assembly transfer system pool	0.04
1-06	Basket drop onto another basket in pool (transfer into pool storage)	Assembly transfer system pool	0.04
1-07	Basket drop onto another basket in pool (transfer out of pool storage)	Assembly transfer system pool	0.04
1-08	Basket drop onto transfer cart or pool floor	Assembly transfer system pool	0.04
1-09	Basket drop back into pool	Assembly transfer system pool	0.04
1-10	Basket drop onto assembly transfer system cell floor	Assembly transfer system cell	0.04
1-11	Basket drop onto another basket in dryer	Assembly transfer system cell	0.04
1-12	Spent fuel assembly drop onto another spent fuel assembly in dryer	Assembly transfer system cell	0.2
1-13	Spent fuel assembly drop onto assembly transfer system cell floor	Assembly transfer system cell	0.2
1-14	Spent fuel assembly drop onto another spent fuel assembly in disposal container	Assembly transfer system cell	0.2

Source: BSC 2001f, Table 5-5.

While underwater, spent nuclear fuel assemblies could be dropped or impacted as a result of a mechanical or control system failure of the wet assembly transfer machine, or as a result of operator error. These event sequences would occur in the assembly transfer system pool area, which is a confinement area with high-efficiency particulate air filtration. Individual spent fuel assembly event sequences that occur underwater are identified in Table 5-6 by sequence numbers 1-01 through 1-04.

During transfer from the dryer to the disposal container, individual spent fuel assemblies could be dropped or impacted as a result of a mechanical or control system failure of the dry assembly transfer machine, or operator error. These event sequences would occur in the assembly transfer system cell, which is a confinement area with high-efficiency particulate air filtration. Individual spent fuel assembly event sequences in the cell are identified in Table 5-6 by sequence numbers 1-12, 1-13, and 1-14.

The strategy is to confine particulate releases within the Waste Handling Building and maintain offsite radiological doses ALARA using the high-efficiency particulate air filters in the ventilation system.

Spent Fuel Assembly Basket Event Sequences—Spent nuclear fuel assembly baskets would first be handled underwater, during transfer out of the basket staging rack. From there the assembly baskets, which would contain a maximum of four pressurized water reactor spent nuclear fuel assemblies or eight boiling water reactor spent nuclear fuel assemblies, could be transferred and staged in the pool storage area to facilitate aging and blending or loaded directly into the incline transfer cart. Baskets that are staged in the pool area would have an additional step of movement from the storage pool to the incline transfer cart. Once loaded onto the incline transfer cart, assembly baskets would be transported out of the pool and into the assembly drying stations, where up to six baskets could be loaded into each of the two assembly dryers. The assembly transfer system pool and cell would both be located in confinement areas with high-efficiency particulate air filtration.

Spent nuclear fuel assembly baskets could be dropped or impacted in the pool during lifting out of the basket staging racks, during transport to the pool storage area, or during transport up the inclined transfer canal as a result of mechanical failures, control system failures, or operator error. Event sequences that occur underwater involving spent nuclear fuel assembly baskets are identified in Table 5-6 by sequence numbers 1-05 through 1-09.

The primary safety strategy is to confine radionuclide particulate releases to the assembly transfer system pool water by designing the pool system consistent with ANSI/ANS-57.7-1988, *American National Standard Design Criteria for an Independent Spent Fuel Storage Installation (Water Pool Type)*. The water treatment system will provide the capability to filter radioactive material, purify the water, and remove floating debris from the surfaces of pools. Workers will be able to use vacuums to remove particles from pool walls and floors (see Section 2.2.4.2.9). This same system provides the capability for cleanup of any radionuclide particulate releases into the pool water.

In addition, spent nuclear fuel assembly baskets can be dropped or impacted onto the floor or in one of the assembly dryers as a result of mechanical or control system failure of the dry assembly transfer machine or operator error. Spent nuclear fuel assembly basket sequences that occur in the cell are identified in Table 5-6 by sequence numbers 1-10 and 1-11.

5.3.2.1.2 Category 2 Event Sequences—Internal

The Category 2 event sequences evaluated in the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f, Section 5.3.3) would occur as a result of drops or collisions among handling equipment, unsealed disposal containers, or unsealed shipping casks. The bounding Category 2 internal event sequences that are expected to result in radiological releases are identified in Table 5-7.

Spent Nuclear Fuel Assembly Basket Collision During Transfer—A spent nuclear fuel assembly

Table 5-7. Category 2 Internal Event Sequences

Event Sequence Number	Event Description	Location	Frequency (per year)
2-01	Spent fuel assembly basket collision during transfer[a]	Assembly transfer system pool	0.007
2-02	Uncontrolled descent of incline transfer cart	Assembly transfer system pool	0.007
2-03	Handling equipment drop onto spent fuel assembly basket in pool[b]	Assembly transfer system pool	0.002
2-04	Handling equipment drop onto spent fuel assembly basket in cell[b]	Assembly transfer system cell	0.00007
2-05	Unsealed disposal container collision	Disposal container handling cell	0.002
2-06	Unsealed disposal container drop and slapdown	Disposal container handling cell	0.008
2-07	Handling equipment drop onto unsealed disposal container	Disposal container handling cell	0.0001
2-08	Unsealed transportation cask drop into cask preparation pit	Assembly transfer system cask preparation pit	0.009
2-09	Unsealed transportation cask drop into cask unloading pool	Assembly transfer system pool	0.009

NOTES: [a]This event encompasses two individual basket collision events, each with the same frequency and consequence.
[b]Event bounds the consequences of handling equipment drops onto a single spent fuel assembly and of a handling equipment drop onto a spent fuel assembly basket in the assembly transfer system pool.
Source: BSC 2001f, Table 5-6.

basket collides with a wall or other heavy object in the assembly transfer system pool, causing a breach and subsequent release. This event could occur during transfer either from the assembly basket rack to the pool area or from the pool area to the incline transfer cart. The pool water serves as a barrier to particulate release, so only the radioactive gases are released to the Waste Handling Building environment.

The primary safety strategy is to confine particulate releases within the assembly transfer system pool by designing the pool system consistent with ANSI/ANS-57.7-1988.

Uncontrolled Descent of Incline Transfer Cart—A remotely operated incline transfer cart containing a spent fuel assembly basket loses control during ascent up the incline transfer canal, resulting in an uncontrolled descent and impact with the assembly transfer system pool, which causes a breach and subsequent release. The pool water serves as a barrier to particulate release, so only the radioactive gases are released to the Waste Handling Building environment.

The primary safety strategy is to confine particulate releases within the assembly transfer system pool by designing the pool system consistent with ANSI/ANS-57.7-1988.

Handling Equipment Drop onto Spent Fuel Assembly Basket in Pool—A lifting yoke (or other heavy object) is dropped onto an uncanistered spent fuel assembly in the assembly transfer system pool, causing a breach and subsequent release. The pool water serves as a barrier to particulate release, so only the radioactive gases are released to the Waste Handling Building environment.

The primary safety strategy is to confine particulate releases within the assembly transfer system pool by designing the pool system consistent with ANSI/ANS-57.7-1988.

Handling Equipment Drop onto Spent Fuel Assembly Basket in Cell—A lifting yoke (or other heavy object) is dropped onto an uncanistered spent fuel assembly in the assembly transfer system cell, causing a breach and subsequent release.

The strategy is to confine particulate releases within the Waste Handling Building by relying on the high-efficiency particulate air filters in the heating, ventilation, and air conditioning system.

Unsealed Disposal Container Collision—A loaded, unsealed disposal container collides with a wall, shield door, or other heavy object, resulting in the release of a fraction of its radiological contents.

The strategy is (1) to confine particulate releases within the Waste Handling Building and maintain offsite radiological doses ALARA by using the high-efficiency particulate air filters in the heating, ventilation, and air conditioning system and (2) to provide design features (e.g., limit switches, redundant controls, emergency switch) and safe load paths that would minimize the likelihood of a collision that could result in a radiological release.

Unsealed Disposal Container Drop and Slapdown—A loaded, unsealed disposal container is dropped by the disposal container bridge crane onto a welding or staging fixture. After dropping, the unsealed disposal container is presumed to slap down onto the floor and release a fraction of its radiological contents. The drop height for this event is the normal handling height in the disposal container handling cell.

The strategy is (1) to confine particulate releases within the Waste Handling Building and maintain offsite radiological doses ALARA by using the high-efficiency particulate air filters in the heating, ventilation, and air conditioning system and (2) to provide design features (e.g., limit switches for lift height, interlocks, redundant controls, redundant cables, physical restraints) that would minimize unsealed disposal container drops and potential radiological releases.

Handling Equipment Drop onto Unsealed Disposal Container—A lifting yoke (or other heavy object) is dropped onto a loaded, unsealed disposal container, resulting in the release of a fraction of its radiological contents.

The strategy is (1) to confine particulate releases within the Waste Handling Building and maintain offsite radiological doses ALARA by using the high-efficiency particulate air filters in the heating, ventilation, and air conditioning system and (2) to provide design features that would minimize handling equipment drops onto spent nuclear fuel inside a disposal container.

Unsealed Transportation Cask Drop into Cask Preparation Pit—A transportation cask, without impact limiters and with its lid unbolted, is dropped from the normal lift height into the cask preparation pit in the assembly transfer system pool area.

The strategy is (1) to confine particulate releases within the Waste Handling Building and maintain offsite radiological doses ALARA by using the high-efficiency particulate air filters in the heating, ventilation, and air conditioning system and (2) to provide design features that prevent or minimize cask drops (e.g., limit switches, interlocks, redundant control circuitry, cable restraints) or reduce the impact of a drop (e.g., a shock absorber at the base of the pit).

Unsealed Transportation Cask Drop into Cask Unloading Pool—A transportation cask, without impact limiters and with its lid unbolted, is dropped by the cask bridge crane into the assembly transfer system cask unloading pool.

The strategy is to confine particulate releases within the assembly transfer system pool by designing the pool system consistent with ANSI/ANS-57.7-1988. In addition, particulate mitigation in the assembly transfer system pool area is provided by the secondary heating, ventilation, and air conditioning confinement ventilation system.

5.3.2.2 Internal Event Sequence with No Radioactive Material Release

For these event sequences, features of the design either prevent the event sequence from occurring or prevent a radionuclide release if the event occurs. Design features to prevent the event sequence can either physically prevent the event from occurring (e.g., by eliminating, at certain steps, the lifting of transportation casks or canistered waste) or reduce the event sequence frequency below the cutoff frequency of one in one million per year (e.g., by using redundant control features in cranes and control systems). Design features that prevent a release are based on the premise that Category 1 and Category 2 event sequences will occur and that affected structures, systems, and components must be designed to prevent the waste form from releasing radioactivity during such an event sequence. Prime examples of this include the waste package event sequences, which establish design bases for the waste package to ensure that the waste package will not breach as a result of Category 1 or Category 2 event sequences. Section 3.5 of this report provides waste package event sequence analyses. Table 5-7 of *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f, Section 5.3.4) identifies these events.

5.3.2.3 Beyond Category 1 and Category 2 Event Sequences

Beyond Category 1 and Category 2 event sequences are event sequences that have less than 1 chance in 10,000 of occurring before permanent closure. This corresponds to an annual frequency of less than 10^{-6} per year, based on an assumed preclosure lifetime of 100 years. Such event sequences are not analyzed further. However, structures, systems, and components reducing event sequences below 10^{-6} per year are considered in the design basis. Appendix A in *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f) considers the impact of lower-temperature operating modes on the identification of beyond Category 1 and Category 2 event sequences. The frequency of two events were found to be influenced by the thermal operating modes. These events are aircraft crash into the surface facility and rockfall onto a waste package in the subsurface facility. Aircraft hazards are impacted by increases in the surface facility's size, which would accompany an operating mode in which spent nuclear fuel is aged before being emplaced underground. However, Appendix A4.2 of *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f) considered the influence of the thermal operating modes on the surface facility size and concluded that the aircraft hazards are likely to remain beyond a Category 1 or Category 2 event sequence. Rockfall onto a waste package in the subsurface becomes more likely with increases in the preclosure period, which would accompany an operating mode with extended forced ventilation. However, Appendix A4.1 of *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f) considered the possible increase in the preclosure period and changes in the thermal operating modes on the drift temperature and concluded rockfall is likely to remain beyond a Category 1 or Category 2 event sequence with design optimization (e.g., optimized ground support features, waste package emplacement strategy). Table 5-12 of *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f, Section 5.4) identifies these events.

