Mountain. The survey was focused on the Amargosa Valley region, but included Beatty, Indian Springs, and Pahrump (Figure 3-76). Of special interest was the proportion of locally grown foodstuff that was consumed by local residents, that is, irrigated with the potentially contaminated groundwater, and details of what food types were eaten on a regular basis.

An initial small-scale survey was conducted in January and February 1997 to facilitate the process of defining the requirements for a more comprehensive survey. Using the knowledge gained from the initial survey, questionnaire design and interviewing procedures were established, and a full-scale sample survey was conducted in the spring of 1997. The full survey included an inverse gradient sample design to provide for more comprehensive survey representation of the inhabitants closer to Yucca Mountain. It was estimated that 13,000 adults reside in the survey area, with 900 adults (7 percent) residing in the Amargosa Valley. The fraction of households contacted over the entire survey area was 16 percent. This fraction ranges from 43 percent in Amargosa Valley to 11 percent in Pahrump (Table 3-23).

Table 3-23. Adults Surveyed for the Food and Water Consumption Model

	Households		
Community	Number Surveyed	Total Number	Percent Surveyed
Amargosa Valley	195	452	43
Beatty	250	751	33
Indian Springs	65	529	12
Pahrump	569	4,993	11
Total	1,079	6,725	16

The survey was conducted within the context of federal guidelines on minimizing respondent burden (Paperwork Reduction Act of 1995, as amended). As was the case for the pilot survey, the comprehensive survey questionnaire design followed the principles developed by the U.S. Office of Management and Budget (OMB 1983a; OMB 1983b). Underlying the entire project were Dillman's (1978) total design method principles and interviewing standards promulgated by the Institute for Social Research, University of Michigan (Guenzel et al. 1983), to maintain high response rates and accuracy. To ensure accuracy, the survey aimed at minimizing sample error and non-sampling error (Andersen et al. 1979, pp. 1-14). The survey included Spanish language interviews to accommodate respondents whose primary language is Spanish. Measures were taken to compensate for subjects who were difficult to interview. Additionally, demographic information was used to compensate for any gender bias that may have arisen (Dillman 1978, p. 248).

The biosphere modeling required estimates of annual consumption of selected foods in terms of weight. Although it was not feasible to collect this type of information directly through the survey, it was feasible to collect frequency information on food consumption. Therefore, data taken from tables compiled through national surveys on food intake (U.S. Department of Agriculture 1993, pp. 18–29) were combined with information from the survey to produce estimates of annual quantities, in kilograms, of the various food groups consumed (CRWMS M&O 1998i, Section 9.4.4.2). The percentage of survey respondents consuming well water and locally produced food from Amargosa Valley and the remainder of the survey area are shown in Figure 3-77. Estimates of the annual quantities of the various food groups are shown in Figure 3-78.

In general, a higher percentage of locally produced food is consumed by residents in the Amargosa Valley than residents in the remainder of the survey area. Well water is consumed by nearly 88 percent of Amargosa Valley residents while 79 percent did so in the remainder of the survey area. Nearly 80 percent of the survey respondents reported consuming locally produced food of some type over the past year in the Amargosa Valley, while only about 57 percent did so in the remainder of the survey area. Thus Amargosa Valley residents have food consumption habits that make them more susceptible to radionuclide intake through the ingestion pathway than their immediate neighbors, supporting their designation as a likely population that includes the critical group.

The survey data were stratified into six subsets to evaluate each group for their appropriateness to represent the three potential reference persons (Section 3.8.3.2)—a statistically average person, a

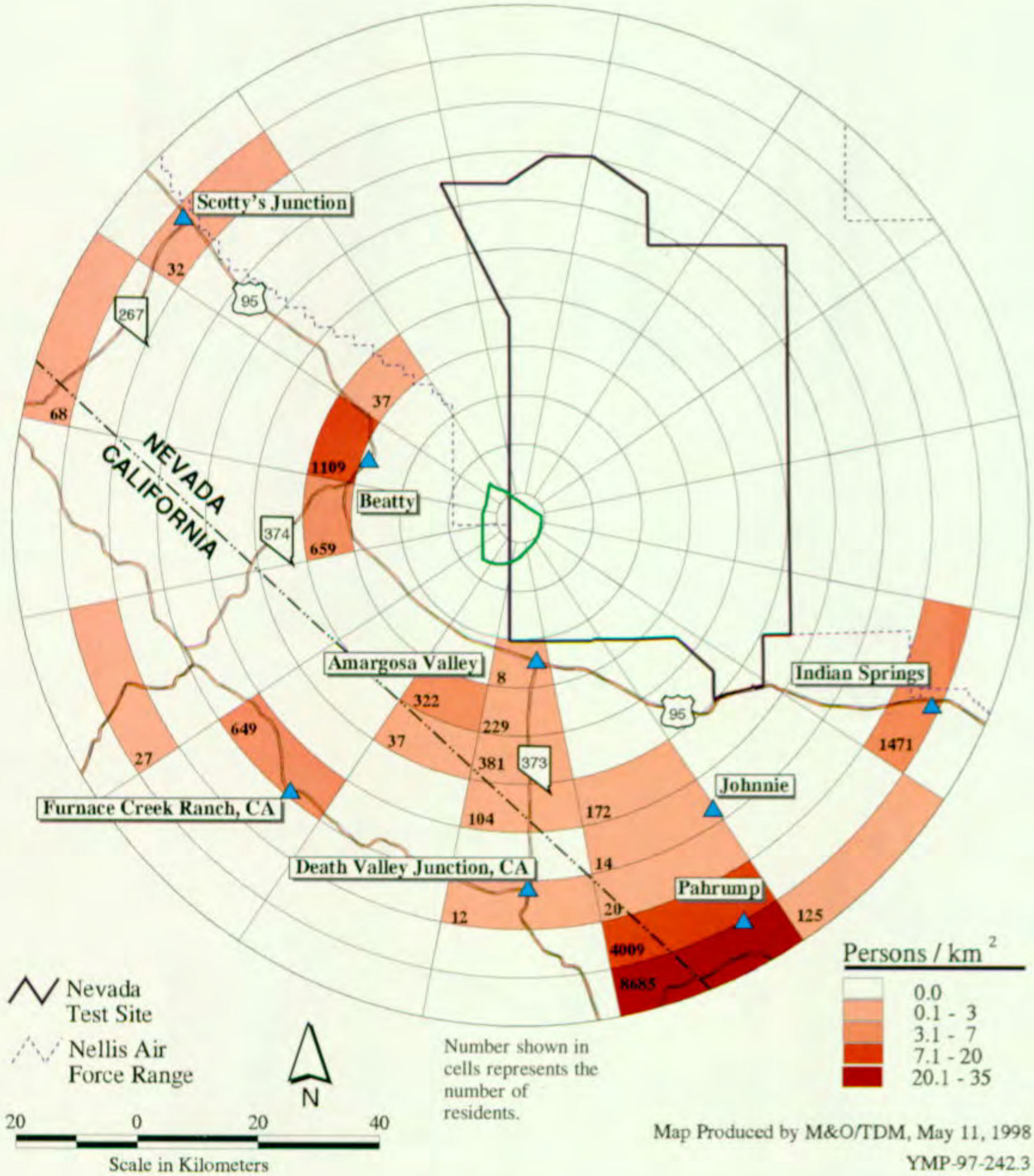


Figure 3-76. Map Showing the Number of Permanent Inhabitants Included in the Biosphere Modeling Work Group's Regional Food and Water Consumption Survey
A larger percentage of people were surveyed in the Amargosa Valley because of proximity to the potential repository.

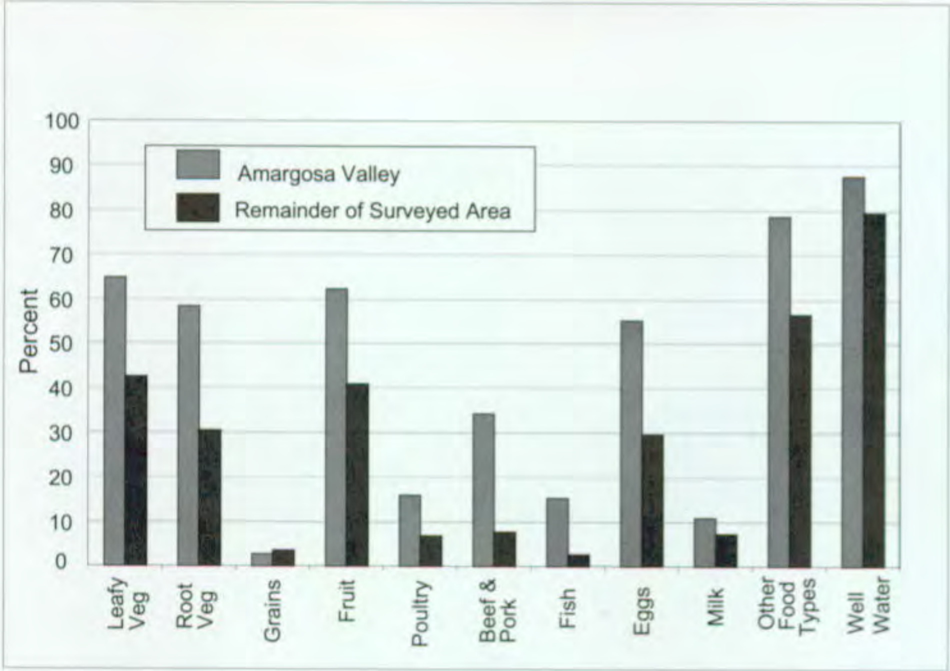
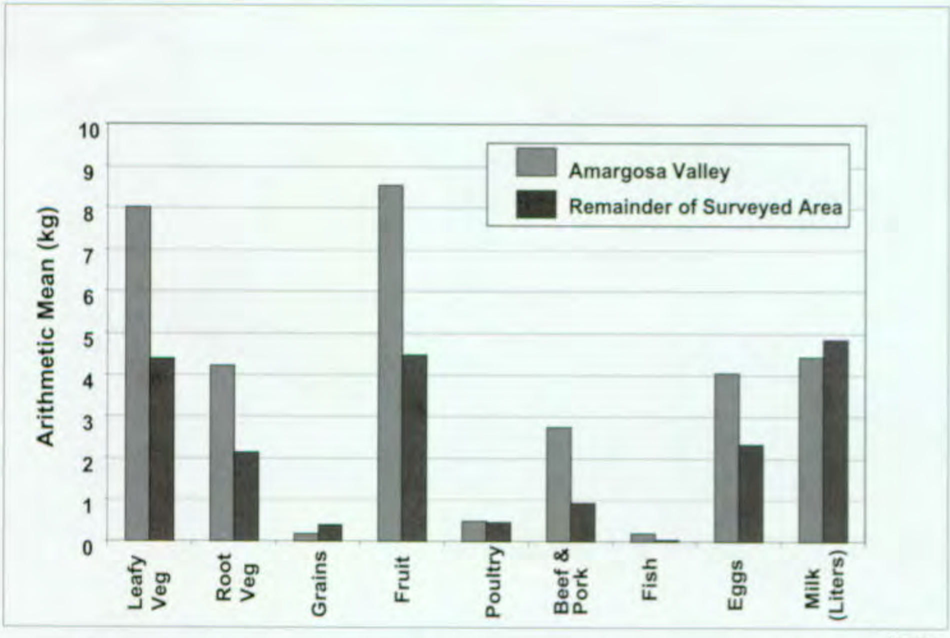


Figure 3-77. The Biosphere Regional Food and Water Consumption Survey
The survey found that a higher percentage of Amargosa Valley residents consume locally produced foods (with the exception of grain) and well water than those in the remainder of the survey area which included Pahrump and Beatty.



Well water: **Amargosa residents consumed 684 liters.**
Remainder of surveyed area consumed 646 liters.

Figure 3-78. Quantities of Locally Produced Food and Well Water Consumed by Amargosa Valley Residents
Survey data show that Amargosa Valley residents annually consume substantially greater quantities (kg) of locally produced food and well water than the residents in the remainder of the survey area.

Table 3-24. Receptor Types Considered in the Total System Performance Assessment Biosphere Modeling Effort (CRWMS M&O 1998i, Ch. 9)

Biosphere Modeling Receptor	Regional Survey Data Subset Correlate	Comments
Subsistence Farmer: Consumes only locally produced food and tap water. Adult spends large amount of time (about 17 hours/day) outdoors engaged in activities required to maintain subsistence. [Represents upper limit of contamination exposure]	Total survey subsistence resident adult	Survey data set for Amargosa Valley subsistence resident adult was deemed too small of a population to yield statistically meaningful interpretations for parameter development.
Residential Farmer: Relative to the subsistence farmer, this receptor consumes half the quantity of locally produced food, (but the same quantity of water) and spends less than half as much time engaged in outdoor activities and about 10 hours/day on weekdays and 5 hours/day on weekends outside Amargosa Valley.	None	This receptor is intended to represent the "median" level of exposure risk relative to the subsistence farmer and the average Amargosa Valley resident.
Average Amargosa Valley Resident: Adult living the average lifestyle of an Amargosa Valley resident. Consumes some locally produced food, almost as much tap water as the other two receptors (about 1.8 L/day versus 2.4 L/day), and spends about 10 hours/day outdoors on weekdays.	Amargosa Valley total population resident adult	Survey data set for Amargosa Valley total population resident adult was sufficiently large to yield statistically meaningful interpretations for parameter development. Because of its proximity to Yucca Mountain, this data subset was deemed more appropriate than the total survey area total population resident adult data set.

resident farmer, and a subsistence farmer (Table 3-24). No person interviewed in Amargosa Valley completely fit the description of a subsistence farmer; consequently, the data from the total survey area for subsistence residents were used to represent a subsistence farmer for the sensitivity studies. Also, none of the survey data subsets were considered to be representative of the resident farmer. The residential farmer was thought to be more representative of a member of the critical group currently residing in Amargosa Valley. For the TSPA-VA base case, the average person was selected as the reference person, again to be consistent with guidance from the National Research Council (National Research Council 1995, p. 52).

3.8.2 Implementation of the Performance Assessment Model

The complexity of the biosphere conceptual model requires a computer program to analyze all the factors. For performance assessment, the biosphere is only one of many component models, and a complex biosphere model is inappropriate for direct inclusion into the TSPA-VA calculations. Therefore, a two-phased approach was used. First, a complex computer model of the biosphere was used to generate biosphere dose conversion factors for each radionuclide of interest. Second, the annual dose from each radionuclide was calculated by forming the product of the biosphere dose conversion factor and the radionuclide concentration in groundwater as predicted by the TSPA computer program. The total annual dose to the reference person was then compiled as the sum of the individual annual doses from all radionuclides.

A biosphere dose conversion factor is a multiplier that converts radionuclide concentration in the groundwater at the well into an annual dose received by a human. To accurately predict the annual dose, the model used for generating the biosphere dose conversion factor must take into account all significant features, events, and processes involved. The model takes into account the pathways taken by radionuclides from a source (in the base case, well water) to and within a human (pathways and radiation dose within a human are approximated using a dose conversion factor). While the TSPA base case considers only nine radionuclides (Section 3.5.1.5), biosphere dose conversion factors were calculated for a unit concentration of 1 pCi/L for 39 radionuclides, including the nine in the base case. The biosphere dose conversion factors were generated by performing multiple evaluations of the biosphere model using input parameters sampled from defined distributions and are, therefore, statistical distributions.

The approach followed in generating the biosphere dose conversion factors comprised a rigorous process (CRWMS M&O 1996d, pp. 13-14). The initial step was to determine an appropriate computer program. Computer programs that have been used in the United States regulatory environment for dose assessment purposes were evaluated. Each program was compared with capability criteria established by the Yucca Mountain Site Characterization Project (CRWMS M&O 1998i, Section 9.2.2.1). GENII-S (Leigh et al., 1993) was the most comprehensive program and included a stochastic modeling capability, and therefore, was selected as the modeling tool for this effort (CRWMS M&O 1996d).

An interaction matrix was developed to identify the features of the biosphere and the events and processes by which radionuclides move from one feature to another (CRWMS M&O 1998i, Table 9-1). Using the interaction matrix, important elements of the biosphere and pathways between these elements were defined. These elements are presented in Figure 3-75. Most of the parameters used by the model define the characteristics of these elements and the pathways.

Although many of the input parameters were derived from local site-specific data obtained through the Yucca Mountain regional survey and weather data, many input parameters were also taken from other published sources (LaPlante and Poor 1997; International Atomic Energy Agency 1994). The input parameters employed in the biosphere modeling realizations are detailed in the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i, Table 9-3). Some of the important uncertain parameters (that is, parameters defined with probability distributions) that were used in the model are outlined in Table 3-25.