5.3.3 Consequence Evaluations

5.3.3.1 Category 1 Event Sequence Consequences

Design Basis Event Frequency and Dose Calculation for Site Recommendation (BSC 2001u) evaluated the consequences of Category 1 event sequences. Offsite radiation doses for Category 1 event sequences and normal operational effluents and emissions were based on the following (BSC 2001u, Section 6.1.1):

- Annual radiation doses for the sum of normal operational effluents and emissions, including all Category 1 event sequences
- Inhalation, ingestion, air immersion, and external exposure to radioactive contamination on the ground surface
- Mitigation, by high-efficiency particulate air filters, of particulate emissions from the repository
- For calculating the atmospheric releases from repository surface facilities, the distance to the receptor was 11 km (7 mi), the closest distance from the potential repository surface facilities to the potential site boundary. The ventilation exhaust locations for the subsurface areas of the repository are located about 3 km (2 mi) closer to the potential site boundary than the repository surface facilities. Therefore, for atmospheric emissions from the ventilation exhaust locations for the subsurface areas of the repository, the distance to the receptor is 8 km (5 mi) (BSC 2001u, Section 3.3)

- Annual average ground-level atmospheric dispersion factors (BSC 2001f, Section 5.3.5.3).

The bounding Category 1 event sequences evaluated for the potential repository are internal event sequences that occur during handling of uncanistered commercial spent nuclear fuel assemblies or spent nuclear fuel assembly baskets in the assembly transfer system (Table 5-6). No releases would occur due to external initiating events; therefore, external events have not been included in the dose calculations. No Category 1 event sequences have been identified for the subsurface facilities. All Category 1 event sequences would occur in surface facility confinement areas with high-efficiency particulate air filtration that is functional in the event sequences. The reliability of the heating, ventilation, and air conditioning system, used in event trees to calculate sequence frequencies, is based on the results of *Reliability Assessment of Waste Handling Building HVAC System* (CRWMS M&O 1999r).

The cumulative radiation doses for Category 1 event sequences, including normal operational effluents and emissions, are summarized in Table 5-8. The dose receptor is a member of the public located at the assumed site boundary. The cumulative radiation dose to this hypothetical average member of the public estimated for Category 1 event sequences was 0.06 mrem/yr total effective dose equivalent (BSC 2001f, Section 5.3.6.1). This very low dose is attributed to several factors, including:

1. The distance between the Waste Handling Building and the nearest unrestricted area of the site boundary (approximately 11 km [7 mi])
2. A maximum allowable radiation dose of 10 mrem/hr at a distance of 2 m (6 ft) from the edge of the transport vehicle
3. Shielding of radiation source within the Waste Handling Building
4. Shielding surrounding the waste package transporter.

All of the Category 1 event sequences occur either in cells, where workers would not be present and would be protected by shield walls, or in pool areas, where particulate radionuclides are retained by the pool water. In addition, the Waste Handling Building ventilation system is designed to control airflow, filter radionuclide particulates, and vent filtered emissions through an elevated stack to the external environment. Therefore, the potential radiation exposure for Category 1 event sequences for workers is calculated at a location outside the Waste Handling Building, at an assumed distance of 100 m (330 ft). These workers are not necessarily the workers that would be involved with

Table 5-8. Summary of Preclosure Category 1 Event Sequence Radiation Doses for the Public and Workers

Case	Dose Type	Radiation Dose
Offsite public, Category 1 event sequences (including normal operational effluents and emissions)	Total effective dose equivalent	0.06 mrem/yr[a]
	External exposure	<< 2 mrem/hr[a]
Workers, Category 1 event sequences (including normal operational effluents and emissions)	Total effective dose equivalent	0.01 rem/yr[b]
	Organ or tissue plus deep dose	0.10 rem/yr[b]
	Skin and extremities	0.13 rem/yr[b]
	Lens of the eye	0.15 rem/yr[b]
Workers, routine occupational exposures	Total effective dose equivalent	0.06 to 0.79 rem/yr[c]

NOTES: [a]BSC 2001f, Section 5.3.6.1—calculated for site boundary.
[b]BSC 2001u, Table 8.
[c]DOE 2002, Tables 4-22, 4-25, 4-28, and 4-31.

waste handling operations and exposed during routine operations; therefore, the radiation doses for workers from Category 1 event sequences are not added to the radiation doses from routine occupational exposures.

The radiation dose to the worker at 100 m (330 ft) from Category 1 event sequences was estimated to be 0.01 rem/yr (BSC 2001u, Table 8). The largest radiation dose to any organ or tissue other than the lens of the eye, plus the deep dose equivalent, was estimated to be 0.10 rem/yr total effective dose equivalent (BSC 2001u, Table 8). The radiation dose to the skin and extremities was estimated to be 0.13 rem/yr total effective dose equivalent (BSC 2001u, Table 8). The radiation dose to the lens of the eye is estimated to be 0.15 rem/yr total effective dose equivalent by summing the total effective dose equivalent and the skin dose (BSC 2001u, Table 8).

The radiation doses for routine occupational exposures for workers are estimated in *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE 2002, Section 4.1.7) and summarized in Table 5-8. Maximum radiation doses ranged from about 0.06 to 0.79 rem/yr total effective dose equivalent, depending on the area of the repository, the phase of operation, and the thermal load alternative (DOE 2002, Tables 4-22, 4-25, 4-28, and 4-31). Section 7 of *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (BSC 2001f) discusses methods that would be used to ensure that occupational radiation doses are ALARA. Worker safety from industrial hazards was also discussed in *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE 2002, Section 4).

5.3.3.2 Category 2 Event Sequence Consequences

Design Basis Event Frequency and Dose Calculation for Site Recommendation (BSC 2001u) evaluated the consequences of Category 2 event sequences. Offsite radiation doses (i.e., in the uncontrolled area) for Category 2 event sequences were based on the following (BSC 2001u, Section 6.1.2):

- Inhalation and air immersion pathways
- Release fractions that take into account the respirable fraction of radionuclide particulates
- Mitigation by high-efficiency particulate air filters of particulate emissions from the surface facilities
- For calculating the atmospheric emissions from repository surface facilities, the distance to the receptor was 11 km (7 mi), the closest distance from the potential repository surface facilities to the assumed site boundary (BSC 2001u, Section 3.3)
- 99.5 percent ground-level atmospheric dispersion factors (BSC 2001f, Section 5.3.5.3).

The radiation doses from bounding Category 2 event sequences were calculated assuming filtration through a high-efficiency particulate air filter. The bounding-consequence Category 2 event sequence is the drop of an unsealed shipping cask.

The highest radiation dose for a member of the public caused by the bounding-consequence Category 2 event sequence was 0.02 rem (BSC 2001f, Section 5.3.6.1). The largest radiation dose to any organ or tissue other than the lens of the eye was estimated to be 0.1 rem total effective dose equivalent (BSC 2001f, Section 5.3.6.1). The radiation dose to the skin and the lens of the eye was estimated to be total effective dose equivalent of 0.04 rem and 0.06 rem, respectively (BSC 2001f, Section 5.3.6.1).

5.4 PRECLOSURE SAFETY: TEST AND EVALUATION PROGRAM

The Monitored Geologic Repository Test and Evaluation Program will include planning, execution, and documentation of the testing, examination, analyses, and demonstrations necessary to verify safe and efficient operation of the repository. The preclosure components of this comprehensive program address all aspects of verification, from the development of test requirements and acceptance criteria to the performance, recording, and reporting of test procedures. The following discussion of the test and evaluation program is based on *Monitored Geologic Repository Test & Evaluation Plan* (CRWMS M&O 2000fj). The test and evaluation plan will be revised at the time of preparation of any license application for conformance of the plan to more specific design information and any additional performance related testing.

This test and evaluation program would include the following activities and objectives.

- Design and component testing will ensure that structures, systems, and components are designed as specified and perform as required.
- Preoperational testing will ensure that structures, systems, and components operate on an integrated basis and will verify processes and validate procedures for the receipt, preparation, emplacement, and movement (i.e., recovery or retrieval) of waste.
- Operational testing will confirm exposure times and radiation levels during repository operations, and that operational safety has been incorporated into structures, systems, and components.

To achieve these objectives, the test and evaluation program defines, plans, and implements a set of integrated test activities focused on ensuring preclosure safety (CRWMS M&O 2000fj, Section 2). These integrated activities are:

- Development testing
- Prototype testing
  - Proof of concept testing
  - Mockup testing
- Component testing
- Construction and preoperational testing
- Hot startup testing
- Periodic performance testing and surveillance.

A confirmation verification tracking system would identify the tests performed throughout the test program. This tracking system would status the program's performance and would be maintained and updated as a test database that would provide a history of structure, system, and component performance. It would be made available to support the licensing process and the operations, maintenance, system upgrade, and support functions.

5.4.1 Development Testing

Development testing supports design activities by confirming design concepts, evaluating alternative design concepts, and investigating the availability of needed technology. For example, development testing will help evaluate and demonstrate the suitability of ground support systems proposed for the emplacement drifts. Development testing will also help evaluate the suitability, adequacy, and availability of instrumentation, monitoring, and control technologies for use in the subsurface environment.

The repository systems would use microprocessor-based instrumentation and control equipment, including operator control stations, digital data acquisition, data processing, network and communications equipment, borehole instrumentation, air sampling instruments, and infrared cameras. Having a good understanding of the reliability of these systems in a high-temperature and high-radiation repository environment is important to ensure public and worker safety during emplacement activities. Field testing of candidate technologies would investigate how to minimize downtime from failures.

5.4.2 Prototype Testing

Prototype testing includes proof of concept testing and mockup testing.

Proof of Concept Testing—Proof of concept prototype testing is performed for the following cases:

- New technologies or design solutions that have little or no history of use at existing nuclear storage facilities or power plants

- Technologies or design solutions that have not been subjected to a test program and are qualified by the NRC or DOE, as applicable, and from which accepted data were collected or analyzed and documented in a defensible source.

This prototype testing would support the development of structures, systems, and components during construction and preoperation (CRWMS M&O 2000fj, Appendix C).

Mockup Testing—While proof of concept testing supports the design process, mockup testing involves simulation or demonstration with operational realism. Mockup testing follows proof of concept testing and supports preoperational and operational activities.

5.4.3 Component Testing

Component testing, if needed, would be performed as part of the procurement process to establish equipment qualification according to the applicable quality level. Component testing, which includes qualification and acceptance testing, would be used for any unique (not off-the-shelf) equipment. Qualification testing verifies, on a limited sampling basis, the proper operation of the component with respect to extreme bounds (as defined by specifications). Acceptance testing, performed for key parameters, establishes confidence that the manufacturing process is producing the correct product. The component vendor, with quality assurance oversight and concurrence, performs component testing. This testing starts at the beginning of fabrication and is completed before installation.

Compliance with identified safety and radiological requirements would be assessed during component testing to document the appropriate details for test performance. Examples of component testing include shock, vibration, and environmental testing for performance of sensors and alarms that have or support safety functions.

5.4.4 Construction and Preoperational Testing

Construction and preoperational testing would begin during repository construction and end before receipt of waste. This test activity includes the following subactivities:

- Installation and checkout testing
- System integration testing
- Event sequence recovery testing
- Cold startup testing.

5.4.5 Hot Startup Testing

To the extent practicable, the preoperational testing described previously would verify compliance with repository performance requirements, including ALARA considerations. Hot startup testing would verify that operation and maintenance systems work properly and confirm that exposure times and radiation levels fall within acceptable limits during actual repository operations. Hot startup testing would begin after the successful completion of construction and preoperational test activities. It would include the following subactivities:

- Testing and confirming exposure times and radiation dose
- Verifying heat removal features and cooling systems
- Confirming acceptable radiation exposure levels.

5.4.6 Periodic Performance Testing and Surveillance

Periodic performance testing would verify system performance and ensure continued proper functioning of structures, systems, and components important to radiological safety, waste isolation, fire protection, nonnuclear safety, and repository operations. Periodic testing would be performed at the Waste Handling Building and the Waste Treatment Building in the surface facilities and at the emplacement drift panels in the subsurface facilities. This testing would also be performed after maintenance and repair activities.

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GLOSSARY

3DEC. A heat transfer and distinct element stress-analysis software code used to simulate thermal-mechanical behavior of a rock mass.

ablation. The process of removing by cutting, erosion, melting, evaporation, or vaporization.

abstracted model. Model that reproduces, or bounds, the essential elements of a more detailed process model and captures uncertainty and variability in what is often, but not always, a simplified or idealized form. *See* abstraction.

abstraction. The essential components of a process model that are extracted for use in a total system performance assessment. The abstraction retains the basic intrinsic form of the process model but does not usually require its original complexity.

access main. Horizontal drift that provides access to waste emplacement drifts. *Also*, main drift.

accessible environment. Any point outside of the controlled area, including: (1) the atmosphere (including the atmosphere above the surface area of the controlled area), (2) land surfaces, (3) surface waters, (4) oceans, and (5) the lithosphere.

actinide. Any element of the actinide series, a series of chemically similar, mostly synthetic, radioactive elements with atomic numbers from 89 (actinium) through 103 (lawrencium).

acute dose. The maximum radiation dose that an individual at the site boundary is expected to receive over a relatively short period (e.g., two hours).

adsorb. To collect a gas, liquid, or dissolved substance on a surface.

adsorption. Transfer of solute mass, such as radionuclides, in groundwater to the solid geologic surfaces with which it comes in contact. The term *sorption* is sometimes used interchangeably with this term.

advection. The process in which solutes are transported by groundwater movement.

ALARA (as low as is reasonably achievable). A process that applies a graded approach to reducing dose levels to workers and the public, and releases of radioactive materials to the environment. The goal of this process is not merely to reduce doses to levels specified by regulations, but to reduce them to levels that are as low as reasonably achievable.

alcove. A small excavation (room) off a main drift. Used for scientific study or for installing equipment.

alkaline. (1) Of, relating to, containing, or having the basic chemical properties of an alkali or alkali metal. (2) Having a pH of more than 7. *See* pH.

Alloy 22. A high-nickel alloy used for the outer barrier of the waste package.

alluvium. Sedimentary material (clay, mud, sand, silt, gravel) deposited by a stream or running water.

alpha particle. A positively charged particle ejected spontaneously from the nuclei of some radioactive elements. It is identical to a helium nucleus and has a mass number of 4 and an electrostatic charge of +2. It has low penetrating power and a short range (a few centimeters in air). *See* ionizing radiation.

ambient. Undisturbed, natural conditions such as ambient temperature caused by climate or natural subsurface thermal gradients.

analogue. Natural or man-made systems that include processes similar to those that could occur in a repository system at Yucca Mountain for which information can be obtained to evaluate long-term (e.g., millennia) or large-scale (e.g., kilometers) behavior.

anion. An atom or group of atoms having a negative charge; a negatively charged ion.

anisotropy. The condition in which physical properties vary when measured in different directions or along different axes. For example, in a layered rock section the permeability is often anisotropic in the vertical direction (from layer to layer) but is isotropic in the horizontal direction (within a layer). *See* isotropy.

annealing. Alternately heating and cooling a metal, alloy, or glass to relieve internal stresses in the material.

annual committed effective dose equivalent. A radiation protection term for the effective dose equivalent received by an individual in one year from radiation sources external to the individual plus committed effective dose equivalent.

ANSYS. A finite element code used in thermal and structural dynamic analyses (e.g., ground support design, performance confirmation, engineered barrier components).

aquifer. A subsurface formation or group of formations that is saturated and sufficiently permeable to transmit quantities of water to wells or springs.

areal mass loading. Used in thermal loading calculations, the amount of heavy metal (usually expressed in metric tons of uranium or equivalent) emplaced per unit area in the repository.

arid. Of a climate: very dry; a region in which annual precipitation is less than approximately 250 mm (10 in.). On average, Yucca Mountain receives about 190 mm (7.5 in.) of rain and snow annually.

ash. Fine or very fine pyroclastic particles, less than 4 mm (0.15 in.) in diameter, that are blown out from a volcanic explosion.

ash-fall tuff. Highly-porous volcanic rock that is a result of magma thrown high into the atmosphere, where it cooled and fell back to earth as a blanket of ash.

ash-flow tuff. Dense, nonporous volcanic rock that is a result of magma flowing at ground level where it remained at a high temperature for a long time, welding magma and ash together.

ASHPLUME. A computer software code used to simulate the ash plume that results from an extrusive volcanic event.

asperity. Roughness or irregularity of surface along the boundary between the walls of faults, joints, or fractures.

assembly transfer system. System in the Waste Handling Building to transfer uncanistered spent nuclear fuel assemblies from the shipping casks to disposal containers.

autocatalytic criticality. A transient criticality in which the usual mechanisms that tend to shut down a criticality are delayed until a high fission rate is achieved.

backfill. The general fill that is placed in the excavated areas of the underground facility. If used, the backfill for the repository may be tuff or other material.

background radiation. Radiation arising from natural radioactive material always present in the environment, including solar and cosmic radiation, radon gas, soil and rocks, food, and the human body.

barrier. Any material, structure, or feature that prevents or substantially reduces the rate of movement of water or radionuclides.

base case. The sequence of anticipated conditions expected to occur in and around the potential repository, without the inclusion of unlikely or unanticipated features, events, or processes. The components that contribute to the base case model are intended to encompass this probable behavior of the repository, based on the range of uncertainty for the various parameters and conceptual models used in constructing the base case.

base case. Design that addresses a nuclear waste storage capacity of 70,000 MTHM (63,000 MTHM commercial spent nuclear fuel and 7,000 MTHM DOE high-level radioactive waste). *Also,* statutory case.

binning. (1) A project-specific term that refers to the process of prioritizing systems, structures, and components based on their importance to radiological safety as well as regulatory or design precedent. (2) In computer modeling of a system, a process of averaging relevant variables over a typical subset of the system, while preserving variability. Used in the TSPA; for example, binning provides multiscale model results for waste package temperature, relative humidity at the waste package surface, and the percolation flux in the host rock 5 m (16 ft) above the emplacement drift.

bioaccumulation. Means by which a living organism could ingest, inhale, or otherwise internally accumulate a foreign substance such as a radioactive particle.

biosphere. The ecosystem of the earth and the living organisms inhabiting it.

biosphere dose conversion factor. A multiplier used in converting a radionuclide concentration at the geosphere/biosphere interface into a dose that a human would experience for all pathways considered, with units expressed in terms of annual dose (i.e., the total effective dose equivalent) per unit concentration.

block-bounding fault. A normal fault that divides the crust into structural or fault blocks of different elevations and orientations; typical of the Basin and Range province.

boiling-water reactor. A nuclear power reactor in which water passing as coolant through the core is turned to steam by direct use of fission heat from the uranium oxide fuel; steam for driving the turbogenerator is formed within the reactor vessel itself rather than in an external heat exchanger and, after being condensed, returns as feedwater to the reactor vessel.

bomb-pulse. A detectable increase in radionuclide concentration related to global fallout from above-ground thermonuclear tests.

borehole. A hole drilled for purposes of collecting site characterization data or for supplying water. Sometimes referred to as a drillhole or well bore.

borosilicate glass. A material used to vitrify high-level radioactive waste in which boron takes the place of the lime used in ordinary glass mixtures.

boundary condition. For a model, the establishment of a set condition (set value), often at the geometric edge of the model, for a given variable; for example, using a specified groundwater flux from infiltration as a boundary condition for a flow model.

bounding. For a mathematical model, relating to a set condition (set value) that is considered to reflect the reasonable extreme for that value in the real-world condition being modeled. *See* boundary condition, bounding value.

bounding value. Specific data point that defines the reasonable extreme limit of a variable in an experiment or model.

breakthrough. The time at which the concentration of a substance, usually in groundwater, arrives at a particular point of interest.

breakthrough curve. The curve describing the rate of arrival of radionuclides transmitted through a medium. The breakthrough curve calculation includes the effects of all flow modes, flow in rock matrix, flow in fractures, and retardation and determines the expected proportion of the radionuclide mass transmitted through the medium as a function of time.

bridge crane. A large overhead crane used for material handling that spans across rails on either side of a structure.

Brownian motion. Random movement of small particles suspended in a fluid. Caused by pressure fluctuations over the surfaces of the particles as they interact with the fluid's molecules.

buoyant convection. Fluid movement, typically in the gas phase, in response to a density gradient. An example is the rising of air when it becomes less dense because of heating followed by its subsequent fall when it cools and becomes denser.

burnup. A measure of nuclear-reactor fuel consumption expressed either as the percentage of fuel atoms that have undergone fission or as the amount of energy produced per initial unit weight of fuel.

burnup credit. A factor used in criticality calculations that accounts for the amount of burnup in certain fuel types. As burnup increases, the capability of the nuclear fuel to support criticality decreases. *See* burnup.

caldera. A large basin-shaped volcanic depression, more or less circular, resulting from the collapse of the ground surface following a rapid volcanic eruption. A caldera may be more than 15 km (9 mi) in diameter and more than 1,000 m (3,300 ft) deep.

calibration. (1) In modeling, the process of comparing the conditions, processes, and parameter values used in a model against actual data points or interpolations from measurements at or close to the site to ensure that the model is compatible with "reality" to the extent feasible. (2) For tools used for field or lab measurements, the process of taking instrument readings on standards known to produce a certain response to check the accuracy and precision of the instrument.

canister. The structure surrounding some forms of waste (e.g., high-level radioactive waste immobilized in glass logs or ceramic disks within cans) that facilitates handling, storage, transportation, and/or disposal.

capillary barrier. A contact in the unsaturated zone between a geologic unit containing relatively small-diameter openings and a unit containing relatively large-diameter openings. Water does not flow from the former to the latter due to capillary forces.

capillary force. A phenomenon that results from the force of molecular attraction (adhesion) between a fluid and different solid materials; this force that causes water to rise in small diameter tubes and, in combination with the effects of gravity, is a means of water movement in the unsaturated zone.

capillary pressure. The pressure due to capillary forces; i.e., the pressure of fluids under the influence of surface tension and adhesion.

Carrier Preparation Building. Surface facility where waste transportation casks and their carriers are prepared before they enter the Waste Handling Building.

carrier/cask handling system. System in the Waste Handling Building that unloads the casks from the carriers and transfers the casks to either the assembly transfer system or the canister transfer system. The system also receives empty casks from the assembly and canister transfer systems and nondisposable canister overpacks from the assembly transfer system and loads them onto carriers for transfer to the Carrier Preparation Building.

cask. A large, shielded container for shipping or storing spent nuclear fuel and/or high-level radioactive waste that meets all applicable regulatory requirements.

cation. An atom or group of atoms having a positive charge; a positively charged ion.

chronic dose. Of a radiation dose, the annual exposure to an individual living at the site boundary and continuously exposed to an averaged level of exposure over a long period of time.

cinder cone. A conical hill formed by the accumulation of cinders and other pyroclasts around a volcanic vent.

cladding. The metallic outer sheath of a fuel rod element generally made of a zirconium alloy. It is intended to isolate the fuel element from the external environment.

clay. A rock or mineral fragment of any composition that is smaller than very fine silt grains, having a diameter less than 0.00016 in. (1/256 mm). A clay mineral is one of a complex and loosely defined group of finely crystalline hydrous silicates formed mainly by weathering or alteration of primary silicate minerals. They are characterized by small particle size and their ability to adsorb large amounts of water or ions on the surface of the particles.

climate. Weather conditions, including temperature, wind velocity, precipitation, and other factors, that prevail in a region averaged over some period of time.

climate state. Representation of climate conditions. Different climate states are used in performance assessment models to represent changes in climate over the time periods of interest.

closure seal. A generic term for the method(s) that would be used to seal all openings (e.g., access ramps, ventilation shafts, and exploratory boreholes) from the surface to the underground repository facilities once a decision is made to permanently close the repository. If left unsealed, the openings could enhance the movement of moisture from the surface into the waste emplacement area and could also serve as conduits for airborne radioactive material to migrate into the atmosphere.

code (computer). The set of commands used to solve a mathematical model on a computer.

codisposal. A packaging method for disposal of radioactive waste in which more than one type of waste, such as DOE spent nuclear fuel and high-level radioactive waste, are combined in disposal containers. Codisposal takes advantage of otherwise unused space in disposal containers and is more cost-effective than other methods of limiting the reactivity of individual waste packages.

cohesion. (1) In geology, the shear strength of the substance, whether cement or adsorbed water, that separates individual grains of rock at their areas of contact. (2) In physics, the attraction between molecules of a liquid that allows formation of drops or films of that liquid.

colloid. A large molecule or small particle that has at least one dimension with the size range of 10^{-9} to 10^{-6} m, suspended in a liquid such as groundwater. Some radionuclides can bind with colloids (either reversibly or irreversibly) depending on the type of colloid) and travel great distances in groundwater. Colloids may form directly from insoluble radioactive material (intrinsic colloids), may result from degraded spent nuclear fuel or glass waste forms (waste form colloids), or may result from other natural or man-made materials with which radionuclides can bind (pseudocolloids).

colluvium. Any loose, heterogeneous sediment deposited by rainwash, sheetwash, or slow continuous downslope creep, usually at the base of a cliff or slope.

committed effective dose equivalent. A radiation protection term for the effective dose equivalent received over a period of time (as determined by the U.S. Nuclear Regulatory Commission) by an individual from radionuclides internal to the individual following one year intake of these radionuclides.

committed dose equivalent. A radiation protection term for the dose equivalent that is committed to specific organs or tissues that will be received from an intake of radioactive material by an individual during the 50 years following the intake.

complexation capability. The ability of a chemical element to unite with other elements to form a complex compound.

component model. The analysis models that are run separately and then combined into process models for running in the TSPA computer model.

conceptual model. A set of qualitative assumptions used to describe a system or subsystem for a given purpose.