The ingestion pathway involves plant and animal uptake factors, irrigation rates, the amounts and

Table 3-25. Selected Uncertain Parameters Used in the Biosphere Modeling
(Source data are found in DTNM09806MW DGENII.000 and are non-Q)

Uncertain Parameter	Distribution
Soil-to-Plant Transfer Scale Factor	Log normal (0.117, 8.51)*
Animal Uptake Scale Factor	Log normal (0.117, 8.51)*
Inhalation Exposure (hr/year)—Amargosa Valley	Triangular (3248, 3869, 4217)
Inhalation Exposure—Mass Load (g/m^3)	Log normal (2.40E-6, 1.54E-4)
Home Irrigation Rate (in./year)—Current Precipitation Regime	Uniform (46, 96)
Crop Re-suspension Factor (/m)	Log normal (5.89E-7, 1.70E-4)*
Crop Interception Fraction (-)	Triangular (0.06, 0.4, 1.0)
Drinking Water Consumption (L/year)—Amargosa Valley	Triangular (0, 683.8, 1487.5)
Leafy Vegetable Grow Time (days)	Triangular (45, 67, 75)
Leafy Vegetables Irrigation Rate (in./year)—Current Precipitation	Triangular (25, 36, 66)
Leafy Vegetables Irrigation Time (month/year)	Triangular (2, 3, 4.9)
Leafy Vegetables Yield (kg/m^2)	Uniform (1.8, 2.6)
Leafy Vegetables Consumption Rate (kg/year)—Amargosa Valley	Log uniform (0.035, 59.68)
Other Vegetables Consumption Rate (kg/year)—Amargosa Valley	Log uniform (0.0045, 38.01)
Fruit Consumption Rate (kg/year)—Amargosa Valley	Log uniform (0.001, 97.69)
Grain Consumption Rate (kg/year)—Amargosa Valley	Log uniform (1.0E-31, 12.33)
Beef Consumption Rate (kg/year)—Amargosa Valley	Log uniform (2.0E-7, 53.11)
Poultry Consumption Rate (kg/year)—Amargosa Valley	Log uniform (5.0E-9, 10.50)
Milk Consumption Rate (L/year)—Amargosa Valley	Log uniform (5.0E-12, 136.03)
Eggs Consumption Rate (kg/year)—Amargosa Valley	Log uniform (0.009, 33.34)

* Truncated Log normal Distribution -- Values Represent: Min. (0.1 percentile), Max. (99.9 percentile)

Note: The log normal distributions are actually truncated distributions, parameterized by the 0.1 and 99.9 percentile values, and not the typical mean and standard deviation.

types of foodstuffs that are consumed by the reference person, etc. The consumption rates for drinking water, vegetables, fruit, grain, beef, poultry, milk, and eggs were estimated from the results of the survey. The soil-to-plant transfer factor represents the activity concentration (Ci/kg, dry weight) ratios between the soil and the edible parts of plants. This factor determines the amount of radioactive material accumulated in plants from soil. Similarly, the animal food transfer coefficient is the ratio of activity concentration in animal product (Ci/kg) to the daily activity intake rate (Ci/day). The coefficient determines the amount of radioactive material in edible animal products resulting from the ingestion of contaminated feed. Data on the transfer factors for local food types are limited; therefore, generic food transfer factors were taken from the International Atomic Energy Agency (International Atomic Energy Agency 1994, Chapter 6).

Inhalation exposure is influenced by two factors: the mass concentration of particles of such size that they can enter and be retained by the lungs and the duration of the exposure. Inhalation exposure, as well as direct external exposure, is dependent on the amount of time the reference person spends outdoors. The external/inhalation exposure mass load is the amount of material, dust for example, in a given volume of outside air. The home irrigation rate is the water application rate to lawns. Although not shown in Table 3-25, irrigation rates change with the different climates modeled and decrease with increasing precipitation.

Not all uncertainty in parameters is reflected in the values and distributions assigned. For example, the soil-to-plant uptake parameter, although defined by a distribution, does not reflect the soil properties of the area surrounding Yucca Mountain, but rather a more generalized temperate soil (International Atomic Energy Agency 1994). A recently completed study of the soils in the Amargosa Valley and the area north extending to Yucca Mountain reported exclusively alkaline soils (CRWMS M&O 1997i, Table 2, p. 8). Soils with naturally high pH, including many of the calcareous soils of the western United States, are associated with deficiencies of high-valance metal cations—iron, manganese, zinc, copper—for agronomic and agricultural production (Brady 1984, Section 11.4, p. 371). Given that many of the radionuclides of concern are metallic cations, the soil-to-plant uptake factors used in TSPA-VA might overestimate the actual amount of radionuclides transferred to plants.

As is usual practice for radiological compliance evaluations, dose conversion factors (not to be confused with the biosphere dose conversion factors) used in the biosphere model were assigned constant values and, therefore, do not appear in Table 3-25. A dose conversion factor converts an amount of a radionuclide into a radiation dose, taking into account such effects as the uptake of a radionuclide by the body, the residence time for a radionuclide in the body, and where a radionuclide is concentrated in the body. There are separate dose conversion factors for ingestion, inhalation, and direct external exposure. The values used in the biosphere modeling were derived using the methodology from the International Commission on Radiological Protection 30 (ICRP 1978) and are similar to those specified by EPA.

The final result of the complex biosphere modeling is the biosphere dose conversion factor. To calculate the radiation dose incurred by the reference person in the TSPA-VA, a biosphere dose conversion factor for a given radionuclide is sampled from its distribution, and the sampled biosphere dose conversion factor is multiplied by the concentration of that radionuclide at the biosphere/geosphere interface. The result is the annual dose rate that the reference person receives from that radionuclide at a given time. The biosphere dose conversion factors are completely correlated in the TSPA-VA calculations; that is, if a large biosphere dose conversion factor is sampled for one radionuclide, then large biosphere dose conversion factors are sampled for all radionuclides. During a TSPA-VA calculation, at appropriate sample times, the climate is allowed to change and biosphere dose conversion factors are changed to the precipitation regime associated with this climate. The sum of the dose rates for all radionuclides is the total annual dose rate at that time. The maximum total dose rate over all times in a TSPA realization is the peak dose rate—the end product of the TSPA-VA calculation.

3.8.3 Results and Interpretation

For each radionuclide, the GENII-S biosphere model performed 130 evaluations to generate a distribution of the biosphere dose conversion factor. Each of the 130 values was sorted into bins and used to construct a histogram. A log-normal distribution was fit to the output and a test for goodness-of-fit indicated that there is no reason to reject the hypothesis that these distributions were log-normal. Log-normal distribution approximations of the biosphere dose conversion factors were used in the TSPA-VA calculations to simplify the parameter sampling. Figure 3-79 presents the histogram and log-normal distribution for the biosphere dose conversion factor for neptunium-237

Analysis of the results of the biosphere modeling shows that the most important pathway in the calculation of the biosphere dose conversion factors is typically the drinking-water-ingestion pathway. The next most important pathway is the leafy-vegetable-ingestion pathway. All other pathways generally affect the biosphere dose conversion factors at a relatively insignificant level. The fractional contribution to the biosphere dose conversion factors from the drinking-water-ingestion pathway for three radionuclides that have an important impact on the TSPA-VA base case results (Sections 4.2 and 4.3) are as follows: 57 percent for technetium-99, 44 percent for iodine-129, and 64 percent for neptunium-237. The fractional contribution from the leafy-vegetable-ingestion pathway are as follows: 37 percent for technetium-99, 21 percent for iodine-129, and 31 percent for neptunium-237. The only other important pathway for these radionuclides is the meat-ingestion pathway for iodine-129, which contributes 26 percent of the iodine biosphere dose conversion factor. These numbers were generated with a deterministic calculation that used the expected values of all parameters and considered the average Amargosa Valley resident and the current climate.

The following two sections present analyses to examine parameters and assumptions that are important to the GENII-S biosphere modeling.

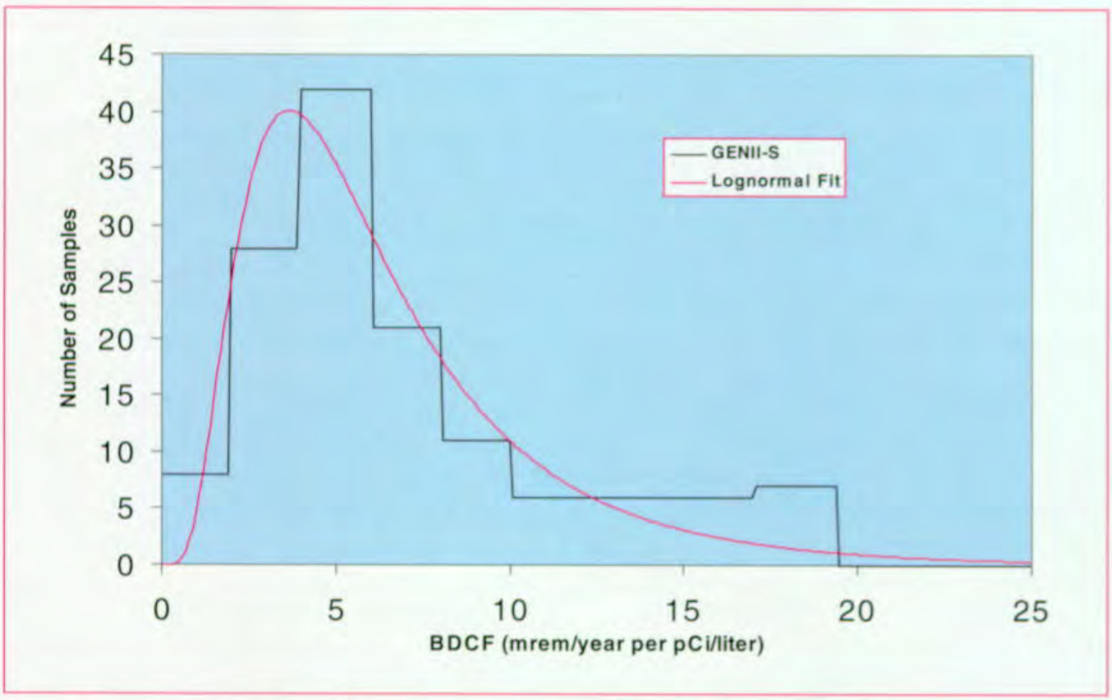


Figure 3-79. Histogram of the Biosphere Dose Conversion Factor for Neptunium-237

The last section contains a description of the biosphere modeling and assumptions used in the assessment of the impact of volcanism on repository performance (Section 4.4). Sensitivity analyses that look at the impact of parameters and assumptions on the complete TSPA-VA base case calculations are discussed in Section 5.8.

3.8.3.1 Biosphere Probabilistic Results

The stochastic biosphere modeling shows that for a given radionuclide the biosphere dose conversion factors can vary by about a factor of three above and below the mean value, giving a range of approximately one order of magnitude. Figure 3-79 shows the distribution predicted by GENII-S, with a mean of 6.6 mrem/year per pCi/liter, and the log-normal distribution used to approximate it. It is useful to understand which input parameters are most responsible for this variance in the output. Rank regression is a technique employed to assess the relationship between the model input and the output. The partial correlation coefficient is a measure of how much the calculated biosphere dose conversion factor is correlated with a given sampled parameter. Forty-six variables were evaluated in the parametric sensitivity study. Depending on which radionuclides were considered, typically less than 10 independent variables accounted for more than 90 percent of the variance of the model output. Table 3-26 summarizes the results of sensitivity analysis for technetium-99, iodine-129, and neptunium-237. Only the parameters with partial correlation coefficients greater than 0.2 are included in the table, which is approximately the level of insignificance in the analysis; the results of the biosphere modeling are even less sensitive to other parameters, such as the inhalation and external exposure parameters.

Of the uncertain or variable parameters that most affect the calculation of the biosphere dose conversion factors, the leafy-vegetable and drinking-water consumption rates are the most important. That is, changes in these two parameters typically produce the largest changes in the biosphere dose conversion factor. That these two parameters are identified by the analysis is not unexpected, because the probability distributions used to define them have relatively large variances, and because the drinking-water and leafy-vegetable pathways were identified as the major contributors to the dose rate (Section 3.8.3). Iodine-129 differs somewhat from technetium-99 and neptunium-237 in that iodine is more easily concentrated in animals and, thus, the beef consumption rate, and to a certain extent the milk consumption rate, are important to its biosphere dose conversion factor.

Parameters that are identified as the next in importance are the crop-interception fraction and the crop resuspension factor. The crop-interception

Table 3-26. Sensitivity Analysis for the Three Most Important Radionuclides for the Base Case Scenario

Radionuclide	Parameter	Partial Correlation Coefficient
Tc-99	Leafy vegetable consumption rate	0.87
	Drinking water consumption rate	0.77
	Crop interception rate	0.29
	Root vegetable consumption rate	0.26
	Eggs yield	0.20
I-129	Leafy vegetable consumption rate	0.62
	Beef consumption rate	0.52
	Drinking water consumption rate	0.46
	Milk consumption rate	0.26
	Crop resumption rate	0.25
	Grain irrigation rate	0.22
	Animal uptake scale factor	0.20
Np-237	Leafy vegetable consumption rate	0.84
	Drinking water consumption rate	0.77
	Crop interception rate	0.31
	Root vegetable consumption rate	0.23
	Eggs yield	0.22

fraction is the fraction of contamination from rainfall, irrigation, or aerosol deposition that is intercepted by and adheres to the plant surface. The crop resuspension factor describes the amount of contaminated dust that settles on the plant surface and can then be resuspended into the air. The adsorbed contaminants are then available for ingestion by foraging livestock and poultry, or direct ingestion by humans. Processes that contribute to these parameters are wind, and overhead irrigation, which is becoming more common in the region.

The least important parameters shown in Table 3-26 are the root-vegetable consumption rate, the eggs yield, the grain irrigation rate, and the animal-uptake scale factor. Although the partial correlation coefficients for these parameters are relatively insignificant, it is important to note that they still rank highly among the 46 parameters defined with probability distributions. The results of this sensitivity analysis can be used to identify the factors driving uncertainty of the model output and determine where attention and resources should be focused in the future (Section 6.5.1.11).

3.8.3.2 Biosphere Comparative Analyses

Several additional calculations were performed to evaluate some of the more important assumptions made in the biosphere modeling on the biosphere dose conversion factor calculations. Of particular interest is how the assumption of an average consumption rate of locally produced foodstuffs influenced the results. Also of interest is the effect of climate on the results.

To examine the effect of consumption rates, three different types of reference persons were defined: the average Amargosa Valley resident (the base case reference person), a resident farmer, and a subsistence farmer, each with defined characteristics (Table 3-24). The significance of the amount of local food and well water consumed on the biosphere dose conversion factors is presented in Figure 3-80. The figure shows that the magnitude

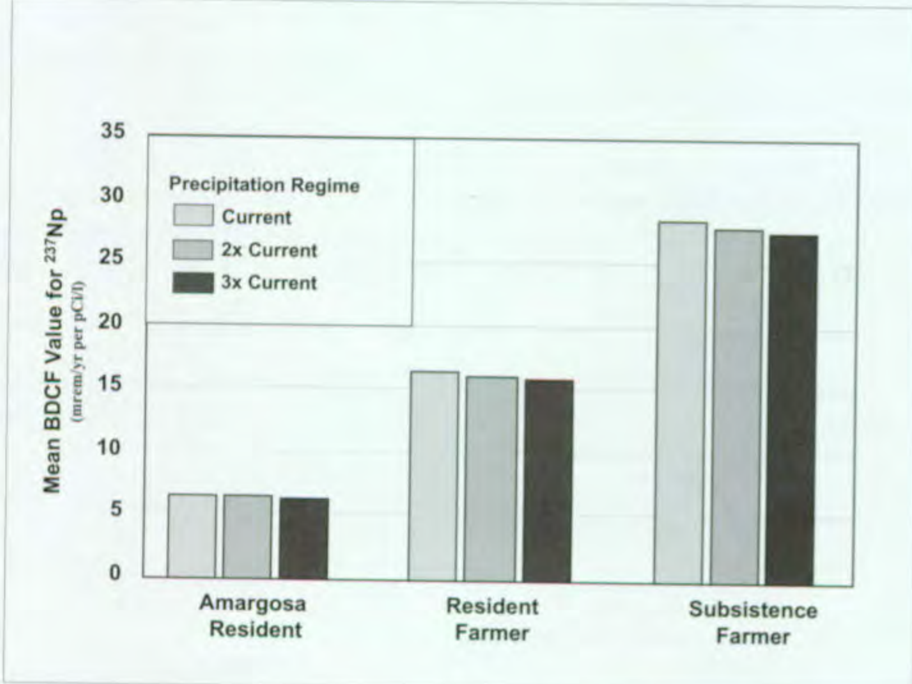


Figure 3-80. Comparison of Biosphere Dose Conversion Factors for Neptunium-237 as a Function of Receptor and Precipitation Regime
The biosphere dose conversion factors generated for the subsistence farmer are approximately 5–6 times greater than those calculated for the average current Amargosa Valley resident. However, the biosphere dose conversion factors do not vary notably among the three precipitation regimes modeled. (BDCF—Biosphere Dose Conversion Factors)

of biosphere dose conversion factor for a given isotope increases with the increased consumption of locally grown food (CRWMS M&O 1998i, Section 9.7.3.1). In general, a subsistence farmer could receive a radiation dose from neptunium-237 approximately five times greater and a resident farmer could receive a dose three times greater than that received by an average inhabitant of the region, respectively. However, these factors are dependent on the radionuclide; for iodine-129, a subsistence farmer would incur a radiation dose rate approximately 10 times greater than an average Amargosa Valley resident.

Also shown in Figure 3-80 is the change in climate, which is modeled only by the change in irrigation rates and has little effect on the biosphere dose conversion factors. For each unit of increase in precipitation, where a unit is the present-day mean annual precipitation, calculated biosphere dose conversion factors were reduced by about two percent. The decrease in dose rate results because the increase in precipitation allows a reduction in irrigation, and thus less contamination to soil and plants. The decrease in dose rate is small because the major pathway—drinking water—is unaffected by an increase in precipitation. Also, most of the precipitation increase is expected to occur in the winter when crops are not produced. The two-percent change is inconsequential in comparison to the approximate order-of-magnitude spread of the biosphere dose conversion factor distributions (CRWMS M&O 1998i, Table 9-22). Data gathered from a survey conducted in late 1997 in Lincoln County, Nevada, are presently being analyzed in order to determine the eating habits of residents in a cooler and wetter environment similar to that anticipated in the future at Yucca Mountain. The impact of this effect will be addressed as a sensitivity study when the data from Lincoln County are analyzed.

3.8.3.3 Biosphere Scenarios Associated with Volcanic Activity

As part of an assessment of the effects of disruptive events on a potential repository at Yucca Mountain (Section 4.4), two biosphere scenarios related to volcanic activity were modeled. The two biosphere scenarios for volcanism were as follows: the reference person is living in the vicinity when the eruption occurs and inhales air containing particles of contaminated ash, and the reference person returns to the region soon after the eruption and lives on and farms the contaminated volcanic soil. The biosphere model used to evaluate the volcanic scenarios involved the same biosphere-pathways model and reference-person parameters as developed for the base case, except that for Scenario 2 contaminated volcanic ash was assumed to be mixed into the upper 15-cm (6-in) of soil.