conduction. The flow of thermal energy through a material. Conduction is affected by the amount of heat energy present, the nature of the heat carrier in the material (its thermal conductivity), and the amount of dissipation.

confidence. (1) In statistics, a measure of how close the estimated value of a random variable is to its true value. (2) Degree of assurance that an argument, such as a safety case, is correct.

confidence interval. An interval that is believed, with a preassigned degree of confidence, to include the particular value of the random variable that is estimated.

conservative assumption. (1) An assumption that results in a calculated release of radionuclides exceeding actual or expected releases. (2) An assumption that uses uncertain inputs and does not attempt to include any potentially beneficial effects.

containment. (1) The confinement of radioactive waste within a designated boundary. (2) The use of design features to contain or reduce radioactive releases or radiation doses.

continuum model. A model that represents fluid flow through numerous individual fractures and matrix blocks by approximating them as continuous flow fields.

controlled area. The surface area, identified by passive institutional controls, that encompasses no more than 300 square kilometers. It must not extend farther south than 36° 40' 13.6661" North latitude, in the predominant direction of groundwater flow and must not extend more than five kilometers from the repository footprint in any other direction. It also includes the subsurface underlying the surface area.

convection. (1) The transfer of heat by the circulation of a fluid (at Yucca Mountain, either water or air). (2) The bulk motion of a flowing fluid (gas or liquid) caused by temperature differences that, in turn, cause different areas of the fluid to have different densities (e.g., warmer is less dense).

convective circulation. The transfer of heat from or within a given area by movement of a liquid or gas having a temperature different than the solids over which it flows.

corrosion resistant material. A material that develops a protective film on its surface, creating a high resistance to corrosion. Such a material, the nickel-based alloy, Alloy 22, would be used as the outer barrier of the two-layer waste package.

coupon. A strip of polished metal, of specific weight and size, used in testing to assess the corrosive effects of liquids or gases. Also used to measure the effectiveness of corrosion inhibitors.

creep. A phenomenon in which strain in a solid increases with time when stress producing the strain is held fixed; it may be associated with temperature or mechanical stress.

critical group. The hypothetical group of individuals reasonably expected to receive the greatest exposure to radioactive materials (proposed 10 CFR Part 63). The critical group has been replaced by the reasonably maximally exposed individual in the final NRC regulations (10 CFR 63.2).

criticality. The condition in which nuclear fuel sustains a chain reaction. It occurs when the effective neutron multiplication factor of a system equals one. *See* effective neutron multiplication factor.

Cross-Drift, Enhanced Characterization of the Repository Block (ECRB) Cross-Drift. An excavation above and across the block of the potential repository excavated in a general northeast–southwest direction.

cumulative distribution. For grouped data, a distribution that shows how many of the values are less than or more than specified values. For random variables, this term is synonymous with distribution function.

cumulative probability. The probability that a random variable will have a value equal to or less than some specified value.

curie. A unit of radioactivity equal to 37 billion disintegrations per second, abbreviated Ci.

darcy. A unit of measurement of the permeability of a porous medium. In a cross section of the medium 1 square centimeter by 1 cm in length, 1 darcy equals the passage in 1 second of 1 cm^3 of the fluid when it has 1 centipoise of viscosity under 1 atmosphere of pressure.

Darcy's law. A fundamental law of porous media, discovered by Henri Darcy in 1856, stating that the flow rate Q is proportional to the cross-sectional area A, inversely proportional to the length L of the sand-filter flow path, proportional to the head drop δH, and proportional to the hydraulic conductivity K. Used in hydrology to describe fluid flow in a porous medium.

decommission. The process of removing from service a facility in which nuclear materials are handled. This usually involves decontaminating the facility so that it may be dismantled or dedicated to other purposes.

DECOVALEX. An international software code testing program.

defense in depth. (1) A design strategy based on a system of multiple, independent, and redundant barriers, designed to ensure that failure in any one barrier does not result in failure of the entire system. (2) A system of multiple barriers that mitigate uncertainties in conditions, processes, and events.

deliquescence. The absorption of water vapor from the air by a crystalline solid, leading to dissolution of the solid.

design alternative. (1) A considered alternative to a major design feature that is important to waste isolation. (2) A fundamentally different conceptual repository design, which could stand alone as the License Application repository design concept.

design bases. Information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be constraints derived from generally accepted state-of-the-art practices for achieving functional goals or requirements derived from analysis (based on calculation or experiments) of the effects of a postulated event under which a structure, system, or component must meet its functional goals. The values for controlling parameters for external events include: (1) estimates of severe natural events to be used for deriving design bases that will be based on consideration of historical data on the associated parameters, physical data, or analysis of upper limits of the physical processes involved and (2) estimates of severe external human-induced events, to be used for deriving design bases, that will be based on analysis of human activity in the region taking into account the site characteristics and the risks associated with the event.

design basis event. (1) Natural or human-induced events that may occur before permanent closure of the geologic repository's operations area and which are used to assess system safety. (2) A natural or human-induced event that is expected to occur one or more times before permanent closure of the repository (referred to as a *Category 1 Event*) or any other natural or human-induced event that has at least one chance in 10,000 of occurring before permanent closure of the repository (referred to as a *Category 2 Event*).

design margin. Margins of safety in specifications for engineered components to account for uncertainty in the conditions to which the components will be subjected and for variability in the properties of component materials. *See* safety margin.

desorption. A physical or chemical process by which a substance that has been adsorbed or absorbed by a liquid or solid material is removed from the material.

devitrification. The conversion of a glassy substance to a crystalline substance; for example, the alteration of glass in vitrified tuff into zeolites.

diagenetic process. The chemical or physical changes that take place in sediments during and after they are deposited but before they consolidate.

diffusion. (1) A process in which substances move from regions of higher concentrations to regions of lower concentrations. (2) The gradual mixing of the molecules of two or more substances due to random thermal motion.

diffusion coefficient. A material's weight, in grams, as it diffuses in 1 second through 1 square centimeter of a medium of a determined concentration gradient (i.e., a medium having a known variability in the concentration of the substance in solution as it travels over distance).

diffusive transport. Movement of molecules or particles due to their concentration gradient. Occurs when dissolved or suspended radionuclides move from regions of higher or lower concentration.

dike. A tabular body of igneous rock that cuts across the structure of adjacent rocks or cuts massive rocks. Most dikes are caused by the intrusion of magma. Some dikes occur as columnar structures.

dilution. The reduction of a dissolved substance's concentration in a solution caused by an increase in the solvent's proportion of the solution. The solvent frequently is water.

dispersion (hydrodynamic dispersion). (1) The tendency of a solute (substance dissolved in groundwater) to spread out from the path it is expected to follow if only the bulk motion of the flowing fluid moved it. (2) The macroscopic outcome of the actual movement of individual solute particles through a porous medium. Dispersion causes dilution of solutes, including radionuclides, in groundwater and is usually an important mechanism for spreading contaminants in low flow velocity situations.

dispersivity. The degree to which small particles of a solid are distributed throughout either a liquid or another solid.

disposal container. The vessel consisting of the barrier materials and internal components in which the canistered or uncanistered waste form is placed. The disposal container includes the container barriers or shells, spacing structures or baskets, shielding integral to the container, packing contained within the container, and other absorbent materials designed to be placed internal to the container or immediately surrounding the disposal container (i.e., attached to the outer surface of the container). The filled, sealed, and tested disposal container is referred to as the waste package, which is emplaced underground.

disposal container handling system. System in the Waste Handling Building that prepares empty disposal containers for loading, receives full disposal containers from the assembly and canister transfer systems, welds and inspects the containers, and transfers them to the waste emplacement system. The system also receives and handles retrieved waste packages from the subsurface and disposal containers that are defective and routes them to the waste package remediation system.

disruptive processes and events. An unexpected process or event that could affect the performance of the repository, including, for example, human intrusion, volcanic activity, and seismic activity.

disruptive scenario. A well-defined sequence of events and processes that could adversely affect repository performance initiated by a disruptive process or event.

dissolution. The dissolving of a solid or gas in a liquid.

distribution. The overall scatter of values for a set of observed data. A term used synonymously with frequency distribution. Distributions have probability structures that are the probability that a given value occurs in the set.

distribution frequency. A representation of how values of an outcome or variable are distributed over the range of expected values.

distribution function. A function whose values are the probabilities that a random variable assumes a value less than or equal to a specified value. Synonymous with cumulative distribution.

DOE spent nuclear fuel. Radioactive waste created by defense activities. The major contributor to this waste form is the N Reactor fuel currently stored at the Hanford Site. This waste form also includes naval spent nuclear fuel.

dose. The amount of radioactive energy taken into (absorbed by) living tissues.

dose equivalent. The product of the absorbed dose in tissue, quality factor, and all other necessary modifying factors at the location of interest. *See* effective dose equivalent *and* total effective dose equivalent.

downgradient. The direction that water will tend to flow as the result of a difference in pressure, as indicated by the elevation change in the potentiometric surface. Based on current understanding of the hydraulic gradient below Yucca Mountain, downgradient is toward the south to southeast of the potential repository location.

DCPT. Used as a dual continuum particle tracking code for modeling transport in dual media, such as fractures and rock maxtrix.

drift. From mining terminology, a horizontal underground passage. Includes excavations for emplacement (emplacement drifts) and access (main drifts).

drip shield. A corrosion-resistant engineered barrier that is placed above the waste package to prevent seepage water from directly contacting the waste package for thousands of years. The drip shield also offers protection to the waste package from rockfall.

DRKBA. Software used to apply a numerical technique for solving probabilistic key block analysis problems in a rock mass according to probabilistic distributions determined based on tunnel mapping data.

dual permeability conceptual model. A conceptual model of groundwater flow in which fractures and rock matrix are represented as separate, interacting continua, with no assumption of pressure equilibrium between fractures and rock matrix. This concept allows modeling groundwater flow as occurring mostly in the fractures, with less flow in the rock matrix, depending on the degree of connection between the rock matrix and fractures and the capillary pressure gradient. The dual permeability model is one of the conceptual models for groundwater and heat flow for fractured, porous media.

effective dose equivalent. The sum of the products of the dose equivalent received by specified tissues and the appropriate weighting factors applicable to each tissue.

effective neutron multiplication factor (K_{eff}). A measurement of nuclear reactivity or criticality potential. Equal to the number of fissions in one generation divided by the number of fissions in the preceding generation, in a finite medium.

effective porosity. The fraction of a given medium's porosity available for fluid flow and/or solute storage.

Eh. A measure of the state of oxidation of a system. Also known as redox potential or oxidation–reduction potential.

electrical resistivity tomography. A radiograph that shows the electrical resistance of a material within a predetermined plane section.

electrolyte. A substance (e.g., an acid, base, or salt) that, when dissolved in a suitable solvent (e.g., water) or when fused, conducts electric current by the movement of ions instead of electrons

empirical model. A model whose reliability is based on observation and/or experimental evidence and is not necessarily supported by any established theory or law. Validity or applicability of such an empirical model is normally limited to situations that lie within the range of the data that were used to develop the model.

emplacement. The placement and positioning of waste packages in the repository emplacement drifts.

emplacement horizon. The area within the repository block where emplacement drifts would be excavated.

engineered barrier. Any component of the engineered barrier system, such as the drip shield, waste package, or invert. *See* engineered barrier system.

engineered barrier system. The waste packages and the underground facility, including engineered components and systems other than the waste package (e.g., drip shields).

Environmental Impact Statement. A detailed written statement to support a decision to proceed with major Federal actions affecting the quality of the human environment. This is required by the National Environmental Policy Act of 1969. Preparation of an environmental impact statement requires a public process that includes public meetings, reviews, and comments, as well as agency responses to the public comments.

eolian deposit. Material deposited by wind, such as sand dunes.

EQ3/6. A computer software code used to estimate equilibrium mineral phases based on thermodynamic equilibrium, thermodynamic disequilibrium, and reaction kinetics.

EQ3NR. Aqueous solution speciation-solubility code component of EQ3/6 that computes thermodynamic state of a solution.

equivalent continuum model. A conceptual model of groundwater and heat flow that is also called a composite porosity model. Key assumptions are that the temperatures and capillary pressures in the rock matrix and fractures are equal. Therefore, the fractures and matrix can be treated as a single composite material, and the hydraulic properties are a combined effect of both fracture and matrix properties.

evapotranspiration. The combined processes of evaporation and plant transpiration that remove water from the soil and return it to the air.

event. (1) An occurrence that has a specific starting time and, usually, a duration shorter than the time being simulated in a model. (2) Uncertain occurrences that take place within a short time relative to the time frame of the model.

event sequence. A series of actions and/or occurrences within the natural and engineered components of a geologic repository operations area that could potentially lead to exposure of individuals to radiation. An event sequence includes one or more initiating events and associated combinations of repository system component failures, including those produced by the action or inaction of operating personnel. Those event sequences that are expected to occur one or more times before permanent closure of the geologic repository operations area are referred to as Category 1 event sequences. Other event sequences that have at least one chance in 10,000 of occurring before permanent closure are referred to as Category 2 event sequences.

event tree. A structurally tree-like diagram that is useful in representing sequences of events and their possible outcomes. Each node, or branching point, represents an event, and each branch from that node represents one of its possible outcomes. Each possible pathway along the tree, from beginning to end of a given line of branching, represents a specific scenario.

expected behavior. (1) The mean value of the probability distribution describing that behavior. (2) The nominal behavior of the repository system and the geologic barrier in the absence of disruptive events.

expected value. A variable's mean, or average, outcome. The weighted average of the number of possible outcomes, with each outcome being weighted by its probability of occurrence. The mean of a probability distribution of a random variable that one would expect to find in a very large, random sample. The sum of the possible values, each weighted by its probability—the center of the random variable's histogram (frequency distribution).