In the first scenario, the dose rate is primarily from inhalation; the dose rate from exposure by submersion in the ash cloud is very small in comparison to the dose rate from inhalation. In this scenario, the biosphere dose conversion factors were calculated for unit concentrations of the radionuclides in air and one hour of exposure. To determine the dose rate from these biosphere dose conversion factors, the quantity of radionuclides expelled by the event is calculated, then the radionuclides are distributed in an appropriate volume of air. Exposure time (in hours) is estimated. The dose rate is then determined by the product of the radionuclide concentration, the biosphere dose conversion factors for inhalation and submersion, and the assumed time of exposure.

In the second scenario, it is assumed that the ash from the eruption contributes its activity to the upper 15 cm (6 in) of the ground surface because this depth encompasses the root zone of most agricultural plants. For these calculations, the groundwater is assumed not to be contaminated because this factor has already been taken into account in the base case biosphere-dose-conversion-factor calculations. As with the base case, the biosphere dose conversion factors include the ingestion, inhalation, and external exposure pathways. Also, as with the base case, the dominant pathway is the ingestion of contaminated foods. To determine the dose rate in a TSPA calculation, it is necessary to determine the amount of radionuclides expelled by a volcanic event and the area over which the radionuclides are dispersed. These two parameters allow the concentration of radionuclide in the soil to be calculated. The concentration of radionuclides in the soil is then multiplied by the volcanic biosphere dose conversion factors to give the annual dose.

Within the time frame allowed for completion of the TSPA-VA, only the dose rate associated with exposure to contaminated ash after the assumed volcanic event (Scenario 2) was calculated (Section 4.4.2). The dose rate associated with inhalation and external exposure during the volcanic event itself (Scenario 1) will be addressed in additional studies if it is deemed that volcanic activity requires further consideration.

4. BASE CASE DEFINITION AND RESULTS

The previous sections in this volume have shown the complexity of the repository system, both the engineered and natural barriers, and the resulting need to break the system into various components in order to analyze repository performance. Section 2 described the specific components or subsystems used in the TSPA-VA models (Figure 2-2), the rationale for choosing these specific components (how they correlate to specific physical-chemical processes and to location within the overall system, and how they are coupled to one another), and the specific methodology used to incorporate and analyze these components in the overall total system performance model (Figure 2-13). Section 3 gave detailed descriptions of each of the key component models: the conceptual basis of these models as determined by experimental data, how to implement these conceptual models into the overall TSPA analyses (including model abstractions or simplifications that are necessary in some instances), the range of uncertainty underlying these component models and their corresponding data, and finally some key results of the component models, that is, how well the models fit available data and how the range of uncertainty in available data leads to a range of predictions of key model output parameters (e.g., groundwater flux).

Having shown in previous sections the necessity of breaking the total system into components and the details of modeling each component, it is now appropriate to describe the results of recombining all the component models into one integral total system performance model, specifically the predictions of this overall model with respect to both total system performance (i.e., dose rate at the accessible environment) and to subsystem performance of the various components (e.g., performance of the saturated zone by itself). A key point is that the assessments will be based on the calculated source term, that is, the predicted radionuclide releases from the repository as the waste packages and waste forms degrade with time. This is a bit different from some of the results in Section 3 (e.g., Sections 3.6 and 3.7), which were based on an idealized source term.

The phrase "base case" has already been defined in more general terms in Section 2.3.3 as it relates to uncertainty. However, in this section "base case" is defined more specifically as it relates to the various component models of the TSPA-VA, that is, which parameter ranges and models are used to define the TSPA-VA base case. The phrase "base case" might also be termed the "baseline" case, where baseline is used in the sense of "likely" rather than as "initial," implying that the "base case" encompasses some of the possible uncertainty regarding predictions of system performance, but not all. For example, consider the fourth type of uncertainty listed in Section 2.3.3.3 regarding features, events, and processes: uncertainty about future events. The TSPA-VA base case encompasses some uncertainty related to climate-change events, but no uncertainty related to volcanism events. Or consider the second type, conceptual model uncertainty. The TSPA-VA base case does not include the DKM/Weeps model discussed in Section 3.1, but does include the DKM model. These are a couple of examples of how the base case represents a narrowed range of uncertainty in comparison to the entire space of possible outcomes. It represents a judgement. However, because of the still remaining uncertainty in all of the component models and future scenarios (features, events, processes), the base case necessarily encompasses multiple outcomes (i.e., realizations) for the future repository behavior. The intent of Section 4 is to quantify the range of possible outcomes of overall system behavior for the TSPA-VA base case and to describe the key parameters of the component models that are most responsible for the performance variation across this range of system behavior (Section 4.3). Before describing this entire range of overall base case behavior, that is, the probability distribution of dose in the biosphere at 20 km (12 miles) downgradient of the repository, a single realization of repository behavior is examined to illustrate in detail how the behavior of the various components influences both the behavior of the dose and the behavior of other components (Section 4.2). This will lead to a more thorough understanding of why the repository behaves as it does and what can be done to develop a more robust (i.e., an inherently safer) system through either better engineering in the drifts or better characterization of the natural system. This single realization is the realization that chooses the expected value of all the stochastic input parameters.

System behavior outside of the range of the base case is also discussed in this section, and in Section 5, including:

- Effects on the system behavior from unanticipated, or low-likelihood, disruptive events such as volcanism (Section 4.4)
- Alternative design options for the repository and waste packages (Section 4.5)
- Less likely conceptual models for various components (Volume 5)

4.1 BASE CASE DESCRIPTION

Figures 2-11 through 2-13 illustrated how the base case component models described in Section 3 are coupled together and what information is passed through the couplings. A detailed description of the base case includes two features: the specific parameter values used within the component-model "boxes" in Figure 2-13, and the specific information and form of that information passed through the arrows that couple the boxes. Uncertain ranges of parameters used within and between the boxes for the TSPA-VA base case have been described in detail in the component sections of Section 3. The purpose of Section 4.1 is to provide a concise summary of these base case parameter values and models as used in TSPA-VA.[1]

The TSPA-VA base case is predicated on the reference repository and waste package design described in detail in Volume 2, and summarized in Section 3.2.2.1 of Volume 3. Variations in this design and their effect on performance are described in Section 4.5 but are not considered in Sections 4.2 through 4.4.

4.1.1 Climate

As explained in Section 3.1.2.1, future climate changes in the TSPA-VA base case are assumed to alternate between three states: present-day or dry climate, long-term average climate (about twice the dry-climate precipitation), and superpluvial climate (about three times the dry-climate precipitation). For the expected-value simulation described in Section 4.2, this sequence of alternating climate states, including the durations of each climate state, is shown in Figure 4-1. As summarized in Table 3-4, the length of the first dry climate, or current climate, is sampled uniformly between 0 and 10,000 years, with an expected value of 5,000 years, which was the value used for the expected-value base case calculation (Section 4.2). The length of subsequent dry climates is sampled uniformly between 0 and 20,000 years, with an average of 10,000 years. The duration of long-term average climates is sampled uniformly between 80,000 and 100,000 years, with an average of 90,000 years, which is the value used for the expected-value base case calculation. The duration of the superpluvial climates is sampled similarly to dry climates, from 0 to 20,000 years. Dry and long-term average climates alternate from the time of repository closure until the long-term average climate that spans the 250,000-year mark. For the expected-value case, the last 10,000 years of that long-term average period is replaced by 10,000 years of superpluvial climate. Then the climate model returns to alternating dry and long-term average climates until the long-term average climate that spans the time of 400,000 years after the first superpluvial. The end of that long-term average climate is again replaced by a superpluvial, and then back to dry, followed by alternating long-term average and dry climates. For the expected-value base case simulation shown in Figure 4-1, about 90 percent of the time is spent in the long-term average climate.

[1] A complete listing of all model parameters is beyond the scope of this document. The complete listing is found in the various chapters of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

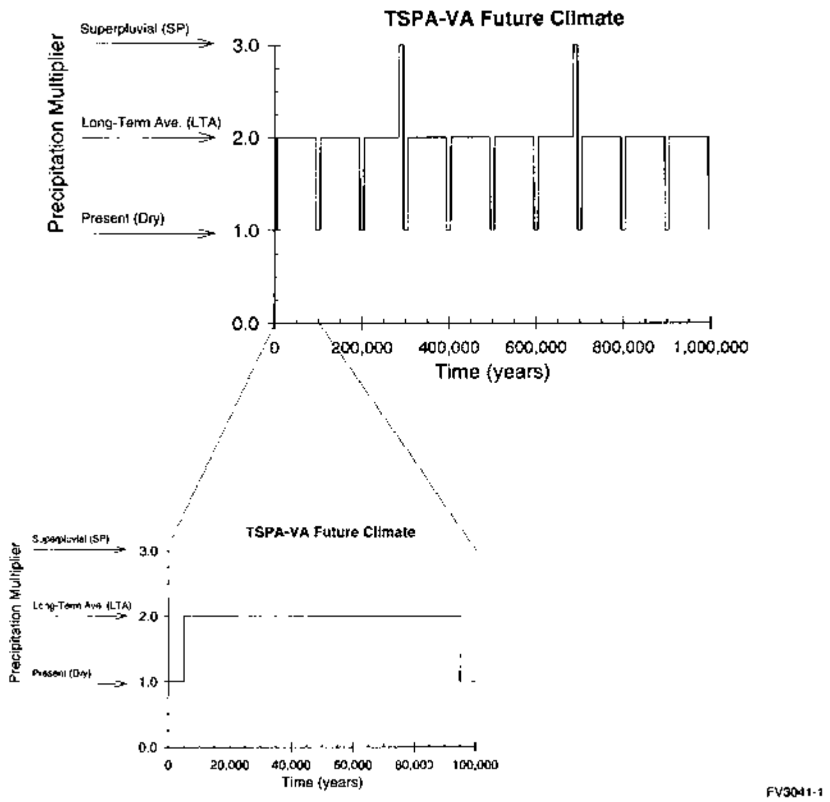


Figure 4-1. Climate History for the Expected-Value Realization of the Total System Performance Assessment for the Viability Assessment Base Case at Two Different Time Scales
For other realizations, the duration of the various climate states is different, since these durations are stochastic parameters.

4.1.2 Unsaturated Zone Flow and Infiltration

Unsaturated zone flow uses three calibrated hydrogeologic property sets for present-day conditions, resulting in three calibrated steady-state flow fields: the "base infiltration" case with mean fracture alpha, which represents the best estimate of current infiltration conditions; the "base infiltration × 3" case with maximum fracture alpha, which represents a pessimistic (very wet) estimate for current infiltration; and the "base infiltration ÷ 3" case with minimum fracture alpha, which represents the most optimistic (driest) estimate for current conditions.[2] To derive flow fields for the other climate states, these three calibrated property sets were used in the TOUGH2 computer code to attain steady states at boundary conditions representing precipitation in the two wetter climates. For the base case, there were 3 × 3 steady-state flow fields accessed by the RIP computer program. All nine resulting flow fields are three-dimensional, with vectors for fracture and matrix flux and for fracture-matrix interflux at each of the approximately 40,000 grid points (actually 80,000 since there are two continua-fracture and matrix).[3]

[2] Originally there were five calibrated hydrogeologic property sets for present-day climate conditions: the three mentioned plus a "base infiltration × 3" case with minimum fracture alpha and a "base infiltration ÷ 3" case with maximum fracture alpha (see Section 3.1). These latter two were dropped from the TSPA-VA base case because of similarities to the other "base infiltration × 3" and "base infiltration ÷ 3" cases.

[3] The unsaturated zone transport model requires only the vectors from the repository horizon down to the water table.

The water table level changes with climate, rising by 80 m (262 ft) from dry to long-term average climate and another 40 m (131 ft) from long-term average to superpluvial climate.

Figure 4-2 illustrates how the flow fields are used throughout the climate history. The figure indicates that infiltration is heterogeneous over the model domain and, therefore, the percolation at the repository horizon is heterogeneous. This heterogeneity is taken into account in the unsaturated zone transport model, which directly uses the three-dimensional flow fields. The repository area in this figure is the area in the center where the grid-block discretization is much greater. The exploratory studies facility is also shown by areas of greater discretization. (The actual repository does not cover the entire area of greater discretization but only about the northernmost 80 percent of that area.) Also shown on the figure is the approximate average infiltration over the repository area for the three climate states.

4.1.3 Drift Scale Seepage

Liquid seepage into the drifts in the model is based on numerous simulations of a three-dimensional, heterogeneous, fracture continuum surrounding the drifts, as described in the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i). The simulations use many different infiltration rates and nine different property sets (Table 3-6) that represent nine combinations of fracture permeability and fracture alpha. The results were statistically weighted and represented as two stochastic response surfaces (Figure 3-13) of seepage fraction, (i.e., the fraction of packages in the drifts that have seeps dripping on them), and seepage flux per package (in m^3/ year). These two

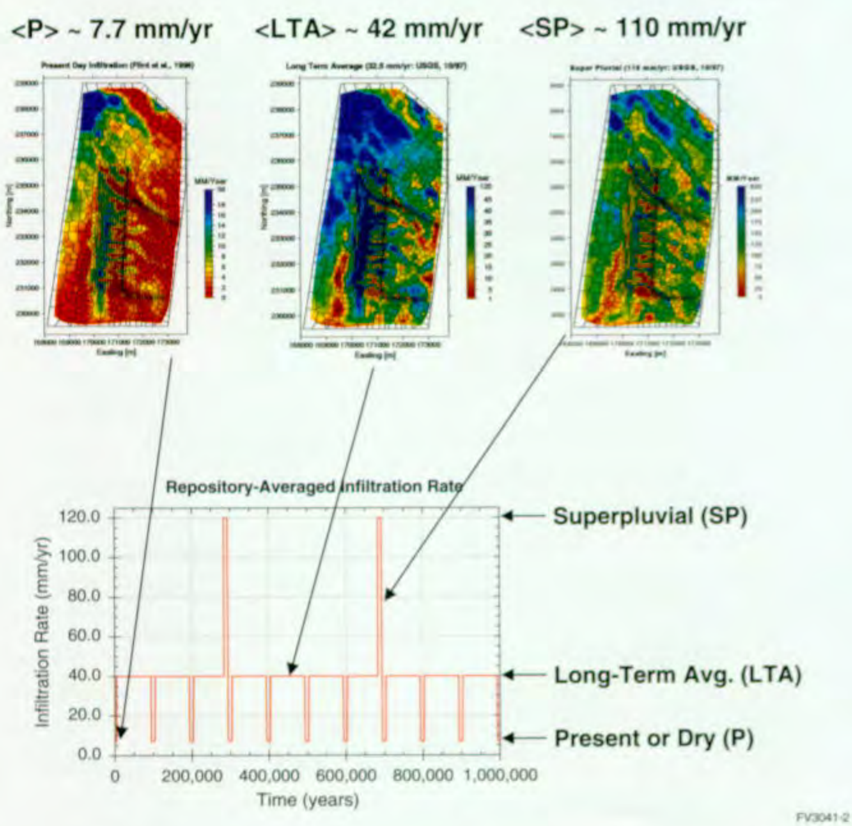


Figure 4-2. Infiltration History for the Expected-Value Realization of the Total System Performance Assessment for the Viability Assessment Base Case
This figure shows the infiltration map for the unsaturated zone flow model in the three different climate states, and the average infiltration over the repository horizon during these climate states.

response surfaces are a function of percolation flux: when the climate changes, different values for seepage fraction and seepage flux are used based on the new percolation flux. For the base case, the seepage flux and seepage fraction were given perfect positive correlation, that is, the same random number was sampled for these two parameters. Also, the same random number was used in all climate states, meaning that although seepage changes with climate state (because percolation flux changes), the different seepage values used in different climate states have the same probability. The seepage fraction and seepage flux histories (along with the corresponding response surfaces) are shown in Figure 4-3. The repository horizon was divided into six discrete regions based on infiltration and thermal-hydrologic response. Each region used a separate value of seepage flux and seepage fraction over the entire region. These values were based on the average percolation flux over each region as determined from the histogram over all grid blocks in the region. The average percolation flux was taken from the steady-state values of percolation predicted by the unsaturated zone flow model. This discretization is illustrated in Figure 4-4.

The seepage flux and fraction used in the TSPA-VA base case are for ambient percolation conditions, that is, unperturbed by the thermal pulse. This is justified because the large changes to seepage from waste heat occur at times earlier than

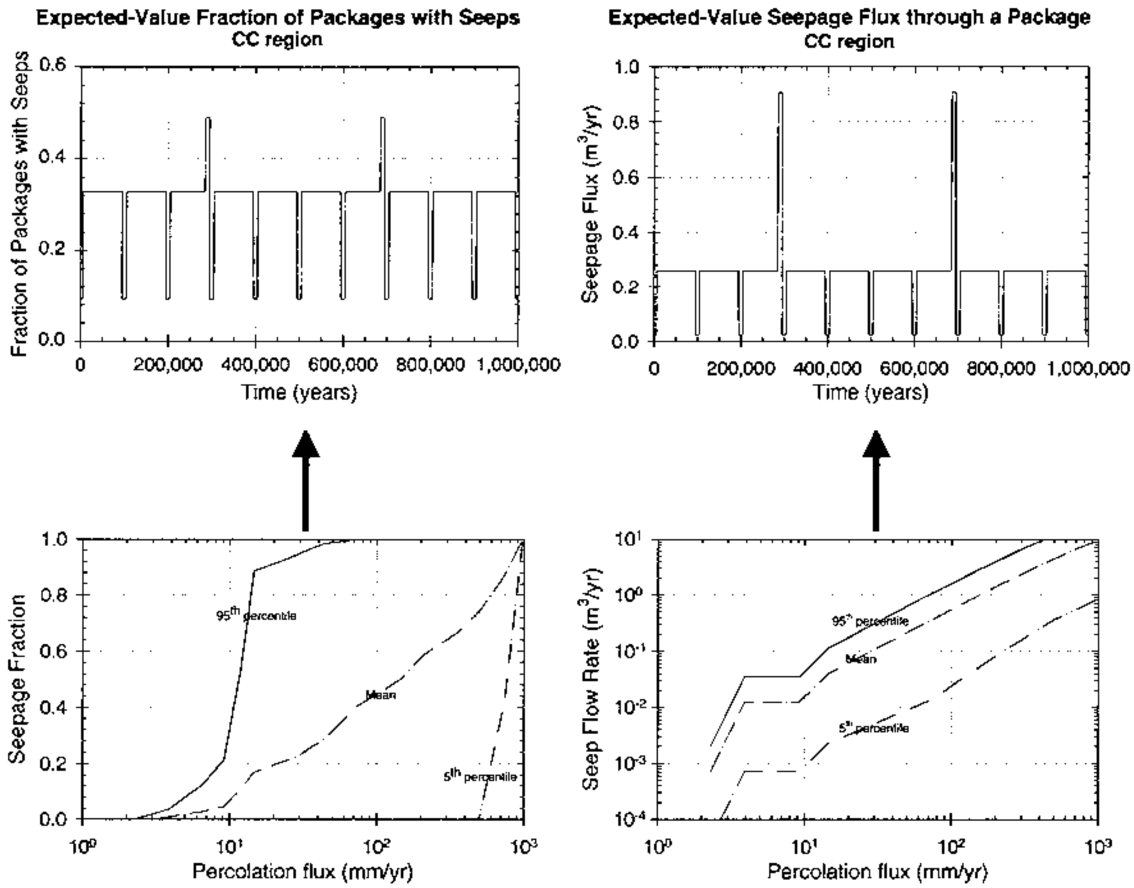


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Figure 4-3. Total System Performance Assessment for the Viability Assessment Base Case Drift-Scale Seepage Model
This figure shows the 1 million-year expected-value histories of seepage flux into the drift and fraction of packages in a given region that encounter seeps (seepage fraction). Also shown is the probabilistic distribution used to predict seepage flux and seepage fraction based on percolation flux at the repository horizon in any of the six regions (see Figure 3-8 for an illustration of the six regions).