Exploratory Studies Facility. An underground laboratory at Yucca Mountain used for performing site characterization studies. The facility includes a 7.9-km (4.9-mi) main loop (tunnel), the 2.8-km (1.7-mi) Enhanced Characterization of the Repository Block (ECRB) Cross-Drift, and a number of alcoves used for site characterization tests such as the Drift Scale Test.

exposure pathway. The course a chemical or physical agent takes from the source to the exposed organism; describes a unique mechanism by which an individual or population can become exposed to chemical or physical agents at or originating from a release site.

exsolve. To separate or precipitate from a solid crystalline phase.

extrusive event. An igneous (volcanic) event occurring at the surface, i.e., a volcanic eruption. In repository performance analyses, molten material is assumed to intersect waste packages in the repository and cause a release of radionuclides. *Compare* intrusive event.

fault. (1) A fracture in rock along which movement of one side relative to the other has occurred. (2) A fracture or a fracture zone in crustal rocks along which there has been movement of the fracture's two sides relative to one another, so that what were once parts of one continuous rock stratum or vein are now separated.

fault displacement. Rupture along the main plane (or planes) of crustal weakness such that the two sides of a fault move relative to one another.

fault source. Fractures in the earth's crust with characteristics that indicate past movement of one side of the fracture relative to the other side.

FEHM. A finite element heat and mass transport computer software code that is used for saturated zone flow and transport and unsaturated zone transport calculations.

film flow. Movement of water as a thin film along a surface.

fissile material. Material capable of undergoing fission with neutrons of any energy, including thermal, or slow, neutrons. The three primary materials in this category are uranium-233, uranium-235, and plutonium-239.

fission. The splitting of a nucleus into at least two other nuclei resulting in the release of two or three neutrons and a relatively large amount of energy.

fission product. Any nuclide, either radioactive or stable, that arises from fission, including both the primary fission fragments and their radioactive decay products. *Also* daughter product, decay product.

FLAC. A specialized computer code developed to solve soil and rock mechanics problems. It is used in conjunction with TOUGH to form a thermal-hydrologic-mechanical code to analyze those coupled effects.

flow field. A fluid's distribution in and through an area, including its velocity and density, as a function of position and time.

flux. The rate of transfer of fluid, particles, or energy passing through a unit area per unit time.

footwall. The rock beneath a fault, bedded deposit, vein, or mine working.

fracture. A break in rock caused by mechanical stresses. A fracture along which there has been displacement of the sides relative to one another is called a fault. A fracture along which no appreciable movement has occurred is called a joint.

fracture continuum. A continuum that represents fluid flow and transport through numerous individual fractures by approximating them as continuous flow and transport fields.

fracture permeability. The capacity of a rock to transmit fluid through fractures in the rock.

frequency distribution. Data grouped into classes (or ranges of values within the overall set of values, such as 1 to 5, 5 to 10, 10 to 20, etc.), with the classes listed in a table (or other format) showing the number of data points that occur in each class.

fuel assembly. A number of fuel rods held together by plates and separated by spacers, used in a reactor. This assembly is sometimes called a fuel bundle.

fuel blending. The process of loading low heat output waste with high heat output waste in a waste package to balance its total heat output. This process applies only to commercial spent nuclear fuel.

fuel blending inventory. The reserve of commercial spent nuclear fuel that will be inventoried in pools in the Waste Handling Building Annex. The spent nuclear fuel will be inventoried in the pools until selected, according to heat output, for fuel blending.

fuel matrix. The physical form and composition of the substance that holds the fissile material.

fugacity. A parameter that measures the chemical potential of a real gas in the same way that partial pressure measures the free energy of an ideal gas.

galvanic corrosion. Electrochemical corrosion caused by the flow of electricity that occurs when two dissimilar metals with differing electrical potentials are near each other in the presence of a conductor such as water with solutes in it.

gamma radiation. Electromagnetic radiation emitted during the radioactive decay process. The gamma ray is the most penetrating wave of radiant nuclear energy. It does not contain particles and can be stopped by dense materials such as concrete or lead.

gamma radiolysis. The breakdown of molecules through exposure to gamma radiation.

gantry. A hoisting machine that slides along a fixed platform or track, either raised or at ground level.

GENII-S. A quasi-stochastic computer software code used to evaluate the dose from the migration of radionuclides in the biosphere that may affect humans through ingestion, inhalation, and direct radiation. It is used to develop biosphere dose conversion factors.

geologic repository. A system for the disposal of radioactive waste in excavated geologic media. A geologic repository includes the engineered barrier system and the portion of the geologic setting that provides isolation of the radioactive waste.

geologic repository operations area. A high-level radioactive waste facility that is part of a geologic repository, including both surface and subsurface areas, where waste handling activities are conducted.

glacial transition. One of the climate states of the future climate model for Yucca Mountain; others examples include modern and monsoon climates.

GoldSim. A software code used as a probabilistic shell for the TSPA component models that, in combination, simulate potential long-term behavior of the repository.

ground control. Support of rock in the subsurface of the repository (e.g., rock bolts and steel sets).

ground support. The system (rock bolt with wire mesh, steel cast, etc.) used to line the main and emplacement drifts to minimize rock or earth falling into the drifts.

groundwater. Water contained in pores or fractures in either the unsaturated or saturated zones below ground level.

groundwater flux. The rate of groundwater flow through a unit area of the aquifer.

half-life. The time in which half the atoms of a radioactive substance decay to another nuclear form. Half-lives range from millionths of a second to millions of years, depending on the stability of the nuclei.

heat exchanger. A device that transfers heat from one medium or system to another; for example, heat from hot fluid contained in a radiator dissipates when the metal walls of the device come in contact with cold air.

heat pipe. A zone characterized by a continuous process of boiling, vapor transport, condensation, and migration of water back to the heat source.

heterogeneous. Being composed of parts or elements of different kinds, such as a mixture of liquid–vapor or liquid–vapor–solid. A condition in which the value of a parameter, such as porosity of rock, varies over space and time.

high-level radioactive waste. The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing; and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and other highly radioactive material that the U.S. Nuclear Regulatory Commission determines by rule requires permanent isolation.

high-level radioactive waste glass. The waste form of high-level radioactive waste in which the radioactive waste is mixed with borosilicate glass.

Holocene. The most recent epoch of geologic time that extends from the end of the Pleistocene to the present, or approximately the past 10,000 years; also the series of rocks and deposits formed during that time.

host rock. (1) The rock unit in which the potential repository would be located. For a repository at Yucca Mountain, the host rock would be the middle portion of the of the Topopah Spring Tuff of the Paintbrush Group. (2) The geologic medium in which the waste is emplaced.

human intrusion. Breaching of any portion of the Yucca Mountain disposal system within the repository footprint by any human activity.

human intrusion scenario. A disruptive event assessed in a separate TSPA according to specific characteristics defined by the U.S. Nuclear Regulatory Commission in 10 CFR 63.113(d). According to the U.S. Nuclear Regulatory Commission rule at 10 CFR 63.322, the human intrusion scenario assumes that a drill penetrates the repository and a waste package during exploratory drilling for groundwater resources. The U.S. Nuclear Regulatory Commission rule requires a determination of the earliest time after disposal that the waste package would degrade sufficiently that a human intrusion could occur without recognition by the drillers (10 CFR 63.321).

hydraulic conductivity. A measure of the ability to transmit water through a permeable medium. A number that describes the rate at which water can move through a permeable medium. The hydraulic conductivity depends on the size and arrangement of water-transmitting openings such as pores and fractures, the dynamic characteristics of the water such as density and viscosity, and the driving force.

hydraulic gradient. The difference in the height of water levels with respect to the distance between two locations.

hydraulic head. The pressure in the liquid expressed as equivalent height (head) of the water column.

hydrogen embrittlement. A process resulting in a decrease of fracture toughness or ductility of a metal due to the presence of hydrogen.

hydrogen-induced cracking. Occurs when atomic hydrogen generated at the surface of a metal migrates into the metal and forms hydride phases with the metal components, causing the metal to be more brittle. Hydride phases cause the metal to be more susceptible to cracking, and, thus, to localized corrosion. *Also* hydride cracking.

hydrogeologic nomenclature. A stratigraphic nomenclature system used for the classification of rock at Yucca Mountain based on the hydrogeologic properties that govern the rock's capacity to hold, transmit, and deliver water. *Compare* lithostratigraphic nomenclature *and* thermomechanical nomenclature.

hydrologic properties. Those properties of a rock that govern the capacity to hold, transmit, and deliver water, such as porosity, effective porosity, specific retention, permeability, and the directions of maximum and minimum permeabilities.

hydrostatic pressure. The pressure, measured at a point in a fluid, due solely to the weight of the fluid in the column above the point.

hygroscopic salt. Of a substance, absorbing or attracting moisture from the air; having an affinity for moisture.

hypogene. Of or pertaining to a substance formed by ascending solutions within the earth, e.g., ore or mineral deposits. Occurring or forming within or below the Earth's crust.

hysteresis. The dependence of system behavior on its history; in particular, failure to return to initial conditions following retraction of stimulus.

igneous. (1) A type of rock that has formed from a molten, or partially molten, material. (2) A type of activity related to the formation and movement of molten rock either in subsurface (plutonic) or on the surface (volcanic).

imbibition. The absorption of a fluid, usually water, by porous rock (or other porous material) under the force of capillary attraction and without pressure.

in situ. In its natural position or place. The phrase distinguishes in-place experiments, conducted in the field or underground facility.

indurated. Of a rock, characterized by a solid, hard structure, hardened by pressure, heat, or cementation.

INFIL. A numerical software code used to analyze infiltration of precipitation.

initiating event. A natural or human-induced event that causes an event sequence.

infiltration. The process of water entering the soil at the ground surface and the ensuing movement downward. Infiltration becomes percolation when water has moved below the depth at which it can be removed (to return to the atmosphere) by evaporation or evapotranspiration.

intrablock faults. Faults (i.e., rock fractures having experienced movement along their plane) found within a block of rock; in this case, within the repository block, which is located within the Topopah Spring Tuff formation.

intrusive event. An igneous event occurring at the underground. In repository performance analyses, molten material is assumed to intersect waste packages in the repository and cause a release of radionuclides. *Compare* extrusive event.

inverse modeling. The model calibration process by which values of important model parameters are estimated and optimized to produce the best fit of the model output to the observed data.

invert. (1) The floor of a drift. (2) The structure constructed in a drift to provide the floor of that drift. In an emplacement drift, ballast in the invert serves as a barrier to migration of radionuclides that might escape from breached waste packages.

ion-exchange resins. (1) Any of a number of (usually organic) materials that are capable of exchanging the included ions with ions in a surrounding solution; used for deionizing water or for chromatography of organic molecules. (2) A synthetic resin that can combine or exchange ions with a solution; such a resin produces the exchange of sodium for calcium ions in the softening of hard water.

ionic strength. A measure of the level of electrical force in an electrolytic solution.

ionizing radiation. (1) Alpha particles, beta particles, gamma rays, x-rays, neutrons, high-speed electrons, high-speed protons, and other particles capable of producing ions. (2) Any radiation capable of displacing electrons from an atom or molecule, thereby producing ions.

irreducible uncertainty. Uncertainty that cannot be further reduced, given current best knowledge, expert insights, and calculational abilities.

irreversible colloid. A colloid with permanently attached radionuclides.

isothermal seepage. The flow of water into a drift under ambient conditions with constant temperature.

isotope. (1) Atoms of a chemical element with the same atomic number. (2) One of two or more atomic nuclei with the same number of protons (i.e., the same atomic number) but with a different number of neutrons (i.e., a different atomic weight). For example, uranium-235 and uranium-238 are both isotopes of uranium.

isotropy. The condition wherein all significant physical properties are equal when measured in any direction or along any axes. *See* anisotropy.