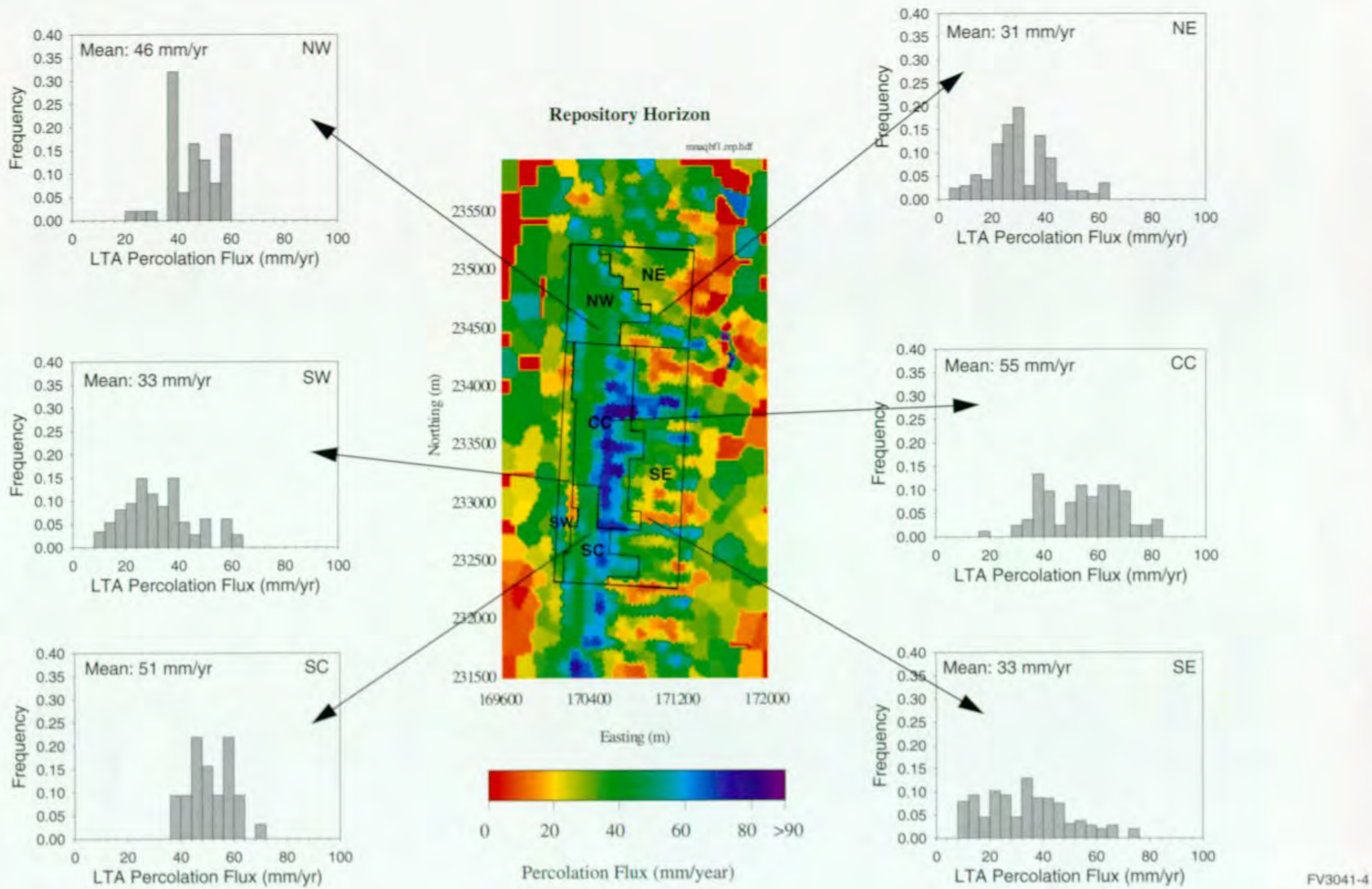

Figure 4-4. Total System Performance Assessment for the Viability Assessment Base Case Percolation Flux at the Repository Level for the "Base Infiltration" Flow Field During the Long-Term-Average Climate

Shown are the six waste package regions, including the histogram (spatial frequency distribution) of fluxes in each region derived from the unsaturated zone flow model and the spatially averaged percolation flux in each region. LTA means long-term average.

the first package failures, and so have no effect. This is discussed in the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

4.1.4 Drift Scale Thermal Hydrology

In the base case, the results of drift-scale, thermal-hydrologic modeling are as follows:

- Waste package surface temperature (T_{wp}) and relative humidity at the waste package surface (RH_{wp}) for seven different package types and heat outputs within six discrete spatial regions of the repository, which is used by the waste package degradation model to represent some of the package-to-package variability within each region[4]

- Average waste form temperature (T_{wf}) and liquid saturation in the invert (S_l) in each of the six regions, which is used by the waste form degradation model[5]

- Average drift-wall temperature (T_{dw}), relative humidity at the drift wall (RH_d), and liquid saturation in the invert (S_l) in the central-central region, which is used by the near-field geochemical models

Each of these results is generated separately for the three different unsaturated zone flow fields: "base infiltration × 3, with maximum fracture alpha" case, "base infiltration, with mean fracture alpha" case, and "base infiltration ÷ 3, with minimum fracture alpha" case. Waste package degradation histories are then generated for each of these unsaturated zone flow scenarios and fed into the RIP computer program. Also, waste form degradation within the RIP program uses all three flow field results. However, near-field geochemistry only uses average drift wall temperature (T_{dw}), relative humidity, and liquid saturation in the invert (S_l) for the "base infiltration with mean fracture alpha" flow field. The waste package degradation model produces different results only for packages that are dripped on versus those that are not. Otherwise, the same package degradation history is used for each inventory type (commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste). Figure 4-5 shows the package-to-package variability in waste package surface temperature (T_{wp}) and relative humidity at the waste package surface (RH_{wp}) in the northeast region predicted by the drift-scale thermal-hydrologic modeling. The details of which waste packages correspond to the curves in this plot are described in the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

4.1.5 Repository Scale Thermal Hydrology

Thermal-hydrology models at the scale of the entire repository were only used in two places in the base case:

- As three-dimensional, conduction-only models that fed the abstraction methodology for drift-scale thermal-hydrologic parameters (see Section 3.2)

- As two-dimensional, thermal-hydrologic models (over an east-west cross section running through about the center of the repository) that provided air mass fraction and gas flux to the near-field geochemical environment models

In both cases, the hydrogeologic property set came from the "base infiltration" flow field from the unsaturated zone flow model (see Section 3.1). For the first set of models (three-dimensional,

[4] Actually, after numerous simulations of waste package degradation in the six discrete repository regions (see Sections 3.2 and 3.4), the failure histories were determined to be similar enough to only warrant further simulations in one of the six regions. The northeast region was conservatively chosen because it had the most variability (although only marginally so), implying slightly earlier package failures, and the package-failure history for this region was used for all the package groups in the RIP program.

[5] In this case, the results for all six regions (rather than just the northeast region) are used in the RIP model.

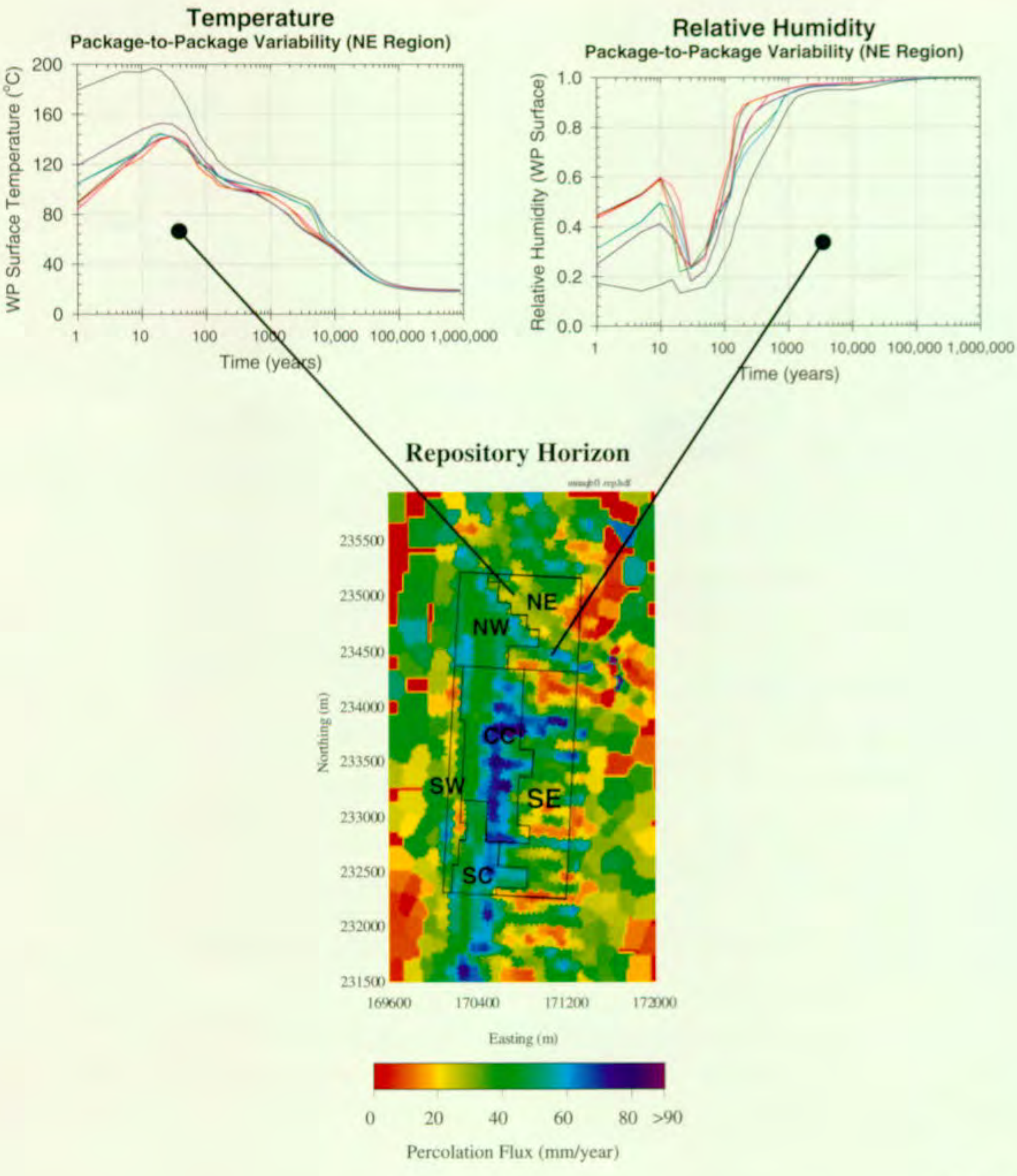

Figure 4-5. Variability of Total System Performance Assessment for the Viability Assessment Base Case, Drift-Scale, Thermal-Hydrologic Parameters (Temperature and Relative Humidity) in the Northeast Region at the Repository Horizon

These plots are for the "base infiltration" flow field using the long-term-average climate. Variability results from spatial variability in infiltration flux, thermal-hydrologic properties, and heat loading of the packages. (The abbreviation "WP" means waste package.)

conduction-only), the climate state was irrelevant, since hydrology was not modeled. For the second set of models (two-dimensional, thermal-hydrology), only the long-term average climate was used. This was appropriate because it is assumed that this climate exists approximately 90 percent of the time.

4.1.6 Near-Field Geochemical Environment

For the base case analyses, the near-field geochemical environment is represented by time-dependent distributions of the gas and water compositions that interact with the waste package and waste form inside the drift. The parameters supplied to the RIP analyses are the oxygen fugacity (fO_2),[6] the solution pH, the total dissolved carbonate (ΣCO_3^{2-}), and the ionic strength (I) of the water. Figure 4-6 shows time histories of these four chemical composition parameters as used by the RIP program for the TSPA-VA base case, derived from the various near-field geochemical models and supplied in tabular form to RIP. The first three parameters are used for waste form degradation modeling and the last one for colloidal transport in the engineered barrier system. These four parameters are derived from considering a more comprehensive 13-component chemical system. The values represent the time-dependent composition of water entering the drift and reacting with iron-oxide corrosion products, as opposed to water that has reacted with concrete. (Concrete-modified water is the subject of a sensitivity analysis considered in Section 5.)

It is clear from Figure 4-6 that these four composition time histories are not smooth functions; they are a series of steps. This step—change approximation is based on a simplified representation of the thermal history derived from various thermal-hydrologic models. The temperature, gas flux, and air mass-fraction histories from these thermal-hydrologic models were approximated as a sequence of steady states, which allowed the near-field environment to be modeled as a sequence of constant-composition states. This is the way all numerical models work—approximations of continuous functions with a series of discrete values. In the case of the near-field environment, the width of the discrete time periods was rather large. For each of these discrete time periods, often referred to as "abstracted" time periods, a batch calculation is performed with a geochemical simulator (EQ3/6; Wolery 1992a) to determine the appropriate chemical composition of the groundwater during that time period based on the temperature and air mass fraction from the thermal-hydrology models and on the incoming composition of the groundwater reacted with iron oxides. The base case values and abstracted time periods represent the thermal history of the "central" portion of the repository, as opposed to the "edge" portion. The values and time periods were generated using both the air-mass fractions and gas fluxes from the two-dimensional, mountain-scale, thermal-hydrologic results for the center portion of the two-dimensional cross section and the average drift-wall temperature history of the central-central region for the "base case infiltration" history, derived from three-dimensional drift-scale thermal-hydrologic modeling.[7] The central-central region was chosen because the excursions from ambient conditions, in particular the period of boiling, are largest for this region and will bound[8] the effects at the edge portions of the repository where the excursions occur for shorter time periods.[9]

4.1.7 Waste Package Degradation

Because modeling every distinct waste package is not computationally efficient, waste packages are grouped according to their inventory (commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste) and environment.

[6] For an ideal gas, oxygen fugacity is the same as oxygen partial pressure, which is about 0.2 atmospheres at sea level.

[7] More details of the rather complicated set of models used to derive the near-field geochemical parameters are given in Section 3.3 and also in the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

[8] "Bound" is always used in the sense of "conservative" (i.e., "bounding" models always predict higher doses).

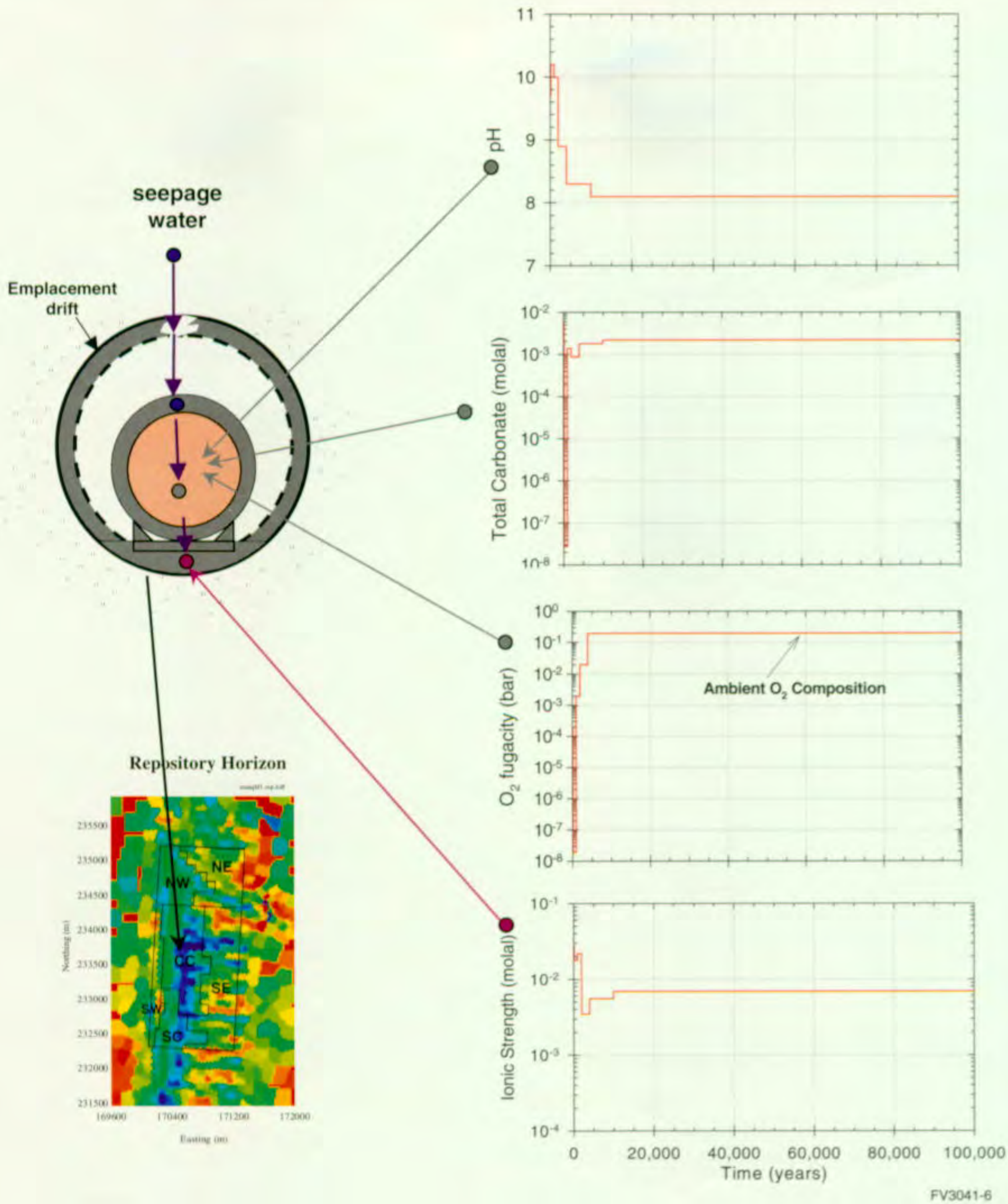

Figure 4-6. Time Histories of the Near-Field Geochemical Parameters
Histories are for pH, Total dissolved carbonates, oxygen fugacity, and infiltration used at various locations in the TSPA-VA base case, based on the expected-value climate history shown in Figure 4-1. These near-field geochemical time histories are based on the central-central (CC) region but are applied to all six waste package regions.

Packages are grouped into four environments based on dripping conditions:

- Packages dripped on during the dry climate and throughout the simulation time
- Packages that are not dripped on during the dry climate but will be in all other conditions (long-term average and superpluvial)
- Packages that are dripped on only during the superpluvial climate
- Packages that are never dripped on[10]

Although seepage behavior through the packages is different for each of these four groups, only two waste package failure histories are used: packages that are always dripped on and packages that are never dripped on. The *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) has supporting analyses to show why degradation of the corrosion allowance material and the corrosion resistant material is sufficiently modeled with only these two conditions. As explained in a footnote in Section 4.2.2, the packages that are never dripped on and the packages dripped on only during the superpluvial climate are modeled with the no-drip failure history from the WAPDEG computer code, and the other two groups are modeled with the always-drip failure history. The packages are also grouped according to the six regions shown in Figure 4-4, which yields different seepage fluxes in different regions, and therefore different rates of radionuclide transport out of the waste packages and through the engineered barrier system. However, as mentioned in a footnote in Section 4.1.4, the same waste package failure histories are used regardless of region, based on the responses in the northeast region.

The waste package degradation analysis for the base case includes a set of nine cases to represent a range of the uncertainty in the median high-nickel alloy (Alloy 22) corrosion rate and the variability of the Alloy 22 corrosion rate among waste packages and patches. Uncertainty and variability are represented by splitting the total variance of the general corrosion rate for Alloy 22 into three different variability and uncertainty combinations: 75 percent variability and 25 percent uncertainty, 50 percent variability and 50 percent uncertainty, and 25 percent variability and 75 percent uncertainty. For each of the variability-uncertainty splits, the median general corrosion rate is sampled from the 5th, 50th and 95th percentiles of the uncertainty variance, respectively. The expected-value base case is the 50 percent variability and 50 percent uncertainty split and the median corrosion rate at the 50th percentile of the uncertainty variance.

In addition to corrosion degradation of waste packages, the base case includes the effect of early or "juvenile" waste package failures by noncorrosion processes such as materials defects, seismic activity, and human-induced factors, such as welding, handling, and transportation. In the base case, juvenile failure is assumed to cause a single patch opening in a commercial spent nuclear fuel package in a dripping environment, in the southeast region. The expected-value base case assumes a single juvenile failure at 1,000 years. The probabilistic base case uses a log-uniform distribution for juvenile failures from

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[9] The multi-scale thermal-hydrology models described in Section 3.2 indicate that much of the repository functions as though it is center-like in thermal-hydrologic behavior; however, there is a gradual transition to edge-like behavior as the repository edges are approached. The major difference between the center and edge portions is that shorter abstracted time periods are appropriate for the boiling regime in the edge region as compared to the central portion of the repository. This difference is also true for the two wetter climate states because of the increased cooling capacity of the system.

[10] It is important to realize that packages never move from one environmental group to another, that is, packages that are "dripped on" are dripped on for the entire simulation history and packages that are "never dripped on" are never dripped on for the entire simulation history.

0.001 percent to 0.1 percent, all occurring at 1,000 years after closure.

Figure 4-7 shows the cumulative package penetrations by corrosion, through either pits, patches, or breaches (pits plus patches), for the base case (50/50 split on uncertainty/variability). The figure also shows the time history of average pit area per package and average patch area per package. Because of their much larger area, patches are far more important than pits in causing releases from the packages. Also, all patches and pits are assumed to encounter seeping water at all times.

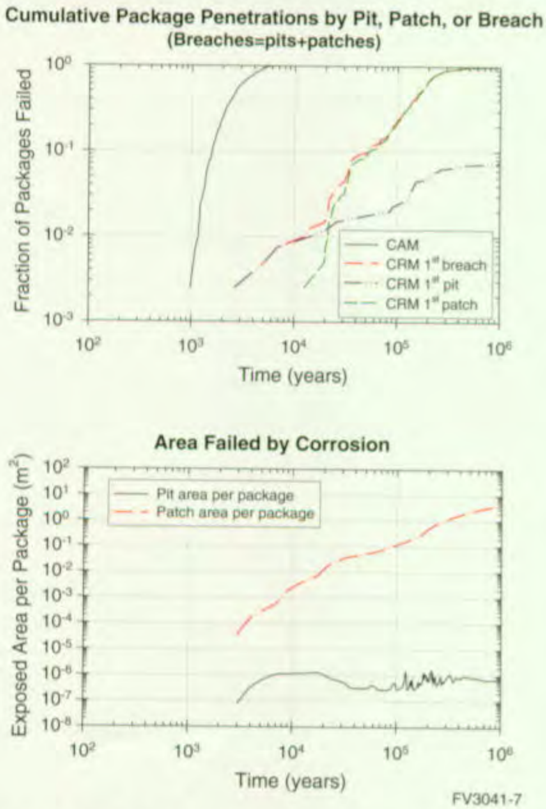

Figure 4-7. Total System Performance Assessment for the Viability Assessment Base Case Waste Package Degradation Histories
Shows cumulative package penetrations by pit, patch, or breach. Also shows growth of average patch area and average pit area per waste package. (CAM—corrosion allowance material [the outer waste package layer made of carbon steel]; CRM—corrosion resistant material [the inner waste package layer made of Alloy 22])

4.1.8 Cladding Degradation

Several mechanisms for cladding degradation of commercial spent nuclear fuel[11] are included in the base case:

- Creep (strain) failure at high temperatures with simultaneous exposure to oxygen
- Juvenile cladding failures (packages emplaced in the repository with cladding already failed)
- Total failure of stainless-steel-clad fuel rods
- Mechanical failure of cladding because of structural failure of the waste package, due to rockfall, corrosion, or seismic events after closure
- Long-term corrosion failure

Juvenile and creep failures combined with failure of the stainless steel cladding is defined at a constant rate of 1.25 percent, beginning at the time of any waste package failure and continuing indefinitely. Most of this is due to the stainless-steel failure. Long-term mechanical failure is initiated with package failure and is defined as a distribution with an expected value of 0.18 percent at 100,000 years after package failure and 2.62 percent at 1 million years after package failure. Long-term corrosion failure is also initiated with package failure, with an expected value of 1.51 percent at 100,000 years after package failure and 7.75 percent at 1 million years after package failure (see Figure 3-54).

[11] In TSPA-VA, cladding on DOE spent nuclear fuel elements is assumed to have failed before emplacement. This is a conservative assumption intended to provide a bound on the maximum possible impact of DOE spent nuclear fuel on repository performance. In reality, not all DOE spent nuclear fuels have faulty cladding (e.g., about 50 percent of N-reactor cladding is intact, as well as 100 percent of naval spent nuclear fuel).

4.1.9 Waste Form Degradation and Mobilization

In the base case, waste form degradation is modeled within the RIP program by defining the dissolution rate equations for fuel exposed by package and cladding degradation. The dissolution rate for commercial spent nuclear fuel is a function of the temperature (T), total dissolved carbonate, oxygen fugacity, and pH of the water contacting the waste form. There is also some uncertainty in the dissolution rate model associated with the equation used to fit the experimental data, and this is a stochastic parameter in the base case. Figure 4-8 shows the commercial spent nuclear fuel dissolution rate as a function of temperature, pH, and total dissolved carbonate. Oxygen fugacity is assumed at atmospheric conditions for these plots. Within the RIP program, the input variables, temperature, total dissolved carbonate, oxygen fugacity, and pH are passed to an external routine, and the dissolution rate is calculated at every time step. The average temperature on the waste package surface is used for the six different repository regions, as calculated by the drift-scale, thermal-hydrology model. This assumption is appropriate because at the time the waste packages begin to fail, the difference between the interior and surface temperatures of the waste package will be small.

The glass dissolution rate is a function of the waste package surface temperature and the pH of the incoming water, and also has a similar stochastic parameter related to uncertainty in the regression fit of the experimental data. A metallic dissolution rate is used for the DOE spent nuclear fuel, which is modeled using a surrogate radionuclide inventory composed mostly of N-Reactor fuel. (The differences in the dissolution rates for the commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste fuels are shown in Figure 4-17.) The specific surface area for commercial spent nuclear fuel is about 0.004 m^2/g, which causes exposed commercial spent nuclear fuel to dissolve in about 1,000 years if the entire surface area is 100 percent wet, which is the assumption for the TSPA-VA. For high-level radioactive waste, the specific surface area is 5.7×10^{-5} m^2/g and for DOE spent nuclear fuel it is 7.0×10^{-5} m^2/g.

4.1.10 Transport in the Engineered Barrier System

Transport in the engineered barrier system is modeled within the RIP program using a series of mixing cells coupled by advective and diffusive "connections" (Figure 4-9). The waste package and the waste form together are represented as a single cell pathway. The concrete invert below the

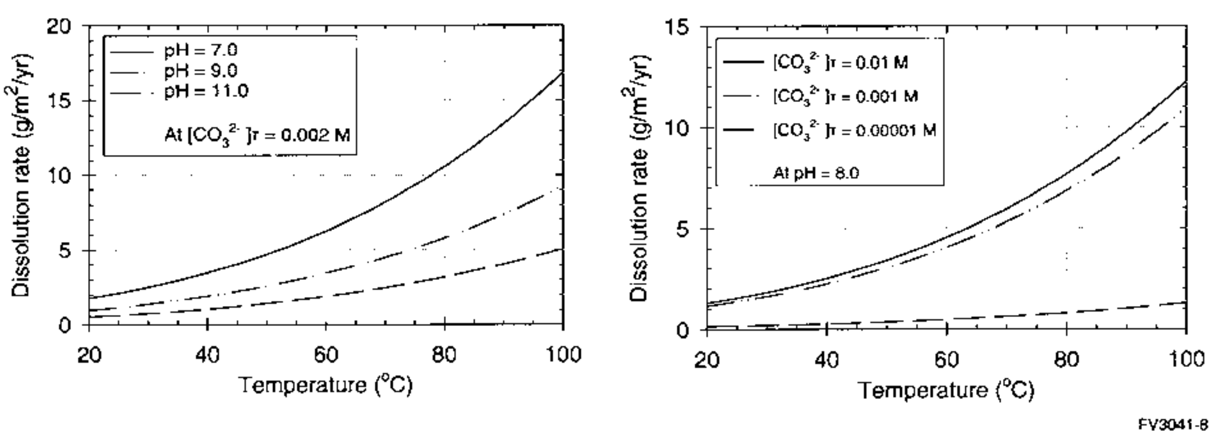

Figure 4-8. Commercial Spent Nuclear Fuel Degradation Rate
Rates are for the TSPA-VA base case as a function of temperature, pH, and total dissolved carbonate in the water contacting the waste form assumes an oxygen fugacity equal to 0.2 atmospheres. (*M*—molar)

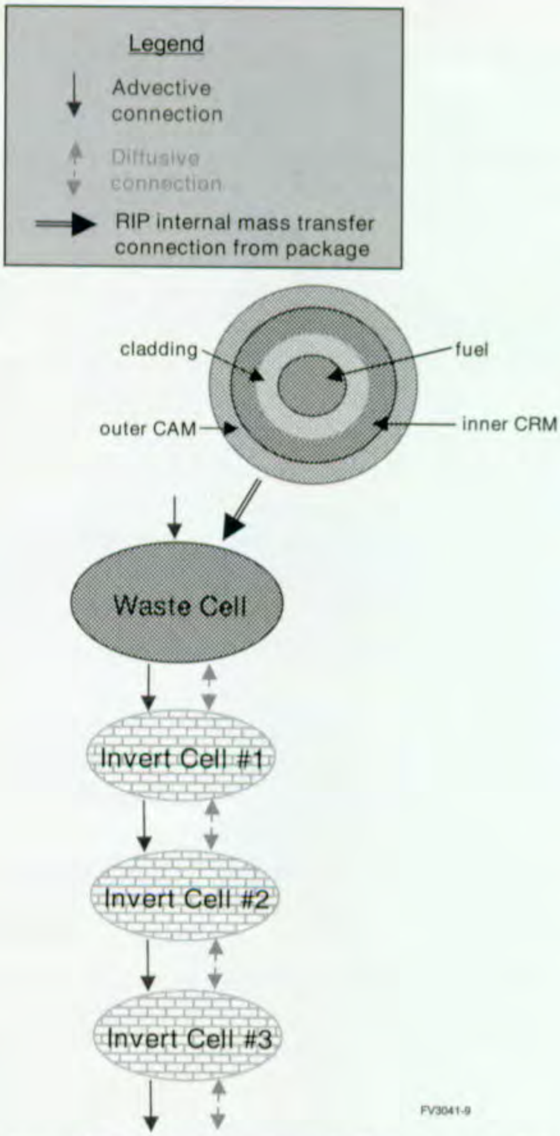

Figure 4-9. Configuration of Cells in the RIP Program for Engineered Barrier System Transport in the Total System Performance Assessment for the Viability Assessment Base Case
(CAM—corrosion allowance material [the outer waste package layer made of carbon steel]; CRM—corrosion resistant material [the waste package layer made of Alloy 22])

package is assumed to be shaped as a semi-circle around the package and is discretized into three cell pathways to decrease the numerical dispersion. The waste-package/waste-form cell pathway is coupled to the first invert cell pathway through an advective connection to represent water seepage through the failed waste package into the invert, and with two diffusive connections for separately modeling the diffusive mass transfer through the pit perforations and patch perforations of a failed waste package. The volumetric flux through the waste package was calculated by scaling the seepage flux into the drift with the available surface area arising from patch and pit failures (see Figure 4-7).[12] The pit perforations and the patch openings also define the area for diffusion.

Volumetric flux through the invert cells is not the same as the reduced flux through the packages but is equal to seepage into the drift. Diffusion through the invert is calculated for a concrete material with 10 percent porosity and 99.8 percent water saturation[13] using an equation for diffusion in a partially saturated medium (Conca 1990; Conca and Wright 1992). The concrete invert also sorbs some of the radionuclides, based on the following sorption coefficients (K_ds) (see Chapter 6 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* [CRWMS M&O 1998i]):

- Neptunium K_d is uniform from 0 to 200 mL/g; expected value is 100
- Protactinium K_d is uniform from 0 to 100 mL/g; expected value is 50
- Plutonium K_d is uniform from 0 to 1,200 mL/g; expected value is 600
- Uranium K_d is uniform from 0 to 100 mL/g; expected value is 50

Colloid-facilitated mobilization and transport of radionuclides in the engineered barrier system is modeled for plutonium-239 and plutonium-242, assuming fast reversible attachment. There are four types of colloids in the engineered barrier

[12] The surface area available for seepage was increased by a "seepage collection factor" to account for the focusing of water dripping on the waste package. This factor was sampled uniformly from 1 to 10, with a mean of 5.5.

[13] The liquid saturation is actually a function of time, as determined by drift-scale, thermal-hydrologic modeling, but it returns to 99.8 percent (that is, ambient saturation) by the time the packages begin to fail.

system: two natural types, iron-oxides and clays; and two types based on fuel degradation, spent nuclear fuel colloids and glass waste colloids. The expected-value colloid partition coefficients (K_cs) for these four types are as follows (see Chapter 6 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* [CRWMS M&O 1998i]):

- 0.24 for iron-oxides
- 0.024 for clays
- 0.087 for spent nuclear fuel particles
- 0.72 for glass particles

A small fraction of the total mobilized plutonium, that is, the plutonium in both the aqueous and reversibly sorbed colloid phases, is assumed to sorb irreversibly to colloids at the edge of the engineered barrier system to account for the fraction observed at various nuclear waste sites, for example, the very small amount observed at the Nevada Test Site from the Benham nuclear blast. This fraction is sampled log-uniformly in the range of 1×10^{-4} to 1×10^{-10}, with a value of 1×10^{-7} for the expected-value case.[14] The basis for this range is discussed in Chapter 6 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

For radionuclides with a very high solubility in the pore water seeping through the packages, the radionuclide mass dissolved at each time step is swept away in the next time step, that is, the carrying capacity of the water is very high. Low solubility radionuclides re-precipitate and are only removed slowly as a linear function of the seepage flux through the package. This effect is most important for neptunium-237, plutonium-239, plutonium-242, and uranium-234. The solubilities used in TSPA-VA are about the same as in the 1995 TSPA, except for neptunium-237, (CRWMS M&O 1995) and are listed in Table 3-15. The most important solubility value is for neptunium-237, which is 0.34 g/m^3 or 1.43×10^{-6} *M* for the expected-value realization shown in Section 4.2.[15] The volume of water available for mobilization of waste at any time step (i.e., for computing solubility) is equal to the volume of water in the pores of the "rind", which is the in growing shell of altered fuel being changed into other minerals through contact with gases and liquids that have entered the degraded waste package. This water volume is based on a porosity of 40 percent and a water saturation of 100 percent in the rind. The rind volume itself is linearly proportional to the fuel rod volume times the fuel dissolution rate (a constant with units of year^{-1}) times the time since first water contact.