J-13 water. Groundwater taken from Well J-13 or water made in a laboratory that has the same chemical composition. The chemical composition of this water is used as the standard for Yucca Mountain ambient groundwater composition for modeling and testing purposes.

joint. A fracture in rock, usually more or less vertical to bedding, along which no appreciable movement has occurred.

joint set. In a rock mass, a group of parallel joints (fractures without displacement).

juvenile failure. (1) Premature failure of a waste package because of material imperfections or damage by rockfall during emplacement. (2) In modeling, a breach in the waste package artificially set to occur early in order to provide insight into system performance. This term is distinguished from mechanistically possible early failures.

key technical issues. Issues important for assessing the long-term safety of a potential Yucca Mountain repository, as defined by the U.S. Nuclear Regulatory Commission (NRC). The issues are (a) Support Revision of the U.S. Environmental Protection Agency Standard/NRC Rule Making; (b) Total System Performance Assessment and Technical Integration; (c) Igneous Activity; (d) Unsaturated and Saturated Flow Under Isothermal Conditions; (e) Thermal Effects on Flow; (f) Container Life and Source Term;

(g) Structural Deformation and Seismicity; (h) Evolution of Near-Field Environment; (i) Radionuclide Transport; (j) Repository Design and Thermal Mechanical Effects.

Large Block Test. A prototype test of thermal-mechanical processes at Yucca Mountain in the middle nonlithophysal zone of the Topopah Spring unit. The test was to develop testing approaches for thermal-hydrologic and other coupled processes.

laser peening. Process of applying laser-generated compressive stresses to the welds of the waste package lids. Hardens the weld and reduces remaining tensile stresses.

Latin hypercube sampling. A sampling technique that divides the cumulative distribution function into intervals of equal probability and then samples from each interval.

Latin square. A method for ordering the observations of an experiment in an $n \times n$ square array of n different symbols, where each symbol appears once in each row and once in each column.

license application. An application to the Nuclear Regulatory Commission (NRC) to construct a geologic repository operations area for the disposal of spent nuclear fuel and high-level radioactive waste. The application would be considered by the U.S. Nuclear Regulatory Commission in any decision whether to grant the U.S. Department of Energy authorization to begin constructing a geologic repository operations area.

line loading. Placing the waste packages at very close distances end-to-end to achieve a more uniform thermal profile along the length of the emplacement drifts. *Compare* point loading; *see* thermal loading.

linear regression. A regression where the relationship between the (conditional) mean of a random variable and one or more independent variables can be expressed by the mathematical equation that describes a line. A relationship between two variables such that the dependence of one variable on the other can be described by (the equation of) a straight line.

linear stepwise regression. An analysis designed to determine variables that have the greatest influence on an output value (e.g., peak dose rate) when there are many variables whose input values go into the calculation. In simple terms, a linear regression is performed for a line in a multidimensional space, and the correlation of the values of different variables to the line are examined by performing the calculation multiple times and varying the value of one variable at a time while holding the others constant. This is a stepwise process in which one variable at a time is examined to determine the impact of its influence on the final outcome (peak dose rate, for instance).

lithologic. Pertaining to rock features such as color, texture, mineral content, and weathering characteristics.

lithophysae. Small, bubble-like holes in the rock caused by volcanic gases trapped in the rock matrix as the ash-flow tuff cooled.

lithophysal. Pertaining to tuff units with lithophysae, voids having concentric shells of finely crystalline alkali feldspar, quartz, and other materials that were formed by entrapped gas that later escaped.

lithostratigraphic nomenclature. A stratigraphic nomenclature system used for the classification of rock at Yucca Mountain based on primary geologic processes (e.g., the depositional character and assemblage of the rock) and secondary geologic processes (e.g., the degree of welding, devitrification, and vapor-phase crystallization). *Compare* hydrogeologic nomenclature *and* thermomechanical nomenclature.

loading curve. A function of the average burnup versus initial enrichment of a fuel assembly, which provides the necessary information on whether a fuel assembly can be loaded, unaltered, into a standard waste package without concern for criticality. *See* burnup.

localized corrosion. A type of corrosion induced by local variations in electrochemical potential on a microscale over small regions. Variations in electrochemical potential may be caused by localized irregularities in the structure and composition of usually protective passive films on metal surfaces and in the electrolyte composition of the solution that contacts the metal.

log normal distribution. A distribution of a random variable *x* such that the natural logarithm of *x* is normally distributed.

logic tree methodology. An analytical approach that involves sequencing the studies in an analysis and then addressing certain attributes in each study

longitudinal dispersion. Dispersion of a solute moving in groundwater in the same direction as the groundwater flow path.

long-term-average climate. The conditions used to represent climate changes through time. Representative of the expected typical climate conditions at Yucca Mountain, with precipitation twice that of the present-day climate.

lookup table. A multidimensional table containing columns of data representing relationships between parameters in the table. A lookup table is a convenient way to represent and implement functional relationships between parameters considered in the model.

low-level radioactive waste. Radioactive waste producing small quantities of ionizing radiation and that is not classified as high-level radioactive waste, transuranic waste, or byproduct tailings containing uranium or thorium from processed ore. Usually generated by hospitals, research laboratories, and certain industries.

main drift. (1) One of four main access tunnels in the potential repository design. *Also* access main. (2) The main north–south trending drift segment in the current Exploratory Studies Facility.

mass spectrometry. A technique for identifying chemical structures in a material. The technique involves sending a beam of ions through a combination of electric and magnetic fields, which deflects ions according to their masses, indicating the atoms and molecules.

matrix. Rock mass between explicitly considered fractures.

matrix diffusion. The process by which molecular or ionic solutes, such as radionuclides in groundwater, move from areas of higher concentration to areas of lower concentration. This movement is through the pore spaces of the rock material as opposed to movement through the fractures.

matrix permeability. The capacity of the matrix to transmit fluid.

matrix porosity. In the solid, but porous, portion of rock, the ratio of openings, or voids, to the total volume of the matrix expressed as a decimal fraction or as a percentage.

mean. For a statistical data set, the sum of the values divided by the number of items in the set. The arithmetic average.

mechanical dispersion. As a process for transport, a type of dispersion by means of physical forces.

median. A value such that half of the observations are less than that value and half are greater than the value.

meniscal flow. Flow of a curved-surface fluid, the curved free surface being due to surface tension and the shape of the solid the fluid flows over.

meteoric water. Groundwater that originates in the atmosphere and percolates to the saturated zone.

metric ton heavy metal (MTHM). A metric ton is a unit of mass equal to 1,000 kg (2,205 lb). Heavy metals are those with atomic masses greater than 230. Examples include thorium, uranium, plutonium, and neptunium. The term usually pertains to heavy metals in spent nuclear fuel and high-level radioactive waste. In this document, MTHM is equal to MTU (metric tons of uranium).

microbially influenced corrosion. Corrosion of the waste package that is induced by the activity of microbes.

microsphere. Spheres of carboxylate-modified latex of varied diameters between 280 to 640 nm (0.000011 to 0.000025 in.), used as colloid tracers in the C-Wells testing.

millirem (mrem). A millirem is one one-thousandth of a rem, which is the unit of equivalent dose. Equivalent dose is a measure of the effect that radiation has on humans. The equivalent dose takes into account the type of radiation and the absorbed dose. *See* rem.

MING. Model code used to estimate impact of microbes on the near-field environment geochemistry.

Miocene Epoch. The fourth of the five geologic epochs of the Tertiary Period, extending from the end of the Oligocene Epoch to the beginning of the Pliocene Epoch.

mitigation. (1) Avoiding an impact by not taking a certain action or parts of an action. (2) Minimizing impacts by limiting the degree or magnitude of the action and its implementation. (3) Rectifying an impact by repairing, rehabilitating, or restoring the affected environment. (4) Reducing or eliminating an impact over time by preservation and maintenance operations during the life of the action. (5) Compensating for an impact by replacing or providing substitute resources or environments.

mixed waste. Waste containing both radioactive hazardous substances and nonradioactive hazardous substances, regardless of whether these types of substances are combined chemically or mixed together.

model. (1) A conceptual description and the associated mathematical representation of a system, subsystem, component, or condition that is used to predict changes from a baseline state as a function of internal and/or external stimuli and as a function of time and space. (2) A depiction of a system, phenomenon, or process including any hypotheses required to describe the system or explain the phenomenon or process.

model validation. A process used to establish confidence that a conceptual model represented in a mathematical model by software or by other analytical means adequately represents the phenomenon, process, or system under consideration.

moderator. Material that contains nuclei that cause energetic neutrons to slow down. In general, the lighter the element, the better it works as a moderator. Hydrogen, the lightest element, is a very efficient moderator; since water contains hydrogen nuclei, it, too, is a very effective moderator.

modern climate. One of the climate states of the future climate model at Yucca Mountain; others include monsoon and glacial transition. Consists of two active components, the tropical and polar air masses, and a more passive component, the westerlies.

molal. Of a solution, containing one mole of solute per one kilogram of solvent. *See* mole.

mole. The fundamental unit used to measure the amount of a substance. Avogadro's number of particles (6.023×10^{23}).

molecular diffusion. The transfer of mass in a fluid by random molecular motion.

monitored geologic repository. A system, requiring licensing by U.S. Nuclear Regulatory Commission, intended or used for the permanent underground disposal of spent nuclear fuel and high-level radioactive waste. A geologic repository includes (a) the geologic repository operations area, and (b) the geologic setting within the controlled area that provides isolation of the radioactive waste.

monsoon climate. One of the climate states of the future climate model at Yucca Mountain; others include modern and glacial transition.

Monte Carlo simulation. An analytical method that uses random sampling of parameter values available for input into numerical models as a means of approximating the uncertainty in the process being modeled. A Monte Carlo simulation comprises many individual runs of the complete calculation using different values for the parameters of interest as sampled from a probability distribution. A different final outcome for each individual calculation and each individual run of the calculation is called a realization. Each realization is equally likely to occur in the Monte Carlo process.

mountain scale. (1) Similar to far-field for processes that are related to the area of the geosphere and biosphere far enough away from the repository that, when numerically modeled, show that releases from the repository are represented as a homogeneous, single source term. The effects of individual, small-scale components such as individual waste packages are not modeled because they are considerably smaller than the scale of the model. (2) A scale of hundreds of meters, or even kilometers, as opposed to tens of meters.

muck. Material excavated from a mine or geologic repository.

MULTIFLO. A computer software code used by the U.S. Nuclear Regulatory Commission to simulate the flow of groundwater and heat in unsaturated porous and fractured media. The code MULTIFLO is similar to TOUGH2.

natural barriers. The physical components of the geologic environment that individually and collectively act to limit the movement of water or radionuclides.

near field. The area and conditions within the repository including the drifts and waste packages and the rock immediately surrounding the drifts. The region around the repository where the natural hydrogeologic system has been significantly impacted by the excavation of the repository and the emplacement of waste.

net infiltration. The water that has infiltrated down from the soil zone or exposed rock surface to a depth below which it cannot be removed by evapotranspiration. Net infiltration is the total infiltration at the surface minus water lost to evaporation and plant transpiration.

neutron absorber. A material (such as boron or gadolinium) that absorbs neutrons. Used in nuclear reactors, transportation casks, and waste packages to control neutron activity.

neutron logging. The analysis of the water content of soil and rocks in a borehole by means of neutron bombardment and the measurement of the reflected radiation.

nominal scenario or nominal case. The performance assessment case, or conceptual model, representing the expected conditions of the disposal system as perturbed only by the presence of the repository, in the absence of disruptive events.

nonstandard fuel. For the purpose of this document, nonstandard fuel is defined as commercial spent nuclear fuel assemblies, single-assembly canisters, and packages that satisfy the following criteria: (1) the maximum nominal physical dimensions are larger than those of intact light-water reactor standard fuel assemblies; (2) assemblies, canisters, or packages that require special handling other than with standard fuel assembly transfer equipment; (3) non-power reactor fuel assemblies that are not packaged in sealed canisters; (4) consolidated fuel rods that require reconfiguration or repackaging; (5) failed fuel assemblies with damaged cladding, structural deformations, or high levels of contamination resulting from released radioactive particulates and activated corrosion products (crud).

NUFT. A nonisothermal unsaturated zone flow and transport software code used for simulation of three-dimensional flow of groundwater, heat, and contaminant transport. It is used for drift-scale thermal-hydrologic calculations.

numerical model. An approximate representation of a mathematical model that is constructed using a numerical description method, such as finite volumes, finite differences, or finite elements. A numerical model is typically represented by a series of program statements that are executed on a computer.

observation drift. A drift near an emplacement drift, from which conditions in the emplacement drifts can be observed without adverse effects from radiation or temperature.

occupational dose. The radiation dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from sources of radiation, whether in the possession of the licensee or other person. Occupational dose does not include, for example, dose received from background radiation, from any medical administration the individual has received, from voluntary participation in medical research programs, or as a member of the public.

one-off sensitivity analysis. A method used to examine the effects of each component model or parameter on overall system performance. These analyses are conducted by fixing one important parameter of a particular component model at either its expected value, the median (or 50th percentile value), or at a specified extreme value (the 5th or 95th percentile value). These analyses are used to display the effect of the change on other measures of system or subsystem performance, as well as the effect on the variance of the projected dose history. By fixing a particular parameter that significantly affected the spread of the overall system performance results, it was possible to directly examine the significance of that parameter on the performance outcome.

order of magnitude. A range of numbers extending from some value to 10 times that value.