4.1.11 Unsaturated Zone Transport

The unsaturated zone transport model for the base case is based on the unsaturated zone flow model. The transport model uses the same three flow fields described above and the same climate states. As with unsaturated zone flow, a dual-permeability model is assumed, and transport is modeled with the FEHM particle tracker in three dimensions. Therefore, transport is assumed to occur in two continua, with fracture-matrix interflow based on the steady-state flow fields generated by the TOUGH2 computer code. The FEHM particle tracker transports particles on the same dual-permeability TOUGH2 spatial grid as used in the flow model. When the climate shifts, a new TOUGH2 flow field is provided from the run-time file directory, and the particles are assumed to be instantly traveling with the new velocities. When the water table elevation rises, the particles in the "lost" vertical distance corresponding to this rise are instantly released in that timestep to the water table. When the water table elevation falls, there is a brief time when there are no particles in this new vertical distance.

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[14] Actually, the exponent is sampled uniformly in the range of -10 to -4, which yields a mean of -7.

[15] Section 5.5.4 and Figure 5-34 discuss the solubility range for neptunium-237 used in TSPA-VA. The mean and upper and lower limits of this range have been reduced by about a factor of 100 from the 1995 TSPA based on new interpretations of experimental data. This is discussed in detail in Chapter 6 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

Sorption based on experimental data is assumed for the radionuclides (see Section 3.6). As discussed in Section 4.2, sorption is important for three key radionuclides in the unsaturated zone; neptunium-237, plutonium-239, and plutonium-242. The sorption coefficient (K_d) value for neptunium-237 is 1.0 mL/g in the vitric and devitrified tuff units and 4.0 mL/g in the zeolitic units. For the two plutonium species, the K_d is 100 mL/g in all units. Matrix diffusion is assumed for all radionuclides but, as shown in Section 5.6, has little effect on the dose rates. The assumed matrix-diffusion coefficient has a mean of 3.2×10^{-11} m^2/s for carbon-14, technetium-99, iodine-129, and selenium-79 and 1.6×10^{-10} m^2/s for neptunium-237, plutonium-239, plutonium-242, uranium-234, and protactinium-231. The expected value for the dispersivity used in the unsaturated zone model is 20 m in all three directions. Figure 4-10 shows an overlay of the source-term discretization at the top of the unsaturated zone model and the discretization of the output at the water table. Between the top and bottom, particles are free to move in any direction. The travel time during the long-term average climate is illustrated for a pulse release for three key radionuclides; technetium-99, neptunium-237, and plutonium-242.

4.1.12 Saturated Zone Flow and Transport

Figure 4-11 is a combination of several figures from Section 3.7 that illustrate the base case model for saturated zone flow and transport. A coarsely discretized (500 m in the x and y directions and 50 m in the z direction), three-dimensional numerical model is first used to simulate flow in the saturated zone. The model determines the general plume direction and flowpath length in the tuff and alluvium, based on the revised geologic framework model for lithostratigraphy (see Section 3.7). Radionuclide transport in the saturated zone is then modeled with six one-dimensional streamtubes placed along the general plume direction determined from the three-dimensional model. The six regions at the water table in Figure 4-11 are used to determine total volumetric flux from the unsaturated zone into each of six one-dimensional streamtube models in the saturated zone. The average specific discharge in the dry climate is assumed to be 0.6 m/year in all six of the streamtubes, based on the results of the saturated zone expert elicitation panel (CRWMS M&O 1998g). The streamtube models are run with the FEHM computer code, assuming an effective continuum, that is, the permeability is approximately equal to the fracture permeability but the porosity is approximately equal to the matrix porosity. The six streamtubes are used to develop unit breakthrough curves based on a constant concentration source. The breakthrough curves are scaled by the area of the streamtubes, which is proportional to the volumetric unsaturated zone flow into each streamtube, to get the correct concentration in each streamtube. The unit breakthrough curves are convolved at each time step with the actual RIP source term during run time. A dilution factor to simulate transverse dispersion is applied to the concentration curves for each 20-km streamtube, based on a probability distribution determined from the saturated zone expert elicitation panel. The same dilution factor (see Figure 4-11) is applied to all radionuclides, all streamtubes, at all time scales. The actual saturated zone concentration used to compute dose rate is essentially a function of the dilution factor; for high dilution factors, effective diffusive mixing among streamtubes is assumed and the six concentrations are added. For low dilution factors, no mixing is assumed and the maximum concentration streamtube is used. (See Figure 4-19 for a plot of the effective dilution factor used. See the end of Section 4.2.1 for a more detailed summary of the basis for either using the summed concentrations or the maximum concentration.) For the TSPA-VA base case, the values used for the various stochastic saturated zone model parameters are listed in Table 3-20.

4.1.13 Biosphere Transport

The final dose rate is calculated within RIP by multiplying a biosphere dose conversion factor by the saturated zone concentration discussed above. Water having a concentration equivalent to the saturated zone concentration is assumed to be drawn from a well 20 km (12 miles) downgradient from the repository. The water is then used as drinking and irrigation water for an "average" adult living in Amargosa Valley (see third row of

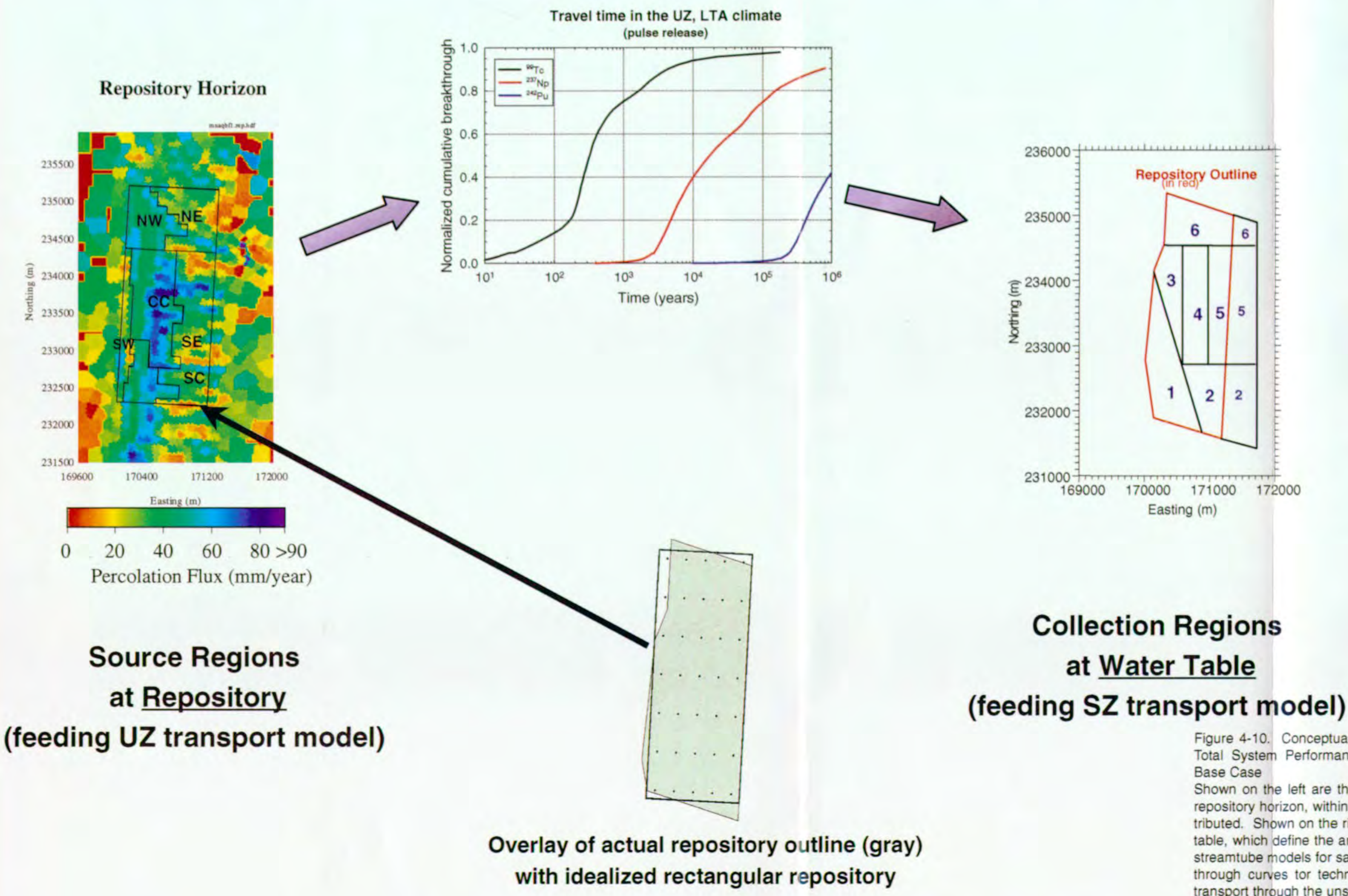

Figure 4-10. Conceptualization of Unsaturated Zone Transport for the Total System Performance Assessment for the Viability Assessment Base Case

Shown on the left are the six waste package or source regions at the repository horizon, within which radionuclide releases are uniformly distributed. Shown on the right are the six "collection" regions at the water table, which define the area and volume flux of the six one-dimensional streamtube models for saturated zone transport. Also shown are breakthrough curves tor technetium-99, neptunium-237, and plutonium-242 transport through the unsaturated zone in the long-term average climate for a pulse source. (The small overlay shows how nearly the six regions at the repository horizon approximate the actual repository layout.) Maps showing six regions can be compared by comparing coordinate grids. (UZ—unsaturated zone; SZ—saturated zone; LTA—long-term average)

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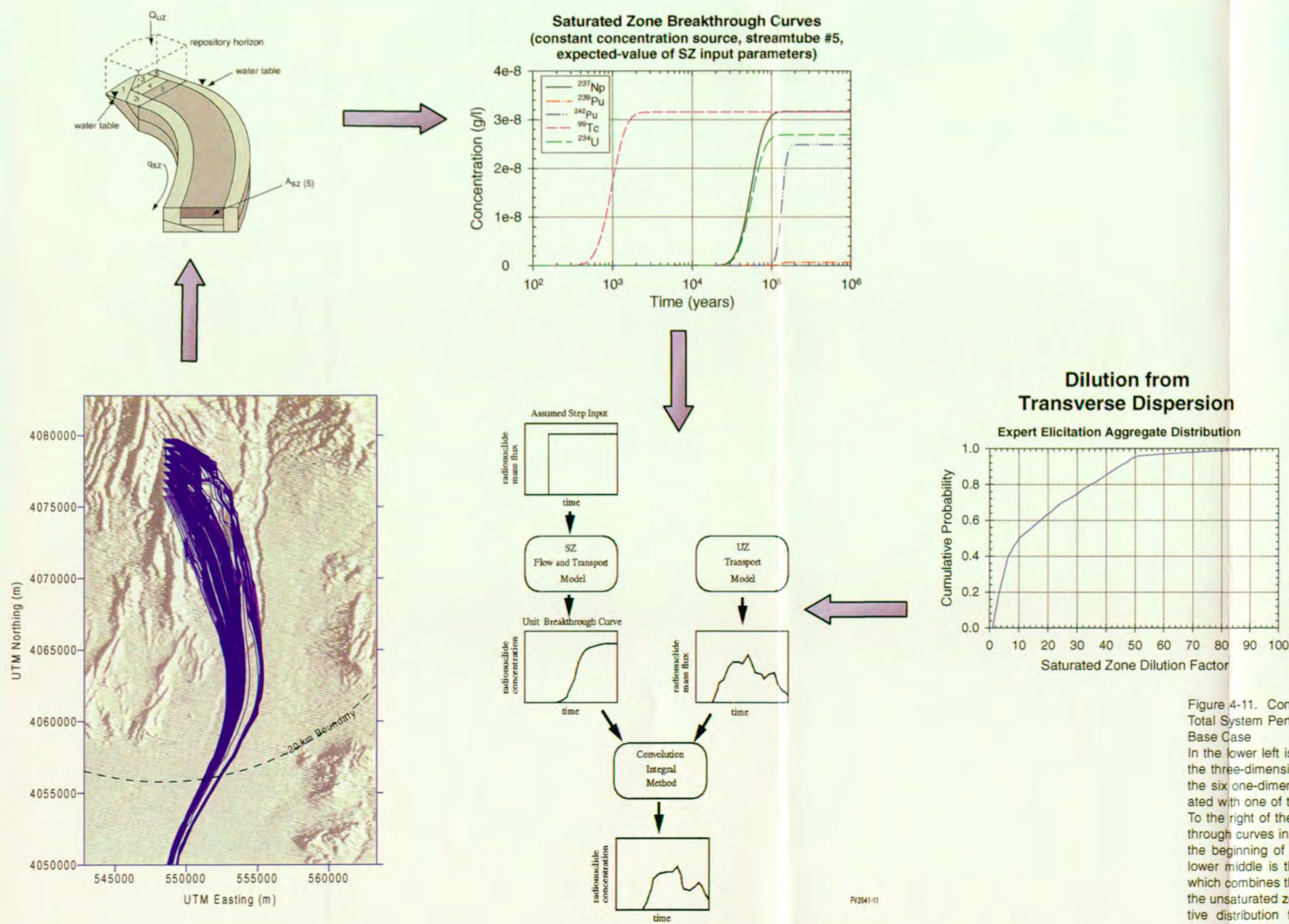

Figure 4-11. Conceptualization of Saturated Zone Transport for the Total System Performance Assessment for the Viability Assessment Base Case

In the lower left is a plan view of the particle pathlines predicted by the three-dimensional flow model. In the upper left is a diagram of the six one-dimensional streamtubes, showing how each is associated with one of the six water table regions depicted in Figure 4-10. To the right of the streamtubes is a plot of 20-km (12 miles) breakthrough curves in the saturated zone for a constant concentration at the beginning of streamtube #5, for five key radionuclides. In the lower middle is the flow chart for the convolution integral method, which combines the breakthrough curves with the mass release from the unsaturated zone. In the lower right is the dilution-factor cumulative distribution function applied to the concentration at 20 km (12 miles). (SZ—saturated zone)

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Table 3-24). The dose rate to the individual (millirems per year) for a given radionuclide is the product of the biosphere dose conversion factor and the saturated zone concentration. Section 3.8 discussed the parameters used in the biosphere modeling. The expected-value biosphere dose conversion factors for the nine radionuclides considered in the TSPA-VA base case, under long-term average climatic conditions, are as follows:[16]

- Carbon-14 = 1.23×10^7 (mrem/year)/(g/m^3)
- Iodine-129 = 8.24×10^4 (mrem/year)/(g/m^3)
- Neptunium-237 = 4.60×10^6 (mrem/year)/(g/m^3)
- Protactinium-231 = 6.56×10^8 (mrem/year)/(g/m^3)
- Plutonium-239 = 2.72×10^8 (mrem/year)/(g/m^3)
- Plutonium-242 = 1.56×10^7 (mrem/year)/(g/m^3)
- Selenium-79 = 9.30×10^5 (mrem/year)/(g/m^3)
- Technetium-99 = 5.29×10^4 (mrem/year)/(g/m^3)
- Uranium-234 = 2.22×10^6 (mrem/year)/(g/m^3)

4.2 DETERMINISTIC RESULTS OF THE TOTAL SYSTEM PERFORMANCE ASSESSMENT BASE CASE

The single realization described in this section is the outcome of sampling all uncertain input parameters in the TSPA-VA component models at the expected value of their ranges (i.e., at their mean or average value). The intent is to illustrate how total system behavior (i.e., individual dose rate) is influenced by the various component or subsystem models and parameters. The method is to compare time history plots of key subsystem parameters (e.g., waste package failure history) with the time history plots of both dose rate and release rate at various system boundaries (e.g., the bottom boundary of the unsaturated zone—the water table).