ORIGEN. Family of software codes that simulate radionuclide decay and estimate buildup and depletion of isotopes in reactor fuel.

overpack. A secondary container used to hold or contain one or more smaller canisters.

oxidation. (1) A chemical reaction, such as the rusting of iron, that increases the oxygen content of a substance. (2) A reaction in which the valence of an element or compound is increased as a result of losing electrons.

paleoclimate. The climate of a past interval of geologic time.

Paleozoic. (1) A geologic era extending from the end of the Precambrian to the beginning of the Mesozoic, dating from about 600 to 230 million years ago. (2) The rock strata formed during this era.

patch. For corrosion modeling, one of two geometries for an opening in a waste package layer created by corrosion (the other geometry is a pit). A patch is generally wider than it is deep.

pedogenic. Of or pertaining to the formation of soil; soil-forming.

perched water. Groundwater of limited lateral extent separated from an underlying body of groundwater by an unsaturated zone.

percolating water. Water passing through a porous substance. In rock or soil it is the movement of water through the interstices and pores under hydrostatic pressure and the influence of gravity. The downward or lateral flow of water that becomes net infiltration in the unsaturated zone.

percolation. The downward or lateral flow of water that becomes net infiltration in the unsaturated zone.

percolation flux. (1) Volumetric percolation rate per unit area. The flux anywhere below the root zone of plants and is no longer susceptible to removal back into the atmosphere by evapotranspiration. (2) Volume of water moving downward or laterally through the unsaturated zone in a given period.

performance assessment. An analysis that forecasts the behavior of a system or system component under a given set of constant and/or transient conditions. Performance assessment includes estimates of the effects of uncertainties in data and modeling. *See* total system performance assessment.

permeability. In general terms, the capacity of a medium (like rock, sediment, or soil) to transmit liquid or gas.

person-rem. A unit used to measure the radiation exposure to an entire group and to compare the effects of different amounts of radiation on groups of people; it is the product of the average dose equivalent (in rem) to a given organ or tissue multiplied by the number of persons in the population of interest.

phase stability. A measure of the ability of matter to remain in a given phase.

phreatic. Of, pertaining to, or deriving from groundwater

phreatophytic plant. A very deep-rooted plant that obtains its water from perched water or from the saturated zone.

pit. For corrosion modeling, one of two geometries for an opening in a waste package layer created by corrosion (the other geometry is a patch). A pit is generally deeper than it is wide.

Pitzer approach. An analytic technique using the Pitzer equation, which estimates the amount of heat produced by the vaporization of organic and simple inorganic compounds.

playa. A nearly level area at the bottom of an undrained desert basin, sometimes temporarily covered with water.

Pliocene. The last of the five geologic epochs of the Tertiary Period, extending from the end of the Miocene to the beginning of the Pleistocene, and the rocks formed during that time.

plume. A measurable discharge of a contaminant, such as radionuclides, from a point of origin. The contaminants are usually moving in groundwater, and the plume may be defined by concentration gradients.

plume of contamination. That volume of groundwater in the predominant direction of groundwater flow that contains radioactive contamination from releases from the Yucca Mountain repository. It does not include releases from any other potential sources on or near the Nevada Test Site.

pluvial. In climatology: relating to former periods of abundant rains, especially in reference to glacial periods. In geology: Said of a geologic episode, change, process, deposit, or feature caused by the action or effects of rain.

point loading. An emplacement drift design in which waste packages are spaced away from each other along the drift. *Compare* line loading; *see* thermal loading.

point of compliance. The place where the DOE must project the amount of radionuclides in the groundwater as defined by proposed 40 CFR 197.

pore water. The water and any material it is carrying that exist in the pore spaces of the rock matrix. *Also* pore fluid.

porosity. The ratio of openings, or voids, to the total volume of soil or rock, expressed as a decimal fraction or as a percentage.

postclosure. The period of time after closure of the geologic repository.

potentiometric. Pertaining to the distribution of groundwater level.

precipitate. A solid particle that has separated from a liquid as a result of physical or chemical changes.

precipitation. (1) The process of substance coming out of solution by the action of gravity or by a chemical reaction. (2) Any form of water particles, such as frozen water in snow or ice crystals, or liquid water in raindrops or drizzle, that falls from clouds in the atmosphere and reaches the earth's surface. (3) An amount of water that has fallen at a given point over a specified period of time, measured by a rain gauge.

preclosure. The period of time before and during closure of the Yucca Mountain disposal system.

preclosure safety evaluation. A preliminary assessment of the adequacy of repository support facilities to prevent or mitigate the effects of postulated initiating event sequences and their consequences and the site, structures, systems, components, equipment, and operator actions that would be relied on for safety.

pressurized water reactor. A type of nuclear power reactor that uses uranium fuel elements cooled and moderated by water under high pressure to keep the water from boiling. The water boiled to generate steam is in an external heat exchanger rather than in the reactor vessel body.

probabilistic analyses. Analyses in which uncertainty in processes and events is represented through probability distributions for the parameters of those processes and events.

probabilistic risk assessment. (1) A systematic process of identifying and quantifying the consequences of scenarios that could cause a release of radioactive materials to the environment. (2) Using predictable behavior to define the performance of natural, geologic, human, and engineered systems for thousands of years into the future using probability distributions.

probability-density function. A frequency distribution such that the bars of a histogram that would represent it are so narrow that their tops would form a smooth curve if connected by a line. This type of distribution can be made if the number of observations of the value of a continuous random variable increases indefinitely, and the width of the range represented by each class (class interval) becomes smaller and smaller. The area under the density function curve between any two points on the curve represents the probability that the value of the random variable will lie between these two values.

process model. A depiction or representation of a process along with any hypotheses required to describe or to explain the process.

pseudocolloid. A colloid from natural or man-made materials, as distinguished from a colloid from insoluble radionuclides (intrinsic colloid) or from altered fragments of spent nuclear fuel or glass waste forms (waste-form colloid). *See* colloid.

pyroclastic. Of or relating to clastic rock material of any size that is formed by volcanic explosion or ejected from a volcanic vent.

Quaternary. The second period of the Cenozoic Era, beginning about 2 million years ago at the end of the Tertiary Period and extending to the present.

rad (radiation absorbed dose). A unit of an absorbed dose of radiation equivalent to 100 ergs per gram

radioactive decay. The process in which one radionuclide spontaneously transforms into one or more different radionuclides called decay products or daughter products.

radioactive waste. High-level radioactive waste and other radioactive materials, including spent nuclear fuel, that are received for emplacement in the geologic repository.

radioactivity. The property possessed by some elements (e.g., uranium) of spontaneously emitting alpha, beta, or gamma rays by the disintegration of atomic nuclei.

Radiologically Controlled Area. An area of the surface repository enclosed by security fences, control gates, lighting, and access detection systems. This area includes the facilities and transportation systems required to receive and ship rail and truck waste shipments, prepare shipping casks for handling, and load waste forms into disposal containers for underground emplacement. It also includes the facility and systems required to treat and package site-generated, low-level radioactive waste for offsite disposal.

radiolysis. The chemical dissociation of molecules caused by exposure to radiation. For example, under certain circumstances, radiation can cause the hydrogen and oxygen molecules in water to separate.

radiolytic corrosion. The process of dissolving or wearing away gradually caused by chemical changes associated with exposure to radiation (i.e., radiolysis).

radionuclide. A radioactive atom with an unstable nucleus that spontaneously decays, emitting ionizing radiation in the process.

raise. From mining terminology, an upward opening, either vertical or inclined, driven in rock from one level to that above it.

reasonably maximally exposed individual (RMEI). Under the U.S. Nuclear Regulatory Commission rule, a hypothetical person who meets the following criteria: (1) lives in the accessible environment above the highest concentration of radionuclides in the plume of contamination; (2) has a diet and living style representative of the people who now reside in the Town of Amargosa Valley, Nevada (the DOE must use projections based upon surveys of the people residing in the Town of Amargosa Valley, Nevada, to determine their current diets and living styles and use the mean values of these factors in the assessments conducted for 10 CFR 63.311 and 63.321); (3) uses well water with average concentrations of radionuclides based on an annual water demand of 3,000 acre-ft; (4) drinks 2 L (0.53 gal) of water per day from wells drilled into the groundwater at the location specified in the regulations; and (5) is an adult with metabolic and physiological considerations consistent with present knowledge of adults.

recharge. The movement of water from an unsaturated zone to the saturated zone.

reference biosphere. The description of the environment inhabited by the reasonably maximally exposed individual. The reference biosphere comprises the set of specific biotic and abiotic characteristics of the environment, including but not necessarily limited to climate, topography, soils, flora, fauna, and human activities.

reflux water. Water that is vaporized near waste packages, migrates to cooler areas, condenses, and then flows back toward the waste packages.

regression analysis. The analysis of a paired dependent variable and the independent variable upon which it depends to quantify the relationship.

rem (roentgen equivalent man). The unit of a dose equivalent from ionizing radiation to the human body. It is used to measure the amount of radiation to which a person has been exposed.

repository block. The portion of rock in Yucca Mountain that would house the repository if the site is found suitable.

repository footprint. The outline of the outermost locations of where the waste is emplaced in the Yucca Mountain geologic repository.

retardation. Slowing of radionuclide movement in groundwater by mechanisms that include sorption of radionuclides, diffusion into rock matrix pores and microfractures, and trapping of large colloidal molecules or particles in small pore spaces or dead ends of microfractures.

reversible colloid. A colloid to which radionuclides are reversibly bound.

revised supplemental TSPA model. The model used in supplemental calculations of total system performance assessment as documented in *Total System Performance Assessment—Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain—Input to Final Environmental Impact Statement and Site Suitability Evaluation.* The model is a modification of the supplemental TSPA model that conforms to the requirements of the final U.S. Environmental Protection Agency rule at 40 CFR Part 197. This model was also used in sensitivity analyses in *Total System Performance Assessment Sensitivity Analyses for Final Nuclear Regulatory Commission Regulations* to address provisions of the final Nuclear Regulatory Commission rule at 10 CFR Part 63. *See* total system performance assessment. *Compare* TSPA-SR model *and* supplemental TSPA model.

rhyolite. A volcanic rock type with a chemical composition similar to granite.

risk. The probability that an undesirable event will occur multiplied by the consequences of the undesirable event.

risk assessment. An evaluation of risks associated with a potential system or action. This assessment focuses on potential impacts on human health or the environment.

roadheader. An underground excavating machine that uses either a transverse or in-line cutter head for excavations.

runoff. Water from rain and snow that flows over land to streams.

run-on. The volume or depth of the routed surface-water flow.

safety case. The set of data and analyses that, collectively, are intended to provide the reasonable assurance that a successful license application would require.

safety margin. The difference between expected performance and the regulatory limit for that performance. *See* design margin.

saturated zone. The region below the water table where rock pores and fractures are completely saturated with water.

scarp. An escarpment, cliff, or steep slope of some extent that is produced by faulting or by differential erosion.

scenario. A well-defined, connected sequence of features, events and processes that can be thought of as an outline of a possible future condition of the repository system. Scenarios can be undisturbed, in which case the performance would be expected, or nominal, behavior for the system. The scenario can also be disturbed if altered by disruptive events, such as human intrusion or natural phenomena such as volcanism.

secular equilibrium. A condition in which a daughter radionuclide has reached a steady-state amount, in terms of radioactive activity, with respect to the amount of its parent radionuclide.

sedimentation. A geologic process in which particles accumulate in water or air and settle in layers of rock.

seepage. (1) The inflow of groundwater moving in fractures or pore spaces of permeable rock to an open space in the rock; the amount of percolation flux that enters the drift in a given time period. (2) Flow of liquid water into an underground opening such as a waste emplacement drift or exploratory tunnel. Does not include water vapor movement into openings or condensation of water vapor within openings.

seepage threshold. A critical percolation flux below which seepage into the openings is unlikely to occur.

seismicity. A seismic event or activity such as an earthquake or vibratory motion.

semiarid. Of a climate: having precipitation, only sufficient for growth of sparse vegetation; a region in which the annual precipitation is about 250 to 500 mm (10 to 20 in.).

sensitivity study. An analytic or numerical technique for examining the effects of varying specified parameters in a computer model. Shows the effects that changes in various parameters have on model outcomes and illustrates which parameters have a greater impact on the predicted behavior of the system being modeled. Also called *sensitivity analysis* because it shows the sensitivity of the consequences (e.g., radionuclide release) to uncertain parameters (e.g., the infiltration rate that results from precipitation).

shotcrete. Cementitious material sprayed onto a surface at high pressure.

site characterization. Activities, whether in the laboratory or in the field, undertaken to establish the geologic and hydrologic conditions and the ranges of the parameters of a candidate site relevant to the location of a repository. These activities include borings, surface excavations, subsurface excavations and borings, and in situ testing needed to evaluate the suitability of a candidate site for the location of a repository but do not include preliminary borings and geophysical testing needed to assess whether site characterization should be undertaken.

sorption. The binding, on a microscopic scale, of one substance to another. A term that includes both adsorption and absorption. The sorption of dissolved radionuclides onto aquifer solids or waste package materials by means of close-range chemical or physical forces is an important process modeled in this study. Sorption is a function of the chemistry of the radioisotopes, the fluid in which they are carried, and the mineral material they encounter along the flow path.

sorption coefficient (K_d). A factor to calculate sorption of one substance to another (e.g., sorption of a radionuclide to a colloid or sorption of a radionuclide to the rock).

source term. Types and amounts of radionuclides that are the source of a potential release of radioactivity from the repository.

spent nuclear fuel. Fuel and the associated hardware withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. Spent fuel that has been burned (irradiated) in a reactor to the extent that it no longer makes an efficient contribution to a nuclear chain reaction. This fuel is more radioactive than it was before irradiation, and it is thermally hot.

splay. A branch of a fault or fault zone.

split. The two halves of an emplacement, performance confirmation, or reserve drift, separated by an exhaust raise.

staging area. An area in the Waste Handling Building or a part of the waste-handling process in which spent nuclear fuel or high-level radioactive waste is retained for future loading in a disposal container.

standard deviation. (1) For a set of observations or a frequency distribution, the square root of the average of the squared deviations from the mean divided by n-1 (where n is the sample size). (2) The square root of the variance.

steady-state criticality. An self-sustained nuclear chain reaction where the reaction remains constant, with the effective neutron multiplication factor equal to one. *See* effective neutron multiplication factor.

steady-state modeling. Modeling a system under the assumption that the variables are not changing with time. For example, flow fields can be simulated at a steady state if the boundary conditions, saturations, and fluxes are not changing with time.

steel set. A steel support used in tunnels, drifts, and shafts.

stochastic. Involving a variable (e.g., temperature, porosity) that may take on values of a specified set with a certain probability. Data from a stochastic process is an ordered set of observations, each of which is one item from a probability distribution.

stochastic model. A model whose outputs are predictable only in a statistical sense. A given set of model inputs produces outputs that are not the same, but follow statistical patterns.

stress corrosion cracking. Preferential corrosion initiation in response to high tensile stresses, requiring the simultaneous action of a corrosion mechanism and sustained tensile stress.

stress intensity. The amount of stress at a given point in a structure. Derived from combined totals of both positive (tension) stress and negative (compression) stress.

stylized human intrusion scenario. A disruptive event assessed in a separate TSPA according to specific characteristics defined by the EPA in 40 CFR 197.26 and the NRC in 10 CFR 63.113(d). According to these regulations, the human intrusion scenario assumes that a drill penetrates the repository and a waste package during exploratory drilling for groundwater resources, one hundred years after final closure of the repository.

subsurface facilities. The repository's underground structures and systems. The surface facilities include the main drifts, exhaust mains, turnouts, emplacement drifts, ventilation shafts, mechanical and structural support systems, underground utilities, waste emplacement and retrieval equipment, and surface-based control systems.

supplemental TSPA model. The model used in supplemental calculations of total system performance assessment as documented in Volume 2 of *FY01 Supplemental Science and Performance Analyses*. The model is a modification of the TSPA-SR model that incorporates new component models and input parameter values for some components. The model was based on specifications of proposed U.S. Environmental Protection Agency and U.S. Nuclear Regulatory Commission regulations. *See* total system performance assessment. *Compare* TSPA-SR model *and* revised supplemental TSPA model.

surface complexation. The process that describes the formation of complex molecules between the solute in the aqueous phase and the reactive groups on the solid surface, under specific chemical conditions.

surface facilities. All permanent facilities within the restricted area constructed in support of site characterization activities and repository construction, operation, and closure activities, including surface structures, utility lines, roads, railroads, and similar facilities, but excluding the underground facility.

system model. The analytical tool to examine the future behavior of the potential repository and its component barriers.

system performance. The complete behavior of a geologic repository system at Yucca Mountain in response to the features, events, and processes that may affect it.

SZ_CONVOLUTE. Software used to calculate saturated zone response curves based upon unsaturated zone radionuclide source terms, generic saturated zone response and expected climate scenarios

tectonic. Pertaining to geologic forms or effects created by deformation of the earth's crust.

Tertiary. The first of two geologic periods of the Cenozoic Era extending from the end of the Mesozoic Era to the beginning of the Quarternary Period, covering a time span from about 65 million to about 2 million years ago.

thermal conduction. The flow of thermal energy through a material. This conduction is affected by the amount of heat energy present, the nature of the heat carrier in the material, and the amount of dissipation.

thermal loading. (1) The spatial density at which waste packages are emplaced within the repository as characterized by the areal power density and the areal mass loading. *See* line loading *and* point loading. (2) The application of heat to a system, usually measured in terms of watt density. The thermal loading for a repository is the watts per acre produced by the radioactive waste in the active disposal area.

thermal stress. Stress caused by temperature changes in a material that is physically restricted and unable to expand or contract accordingly.

thermal-mechanical effects. Changes in the geomechanical properties of the repository host rock produced by heating of the rock associated with the emplacement of radioactive waste in the repository. An example might be decreased rock strength related to increased fracturing caused by heating of the rock.

thermogravimetric analysis. A method of analysis that measures the loss or gain of weight by a substance as the temperature of the substance is raised or lowered at a constant rate.

thermomechanical nomenclature. A stratigraphic nomenclature system used for the classification of rock at Yucca Mountain based on the thermal and mechanical properties of the rock. *Compare* lithostratigraphic nomenclature, hydrogeologic nomenclature, *and* lithostratigraphic nomenclature.

three-dimensional model. A three-dimensional representation of physical conditions and/or processes.

total effective dose equivalent. For purposes of assessing doses to workers, the sum of the deep-dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). For purposes of assessing doses to members of the public (including the reasonably maximally exposed individual), total effective dose equivalent means the sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). *See* annual committed effective dose equivalent.

total system performance assessment (TSPA). A risk assessment that quantitatively estimates how the proposed Yucca Mountain disposal system will perform in the future under the influence of specific features, events, and processes, incorporating uncertainty in the models and data.

TOUGH2. A computer software code used to simulate three-dimensional flow of groundwater and heat in unsaturated or saturated porous and fractured media. It is the basis for the unsaturated zone flow process model.

TOUGHREACT. Thermal-hydrologic-chemical software code used to simulate the water composition on the drip shield and waste package.

tracer. A substance or dye used in hydrologic tests to observe the movement of groundwater and sorbing and nonsorbing chemical species. Also applicable to gas injection tests using gaseous tracers to measure the breakthoughs at observation points.

transmissive fracture. A fracture in rock through which groundwater could flow.

transparency. The ease of understanding the process by which a study was carried out, which assumptions drove the results, how they were determined, and the rigor of the analyses that led to the results. Transparency provides a reader or reviewer with a clear picture of what was done in an analysis, what the outcome was, and why.

transpiration. The process by which water absorbed by plants, usually through the roots, is evaporated into the atmosphere from the plants' surfaces. It is an important process for removal of water that has infiltrated below the zone where it could be removed by evaporation.

transport. A process in which substances carried in groundwater move through the subsurface by means of the physical mechanisms of convection, diffusion, and dispersion and the chemical mechanisms of sorption, leaching, precipitation, dissolution, and complexation. Types of transport include advective, diffusive, and colloidal transport.

transportation cask. A heavily shielded container that meets applicable regulatory requirements used to ship spent nuclear fuel or high-level radioactive waste.

transuranic waste. Waste materials (excluding high-level radioactive waste and certain other waste types) contaminated with alpha-emitting radionuclides that are heavier than uranium with half-lives greater than 20 years and that occur in concentrations greater than 100 nanocuries per gram.

Transuranic waste is primarily a result of treating and fabricating plutonium, as well as from research activities at DOE defense installations.

transverse dispersion. The spreading of a solute in groundwater in directions perpendicular to the direction of the groundwater flow path.

travertine. A finely crystalline, massive deposit of calcium carbonate formed by chemical precipitation from solution in surface and groundwaters or in limestone caves, as stalactites, stalagmites, or dripstone.

TSPA-SR model. The model used in the calculations of total system performance assessment as documented in *Total System Performance Assessment for the Site Recommendation*. The model was based on specifications of proposed U.S. Environmental Protection Agency and U.S. Nuclear Regulatory Commission regulations. *See* total system performance assessment. *Compare* supplemental TSPA model *and* revised supplemental TSPA model.

tufa. A chemical sedimentary rock composed of calcium carbonate formed by evaporation around the mouth of a spring, along a stream, or as a thick concretionary deposit in a lake or along its shore.

tuff. Igneous rock formed from compacted volcanic fragments from pyroclastic (explosively ejected) flows with particles generally smaller than 4 mm (0.16 in.) in diameter. The most abundant type of rock at the Yucca Mountain site. Nonwelded tuff results when volcanic ash cools in the air sufficiently that it doesn't melt together, yet later becomes rock through compression. Welded tuff results when the volcanic ash is hot enough to melt together and is further compressed by the weight of overlying materials.

two-dimensional model. (1) A two-dimensional slice through an entity, such as the earth's crust, usually in the horizontal and vertical directions, on which known features are placed and are used to predict likely features that may exist between points of known data. (2) Mathematically, a model that represents physical conditions or processes; this mathematical model is composed of both horizontal rows and vertical columns of grid cells arrayed in L-shaped configurations only one grid cell thick.

UDEC. Distinct element code used to perform underground opening stability analysis.

uncertainty. A measure of how much a calculated or estimated value that is used as a reasonable guess or prediction may vary from the unknown true value.

underground facility. The underground structure, backfill materials, if any, and openings that penetrate the underground structure.

unsaturated zone. The zone of soil or rock below the ground surface and above the water table.

unzipping. The splitting of the cladding on a fuel rod.

uptake. Intake by and exposure of the receptor to a contaminant.

van Genuchten's capillary-strength parameter. A parameter in a functional relationship between saturation (or water content) and potential (a measure of the suction due to capillary forces). *Also* α-parameter.

variability (statistical). A measure of how a quantity varies over time or space.

vitric tuff. Volcanic rock composed of glassy shards of volcanic ash.

vitrified high-level radioactive waste. A type of processed high-level radioactive waste where the waste is mixed with glass-forming chemicals and put through a melting process. The melted mixture is then put into a canister where it becomes a dry, solid "log" of waste in a glassy matrix.

WAPDEG. A computer software code used to analyze drip shield and waste package degradation.

wash. Term used in the southwest for a broad, gravelly, dry bed of an ephemeral stream, generally in the bottom of a canyon.

waste form. A generic term that refers to the different types of radioactive wastes.

Waste Handling Building. In the North Portal Area, a structure designed to support waste handling operations and the loading and staging of waste packages.

waste package. A sealed container containing waste that is ready for emplacement. The waste package includes the waste form and any containers, spacing structures or baskets, and other absorbent materials immediately surrounding an individual waste container placed internally to the container or attached to the outer surface of the disposal container.

waste package remediation system. A repair facility for disposal containers and waste packages that have failed the weld inspection processes, that are defective or abnormal, or that have been selected for retrieval from the repository for performance confirmation examinations.

waste stream. Input of waste into the repository over time.

water table. (1) The upper limit of the portion of the ground wholly saturated with water. (2) The upper surface of a zone of saturation above which the majority of pore spaces and fractures are less than 100 percent saturated with water most of the time (unsaturated zone) and below which the opposite is true (saturated zone).

welded tuff. A tuff that was deposited under conditions where the particles making up the rock were heated sufficiently to cohere. In contrast to nonwelded tuff, welded tuff is considered to be denser, less porous, and more likely to be fractured (which increases permeability).

xerophytic. Plants adapted to low moisture conditions.

Yucca Mountain disposal system. A combination of underground engineered and natural barriers within the controlled area that prevents or substantially reduces releases from the waste.

Young's modulus. The ratio between tensile or compressive stress and elongation of a solid stressed in one direction.

zeolites. A large group of hydrous aluminosilicate minerals that act as molecular "traps" because they can adsorb molecules with which they interact. At Yucca Mountain, they are secondary alteration products in tuff rocks, caused by exposure to groundwater. Zeolites could act to retard the migration of radionuclides.

Zircaloy. A family of alloys of zirconium that may have any of several compositions. These alloys are frequently used as a cladding material.

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