Figure 4-12 illustrates the dose rate versus time from all biosphere dose pathways to an average adult (see Section 3.8) residing 20-km (12-miles) downgradient of the repository, for the expected-value simulation. The three graphs in this figure correspond to three different time frames:

- The first 10,000 years after repository closure
- The first 100,000 years after repository closure
- The first 1 million years after repository closure, which is the likely maximum geologically stable time period according to the National Research Council panel on Yucca Mountain performance standards (National Research Council 1995, p. 72)

The shape and timing of various dose histories in these plots is explained in subsequent figures in this section. Key points about the three plots in Figure 4-12 are the following:

1. Within the first 10,000 years, the only radionuclides to reach the biosphere are the nonsorbing radionuclides with high inventories, technetium-99 and iodine-129, and the total peak dose rate is about 0.04 mrem/year.
2. Within the first 100,000 years, the weakly sorbing radionuclide neptunium-237 begins to dominate doses in the biosphere at about

[16] Justification for the use of only these nine radionnclides is presented in Chapter 6, Appendix C of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

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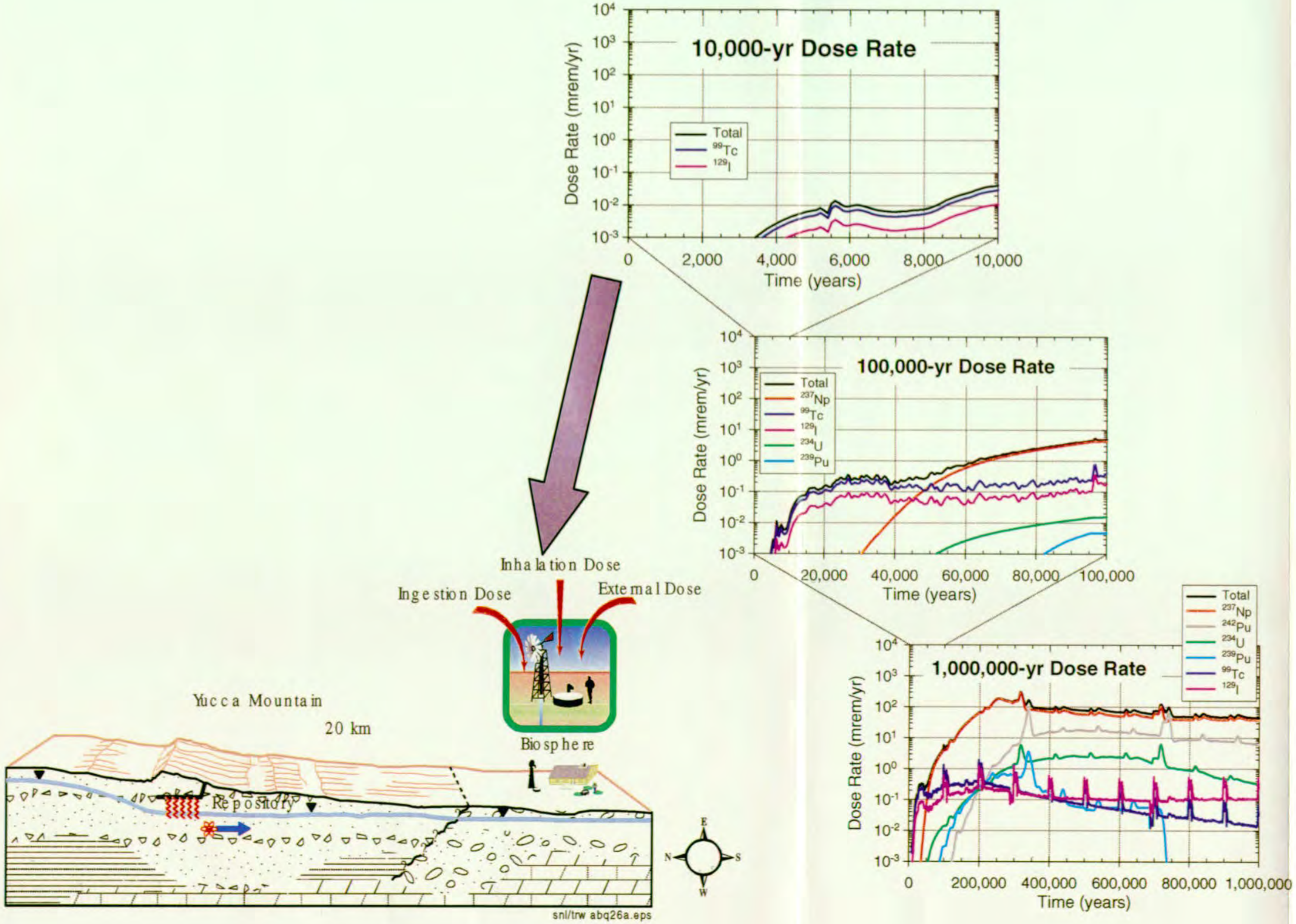

Figure 4-12. Dose Rate to an "Average" Individual Withdrawing Water from a Well Penetrating the Maximum Plume Concentration in the Saturated Zone 20 km (12 miles) Downgradient from the Repository

This figure shows the most important radionuclides in different time periods: technetium-99 and iodine-129 within the first 10,000 years; technetium-99 and neptunium-237 within the first 100,000 years; and neptunium-237 and plutonium-242 within the first 1 million years.

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50,000 years, with the total dose rate reaching about 5 mrem/year.

3. Within the first 1 million years, neptunium-237 continues to be the major contributor to peak dose rate, which reaches a maximum of about 300 mrem/year at about 300,000 years after closure of the repository, just following the first climatic superpluvial period.[17] The radionuclide plutonium-242 is also important during the 1 million-year time frame and has two peaks at about 320,000 years and 720,000 years, closely following the two superpluvial periods. There are regularly spaced spikes in all the dose rate curves (more pronounced for nonsorbing radionuclides such as technetium-99 and iodine-129) corresponding to the assumed climate model for the expected-value base case simulation, as shown in Figure 4-1. As described below, these spikes are a result of assumed abrupt changes in water table elevation and seepage through the packages.

4.2.1 Ten-Thousand-Year Dose Rates

Within the first 10,000 years after closure, technetium-99 and iodine-129 are the dominant radionuclides to reach the biosphere. Relatively large inventories of technetium-99 and iodine-129 are emplaced in the commercial spent nuclear fuel packages. Following are three important qualities of these two radionuclides:

- High solubility in the Yucca Mountain pore water seeping into the waste packages,[18] which allows rapid release from the waste packages and the engineered barrier system compared to releases of the actinide elements, that is, uranium, neptunium, curium, and americium

- Negligible sorption onto the tuff rock matrix in the natural barriers, which allows relatively rapid transport through the natural system compared to other, more chemically reactive, elements[19]

- Relatively slow decay compared to the 10,000-year time span[20]

Because both technetium-99 and iodine-129 behave similarly with respect to these three key attributes, only the movement of technetium-99 needs to be tracked through the system to explain the behavior of both ions, and the behavior of the total dose rate within 10,000 years.

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[17] This dose rate is comparable to the annual mean background radiation in the United States from natural sources (NCRP 1987, p. 149; National Research Council 1990, p. 19). The dose rate shown in Figure 4-12 reaches a peak after the superpluvial period (see Figure 4-1) because of a sorption delay affecting neptunium-237 transport in the saturated zone. Also, as discussed in Chapter 6, Appendix C of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i), the predicted total dose rate of 300 mrem/year would perhaps increase by 15 to 20 percent if the daughter products of neptunium-237 (uranium-233 and thorium-229) had been included in the model.

[18] The composition of the water seeping into the waste packages may be altered from ambient (i.e., pre-repository) conditions because of thermal-chemical processes, including evaporation and recirculation during the early high-temperature period and chemical reaction with the emplaced in-drift, waste package, and waste form materials. Thus, the use of ambient pore-water solubility in the radionuclide mobilization model in TSPA-VA is an approximation. See the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) for clarification of this modeling assumption.

[19] Both technetium and iodine are assumed to be dissolved in the pore water in their anionic, oxidized forms as pertechnetate (TcO_4^-) and iodide (I^-). Under ambient or chemically unaltered conditions in Yucca Mountain, anions have a low affinity for sorption onto the volcanic tuff rocks, which implies that they can travel more rapidly through the unsaturated zone than other radionuclides that are in a cationic, or positively charged, state in the oxidized groundwater. Furthermore, pertechnetate and iodide have a lower potential for incorporation into minerals via precipitation than some other elements, such as the actinide elements, particularly under oxidizing conditions.

[20] Technetium-99 has a half life of about 213,000 years and iodine-129 about 15,700,000 years.

Figures 4-13 through 4-15, discussed in detail in this section, show the primary causes for the shape and timing of the 10,000-year dose rate graphs for technetium-99 and iodine-129. Two primary sets of plots are shown:

- Time histories of activity release and radionuclide concentrations at the downgradient boundaries of various parts of the natural and engineered barriers (e.g., engineered barrier system, unsaturated zone, and saturated zone)

- Time histories of key subsystem parameters

Figure 4-13 shows the performance of the engineered barrier system. It is evident from Figure 4-13 that the initial releases of technetium-99 (and also iodine-129 and carbon-14) from the waste packages and the engineered barrier system are caused by the first package failure at 1,000 years, which is an assumed juvenile, or early, failure arising from possible manufacturing defects, rockfall, or seismic activity. This failure is assumed to occur in the southeast repository region, the largest in area of the six regions (see Figure 4-4). The small initial peak and oscillation in release from the engineered barrier system in the southeast region at about 1,000 years is caused by the initial gap fraction release from the single juvenile-failure package. The gap fraction is the 2 percent of the total technetium-99 inventory residing in very soluble form in the gap between the fuel matrix and the cladding.

After the initial juvenile failure at 1,000 years, the first corrosion failures occur at about 4,100 years and 4,200 years (the first one in the southeast region and second one in the central-central region), indicated by the bar heights equal to 1 in the histogram plot of package failures. After that, more packages fail by corrosion in all the various regions starting at 5,000 years. Besides this waste package failure history, the other key event influencing performance in the 10,000-year time frame is the abrupt climatic change from current dry conditions to the relatively wetter long-term average climate, occurring at 5,000 years in this expected-value realization. This climatic change is evident in both the waste package and engineered barrier system mass-release graphs and causes the abrupt peak in releases at 5,100 years. However, this mass-release peak originates almost solely from the single juvenile-failure package and not from the two corrosion-failed packages.[21] For example, the advective/diffusive release plot shows that the two corrosion-failed packages at 4,100 and 4,200 years have a negligible influence on either total[22] advective or total diffusive mass-release curves prior to the climate change at 5,000 years.[23]

This lack of influence is a result of the very small initial patch area per package, shown graphically in Figure 4-13. This small area means a low diffusive flux out of the corrosion-failed packages and a low seepage flux into the corrosion-failed packages. The latter causes low advective releases during the dry climate (Figure 4-13).[24] (The seepage flux

[21] The juvenile-failure package has the same clad failure fraction initially (about 1.25 percent) as the corrosion-failed packages.

[22] "Total" as used here means the total releases from all waste packages in the entire repository.

[23] During the dry climate in the first 5,000 years, the technetium-99 releases from advection and diffusion are about the same from the juvenile-failure package and add up to the total release curve from the engineered barrier system. Later, in the 100,000-year and 1 million year time frames (see Figures 4-16 and 4-21), diffusion dominates over advection by about a factor of 10 for technetium-99 releases during the dry climate periods. This is because technetium-99 is not released by advection from the package at nearly as high a rate during the dry climate compared to the long-term average climate. Therefore, a higher concentration of technetium-99 builds up in the waste package during the dry climate, producing a much higher concentration gradient from the inside to the outside of the package. This concentration gradient is the force that produces diffusive releases. However, at very early times, from 1,000 to 5,000 years, the patch area on the single juvenile-failure package is too small to allow diffusive releases to domiuate advective releases (compared to the larger patch area available on corrosion-failed packages at later times).

[24] On the seepage flux plot, the red line iudicates the average seepage flux through the packages, averaged over all six repository regions. The six gray lines (hard to see individually because they overlap) are the seepages into the six individual regions. Similarly, for the seepage into the drift, the blue line represents an areal average over the regions.

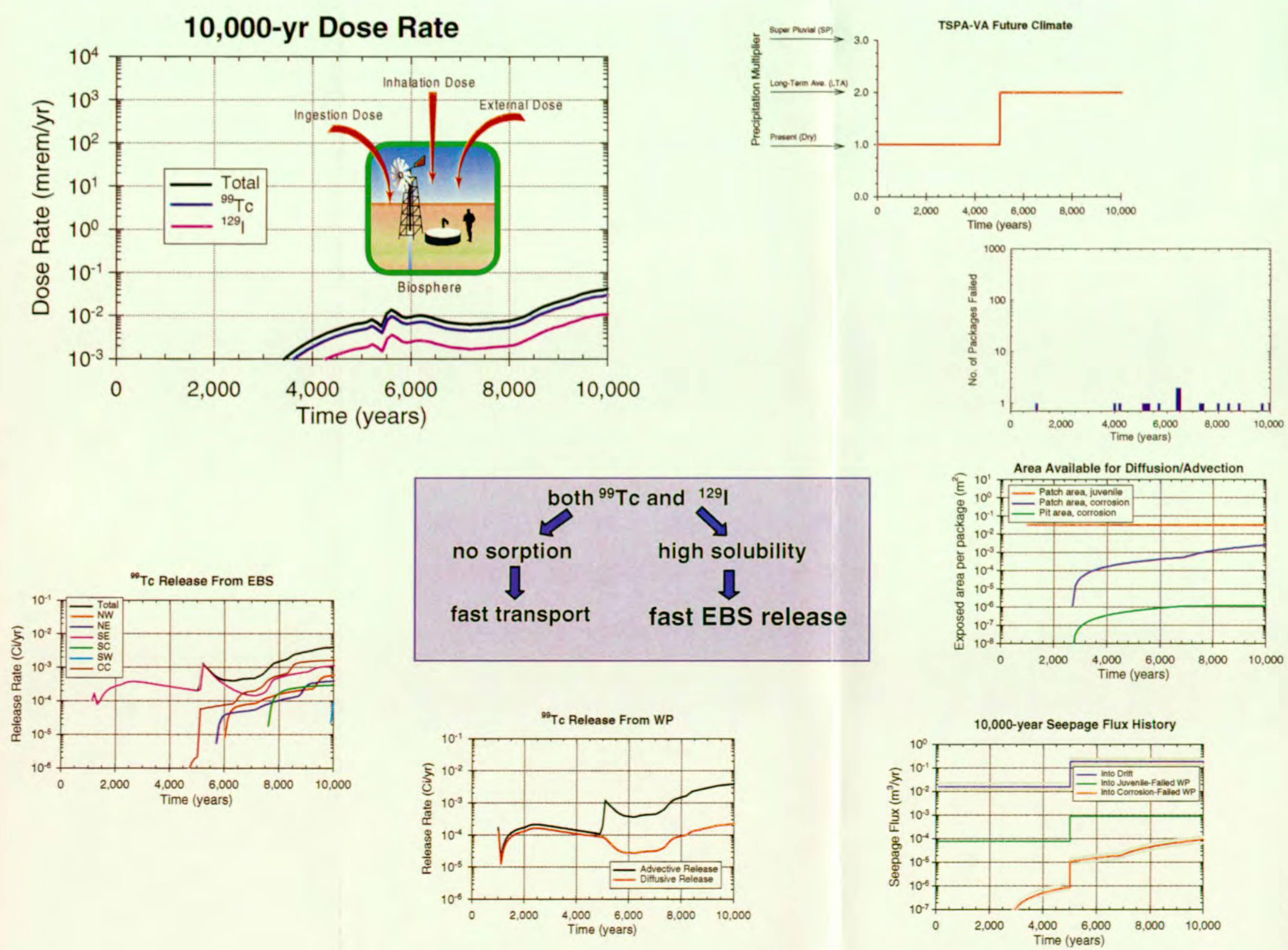

Figure 4-13. Effects of Climate Change, Seepage Flux, and Waste Package Degradation on Engineered Barrier System Releases and Dose Rates in the First 10,000 Years After Waste Emplacement
(EBS—engineered barrier system; WP—waste package)

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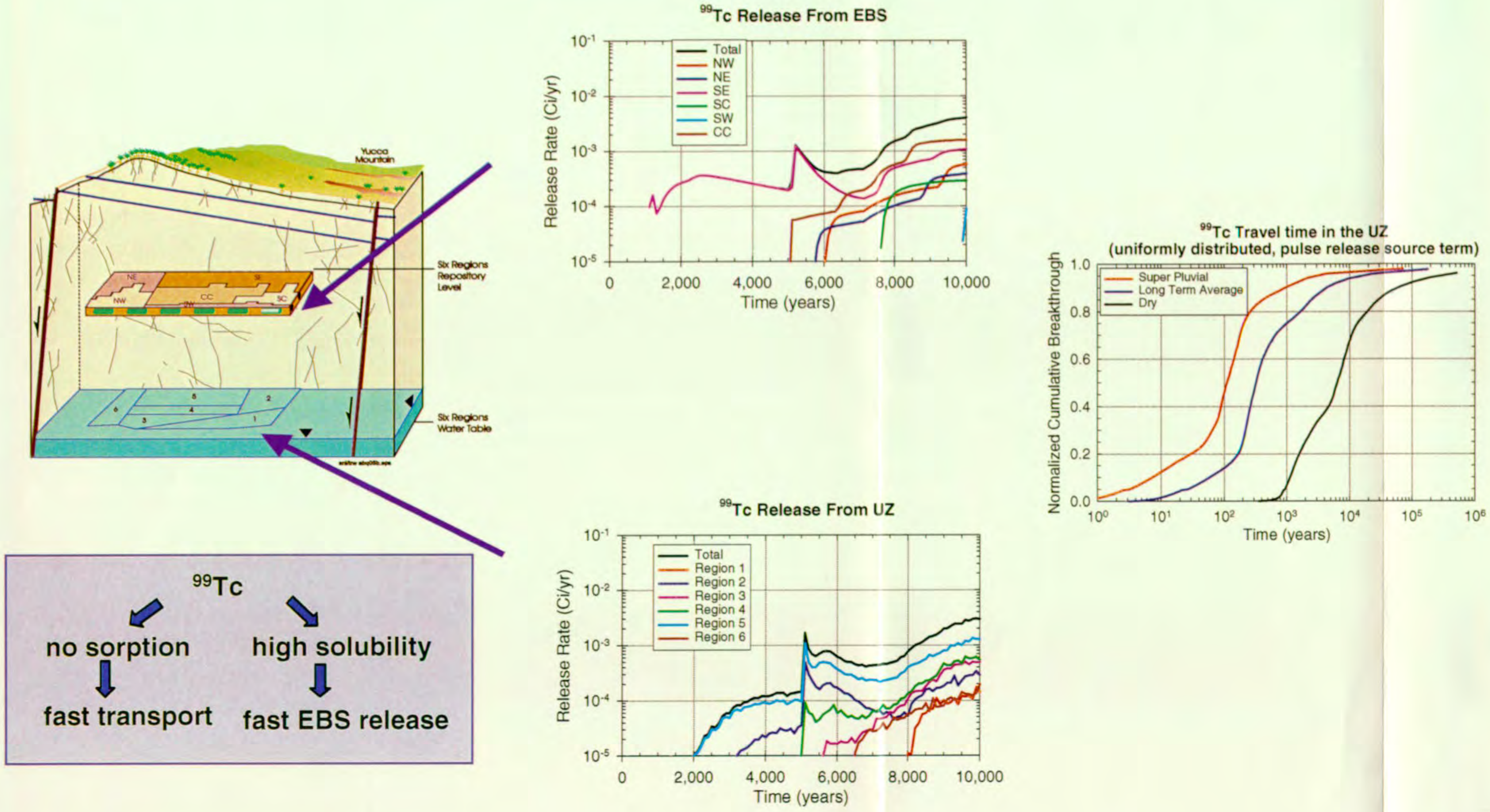

Figure 4-14. Performance of the Unsaturated Zone with Respect to Technetium-99 During the First 10,000 Years After Waste Emplacement
Shown is the effect of different climate states, the engineered barrier system released from various repository regions, and the unsaturated zone releases to the various water table regions. (UZ—unsaturated zone; EBS—engineered barrier system)

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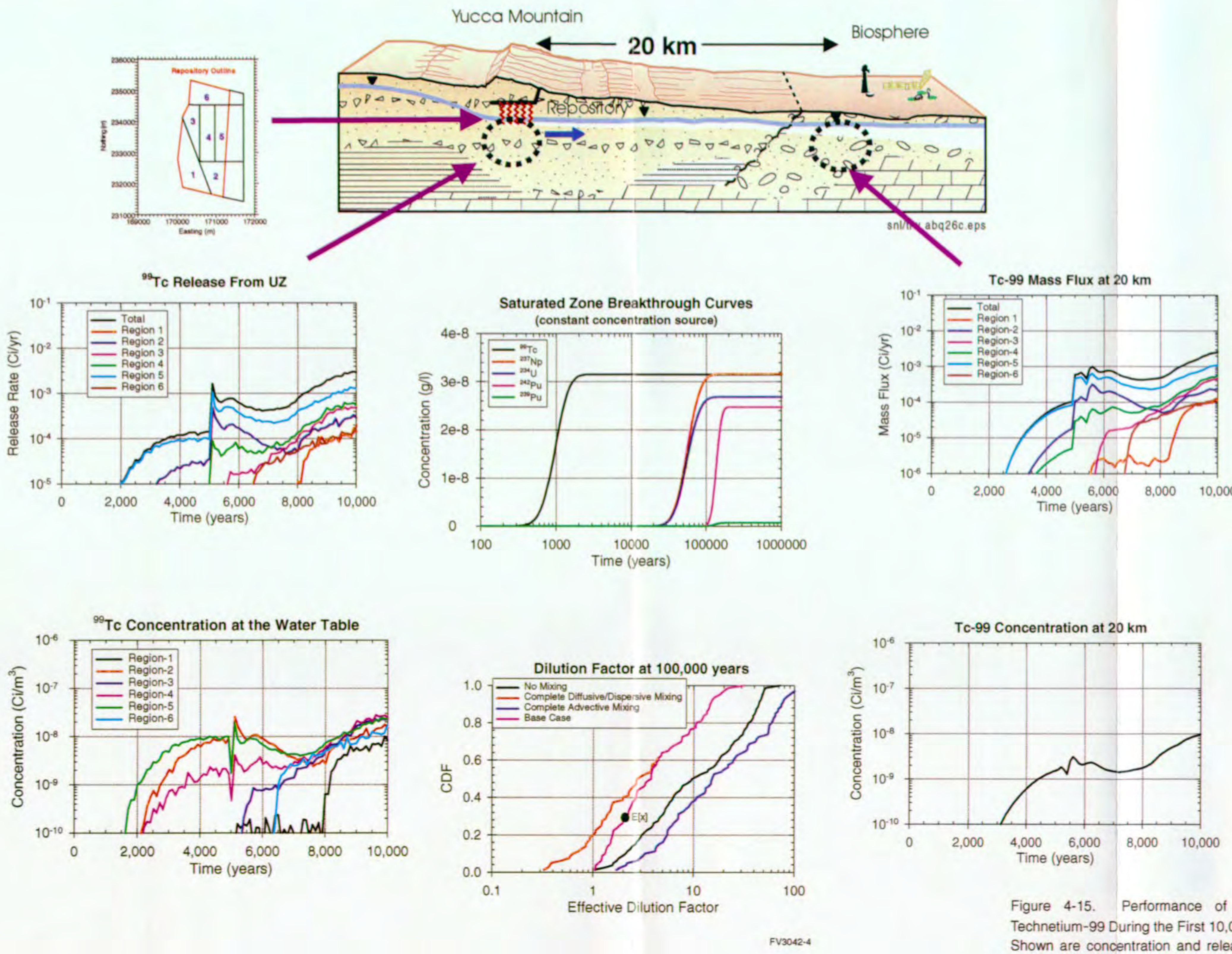

Figure 4-15. Performance of the Saturated Zone with Respect to Technetium-99 During the First 10,000 Years After Waste Emplacement
Shown are concentration and releases in six stream tubes (each associated with one of six regions at the water table level) and the impact of dilution in the saturated zone. Shown in the upper left corner are the six water table regions. (UZ—unsaturated zone; CDF—cumulative distribution function)

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into the drift is the same for both the juvenile package and the corrosion packages, but the seepage into and through the packages is different because of the different patch area.) The single juvenile-failure package was conservatively assumed to have a larger failed area (0.031 m^2, which stays constant with time) than the initial corroded patch area per package for the corrosion failed packages. The patch area on corrosion failed packages eventually equals the failed area of the juvenile failure package at about 30,000 years.

At 5,000 years, the climate becomes wetter and the seepage flux through both the juvenile-failure package and the corrosion-failed packages abruptly increases, as shown on the plot of seepage flux. This increase produces a burst of technetium-99 release from the repository at 5,000 years, shown by the peak in the technetium-99 advection curve. The diffusive release curve drops off during the long-term average climate because the removal of technetium-99 by advection decreases the concentration in the waste package. The burst in technetium-99 release at 5,000 years is caused almost solely by the greater flushing of the juvenile failure package, with only a negligible contribution from the corrosion-failed packages. The contribution from the corrosion-failed packages is very small because the patch area and seepage flux are low enough that the solubility limit for technetium-99 becomes important for those packages.

Figure 4-13 has shown the major factors controlling the 10,000-year repository performance. These factors are related to the engineered barrier system, primarily waste package performance and seepage. Some additional contribution to performance is gained within the natural geologic setting, that is, the barriers of the unsaturated and saturated zones, but primarily only during the dry climate up to 5,000 years. This additional performance is caused by delay in transport because of slow liquid flow through the 300 m of unsaturated zone between the repository and the water table. It is also caused by the relatively long (20 km, or 12 miles) travel distance in the saturated zone—from the repository footprint at the water table to the water withdrawal location. On the other hand, in the long-term average climate, the delay because of travel time in the unsaturated zone is greatly reduced by the higher overall infiltration flux, from an average of about 7 mm/year to an average of 40 mm/year. The delay is also influenced by the increase in proportion of the unsaturated zone travel path associated with fracture flow and the much greater proportion of fracture flow (compared to matrix flow). The delay during the long-term average climate is only about 300 years, based on the 50 percent point of the breakthrough curve.

Figure 4-14 shows the effect of the unsaturated zone on technetium-99 activity releases. The fast transport of nonsorbing radionuclides is apparent. However, comparing the total activity release from the engineered barrier system with the total activity release from the bottom of the unsaturated zone does not provide a measure of the technetium-99 travel time through the unsaturated zone because the engineered barrier system releases have only been occurring over a 4,000-year period prior to the switch to long-term average climate. Thus, the full time needed for completely traversing the 300 m of unsaturated zone during a dry climate is not realized before the switch to long-term average climate. This fact is shown in the pulse-release, breakthrough-curve plot shown in Figure 4-14 for the three climate states. The 50 percent point on the breakthrough curves requires about 6,000 years in the dry climate and 300 years in the long-term average climate. Another point with regard to these activity release plots is that the sharp peak in the total unsaturated zone release curve at 5,100 years is *not* caused by the transport of the sharp peak in the total engineered barrier system release curve. Rather, it is caused by a rapid (modeled as instantaneous) 80-m rise in the water table when the climate changes from dry to long-term average. This rise causes an instantaneous burst of technetium-99 into the saturated zone from the 80 m of unsaturated zone pore water, which produces the activity-release peak at the base of the unsaturated zone.[25]

Examination of the engineered barrier system release plot in Figure 4-14 also reveals that during the period from 5,000 to 10,000 years, packages have failed by corrosion in all six repository regions. The failures can be seen in the curves for the six individual regions as they begin to appear at various times corresponding to initial corrosion failures of packages in the various regions. The southwest region, being the smallest and driest region, has the longest delay in initial failures, with the release curve appearing just before 10,000 years.

Further comparison of the engineered barrier system release plot with the unsaturated zone release plots reveals some of the behavior of the unsaturated zone between the repository and the water table. The initial juvenile failure of a waste package has occurred in the southeast repository region, which approximately overlies regions 2, 4, and 5 at the water table (see Figure 4-4). Therefore, during the period of release from the juvenile-failure package (1,000 to 5,000 years), the radionuclide mass arrives at the water table only in saturated zone regions 2, 4, and 5. This arrival location is also because of the generally southeasterly direction of lateral diversion in the unsaturated zone as described in Sections 3.1 and 3.6.

Similarly to the unsaturated zone, the saturated zone provides its best performance during the dry climate periods that occur approximately every 90,000 years on average (see Section 3.1). During these dry periods, the delay in the saturated zone for nonsorbing radionuclides such as technetium-99 and iodine-129 is about 1,000 years (Figure 3-72). During the long-term average climate, the delay is equal only to 1000/5.44 = 184 years, where 5.44 is the flux multiplier used for the long-term average climate relative to the dry climate (Section 3.7).[26] The performance of the saturated zone during the first 10,000 years after closure is illustrated in Figure 4-15, which compares activity releases at the base of the unsaturated zone, or entry to the saturated zone, to activity releases at the end of the saturated zone, at 20 km (12 miles). Seeing the time delay in either the dry or long-term average climate from these two activity release figures is difficult. However, comparing the temporal location of the water-table-rise peak on the unsaturated zone release figure with the location on the saturated zone release figure indicates that the peak is delayed by at least the expected 200-year travel time in the long-term average climate. In the dry climate, all that can be compared is the temporal location of a specific value of the activity release rate (e.g., the 10^{-4} Ci/year rate). This location is at about 3,600 years on the unsaturated zone release plot and about 4,700 years on the saturated zone release plot, which demonstrates the approximate 1,000-year delay during the dry climate (shown in Figure 4-15 by the technetium-99 breakthrough curve, for a constant-concentration source).

The other key feature of saturated zone performance is the dilution it provides because of plume dispersion. This dilution is illustrated on Figure 4-15 by the two concentration plots, one at the beginning of the saturated zone for the six regions, or streamtubes, and one at the end where a single concentration value is chosen for determining dose rate. An explanation of the choice for saturated zone concentration has already been given in Section 3.7.2.3. Here the explanation is briefly summarized with the plot of effective dilution factor shown in Figure 4-15. In particular, the dilution factor range of 1 to 100 was based on the *Saturated Zone Expert Elicitation* (CRWMS M&O 1998g), which assumed a plume dimension of approximately the size of each of the streamtubes in the TSPA-VA saturated zone model. In reality, there are six saturated zone streamtubes adjacent to one another. Therefore, at high dilution factors within the 1-to-100 range, the radionuclides will overlap and mix by diffusion, which can be approximated by adding the concentrations in the six streamtubes to determine a concentration to use for calculating dose. This is the appropriate scenario for the expected value case, which assumes a dilution factor of 10 in each individual streamtube. In this case, the diffusional mixing method results in an effective dilution factor of

[25] The sharp peak in the engineered barrier system curve at 5,200 years is transported through the unsaturated zone and appears in the unsaturated zone release curves as a subdued peak at about 5,600 years.

[26] The superpluvial climate flux multiplier is 14.7.

only four, as shown by comparing the concentration plots at the beginning and end of the saturated zone. The cumulative distribution function plot for the effective dilution factor can also be examined, and it shows that the median probability of 0.5 corresponds to a dilution factor of four. (This median value is the median of a 100-realization run [see Section 4.3], so this dilution factor is not exactly the same as the expected-value run being described in this section, though it is nearly the same.) For low dilution factors, the saturated zone model chooses the maximum of the six streamtubes because this is effectively the case of no or very little mixing between the streamtubes. For the expected-value run, this choice would cause a dilution of 10; however, as previously mentioned, a value of 10 is high enough to invoke the diffusive mixing scenario. For completeness, the effective dilution plot also shows the case of advective mixing of the six streamtubes, which means averaging the concentration of all six streamtubes, and corresponds to the scenario of a well intersecting all six streamtubes and then having their waters mixed in the wellbore.

A final summary point about the 10,000-year dose rates is that only nonsorbing radionuclides such as technetium-99 and iodine-129 reach the biosphere because most of the other radionuclides react to some degree with the tuffaceous rocks in the geosphere. These radionuclides are delayed enough by these reactions that they do not reach the biosphere during the first 10,000 years after repository closure.

4.2.2 One-Hundred-Thousand Year Dose Rates

Within the first 100,000 years after closure, technetium-99 and neptunium-237 are the dominant radionuclides to reach the biosphere; technetium-99 dominates dose rates up until about 50,000 years, after which neptunium-237 finally reaches the biosphere and dominates thereafter. There are relatively large inventories of both radionuclides in the commercial spent nuclear fuel packages. The three important qualities about technetium-99 were pointed out in the previous section: high solubility in Yucca Mountain porewater, no sorption on the volcanic tuffs, and relatively slow decay. Following are three key qualities about neptunium-237:

- Relative low solubility in the porewater seeping into the waste packages,[27] implying that neptunium-237 mass release from failed waste packages is "solubility-limited" until the failed inventory is nearly depleted. In other words, the neptunium-237 release rate from the packages will be linearly proportional to the liquid flow rate through the failed packages.

- Weak, but nonzero, sorption onto the tuff rock matrix in the natural barriers, which means a lower transport rate through the unsaturated and saturated zone than for technetium-99 (about a factor of 20 to 80 times slower in the rock matrix, but the same through fractures). However, the neptunium-237 transport rate is still relatively rapid compared to more chemically reactive elements.[28]

- Relatively slow decay compared to the 100,000-year time span.[29]

In the following discussion, mainly the behavior of these two key radionuclides, technetium-99 and neptunium-237, is tracked, but the behavior of plutonium-239 is also discussed. There is also a large inventory of plutonium-239, but the radionuclide has a much higher sorption than neptunium, implying much slower transport through rock

[27] As with technetium-99, the solubility used for neptunium-237 is the "ambient" solubility in pre-repository pore water.

[28] Because the pore waters are assumed to be oxidizing, neptunium is thought to be present in ionic form mainly in the Np^{+5} oxidation state, as the NpO_2^+ ion at ambient pH levels (below 8).

[29] Neptunium-237 has a half life of about 2,140,000 years.

matrix. However, because plutonium-239 has been observed in colloidal form in recent measurements at the Nevada Test Site, caused by rapid migration in the groundwater away from underground nuclear test detonations, it is important to consider its possible future behavior at Yucca Mountain.[30]

Figures 4-16 through 4-20, discussed in detail in this section, show the primary causes for the shape and timing of the 100,000-year technetium-99 and neptunium-237 dose rate graphs. As in the previous section, two primary sets of plots are displayed: time histories of activity release or concentration at the downgradient boundary of various parts of the natural and engineered barriers (i.e., engineered barrier system, unsaturated zone, and saturated zone); and time histories of key subsystem parameters. Figure 4-16 portrays the 100,000-year performance of the engineered barrier system. Similarly to the 10,000-year performance, the key factors are climate change and waste package failure history. Consider first the waste package and engineered barrier system releases of technetium-99 and also its dose rate in the biosphere. All three of these time histories mirror the jagged nature of the instantaneous waste package failure history. This instantaneous failure curve is a histogram of package failures, with the interval size being equal to the time step size of 333 years.[31] Ten or 11 packages are about the most that fail (by patch penetration) in any 333-year period. The reason that the release-rate curves and dose curve for technetium-99 mirror the package failure rate is that technetium-99 is not solubility limited at the values of seepage flux flowing through the packages after 10,000 years. Therefore, technetium-99 is flushed very rapidly from a failed package. As package inventories become available for release via the failure histogram, their inventories of technetium-99 are almost instantaneously released, which causes this mirroring of releases with package failures.

The releases of neptunium-237 are different from technetium-99 releases because of neptunium-237's low solubility. Low solubility causes the shape of the neptunium-237 release curve to be relatively smooth and controlled by two features: the flow rate of water through the packages and the cumulative amount of failed packages or inventory. The cause and effect of the first feature, seepage flux, is evident when the climate changes from long-term average to dry at 95,000 years. When the dry climate is reestablished, advective releases of neptunium-237 are dramatically reduced because of the linear proportionality between flow rate and mass release for radionuclides that have reached their solubility limit in the water seeping through the waste packages. Furthermore, the neptunium-237 diffusive releases do not abruptly change with the switch from long-term average climate to dry climate at 95,000 years because the accompanying change in seepage flux does not change the neptunium-237 concentration in the packages, which is the driving force behind diffusion.[32] In the dry climate, the technetium-99 advective release rate also drops by about the same factor as the neptunium-237 release rate. The lower release rate gives the appearance that technetium-99 is also solubility limited. However, this is not the case, as shown by the significant increase in diffusive releases for technetium-99 during the dry climate. The explanation for this phenomenon is found in the "seepage fraction" plot shown in Figure 4-16, which shows the fraction of all packages that are exposed to seeps in each of the six repository regions, plus the average over the regions. This average fraction drops from about 0.27 to 0.045 with the switch to dry climate; however, the packages keep failing at about the same rate during the dry climate,[33] continually exposing technetium-99 inventory to contact with water, but in this case stagnant or nonflowing water. Because of the very high solubility of technetium-99 (0.34 g/m^3 or 1.43×10^{-6} *M* for this expected-value realization), the concentration of

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[30] Plutonium-239 has a relatively short half life of about 24,000 years.

[31] If divided by the total number of packages, this instantaneous failure history becomes a probability density function (pdf) for waste package failure as a function of time.

[32] This, in fact, is the key point about reaching a solubility limit: that the amount of water or water flow present does not alter the concentration (mass/volume) of the radionuclide in solution.

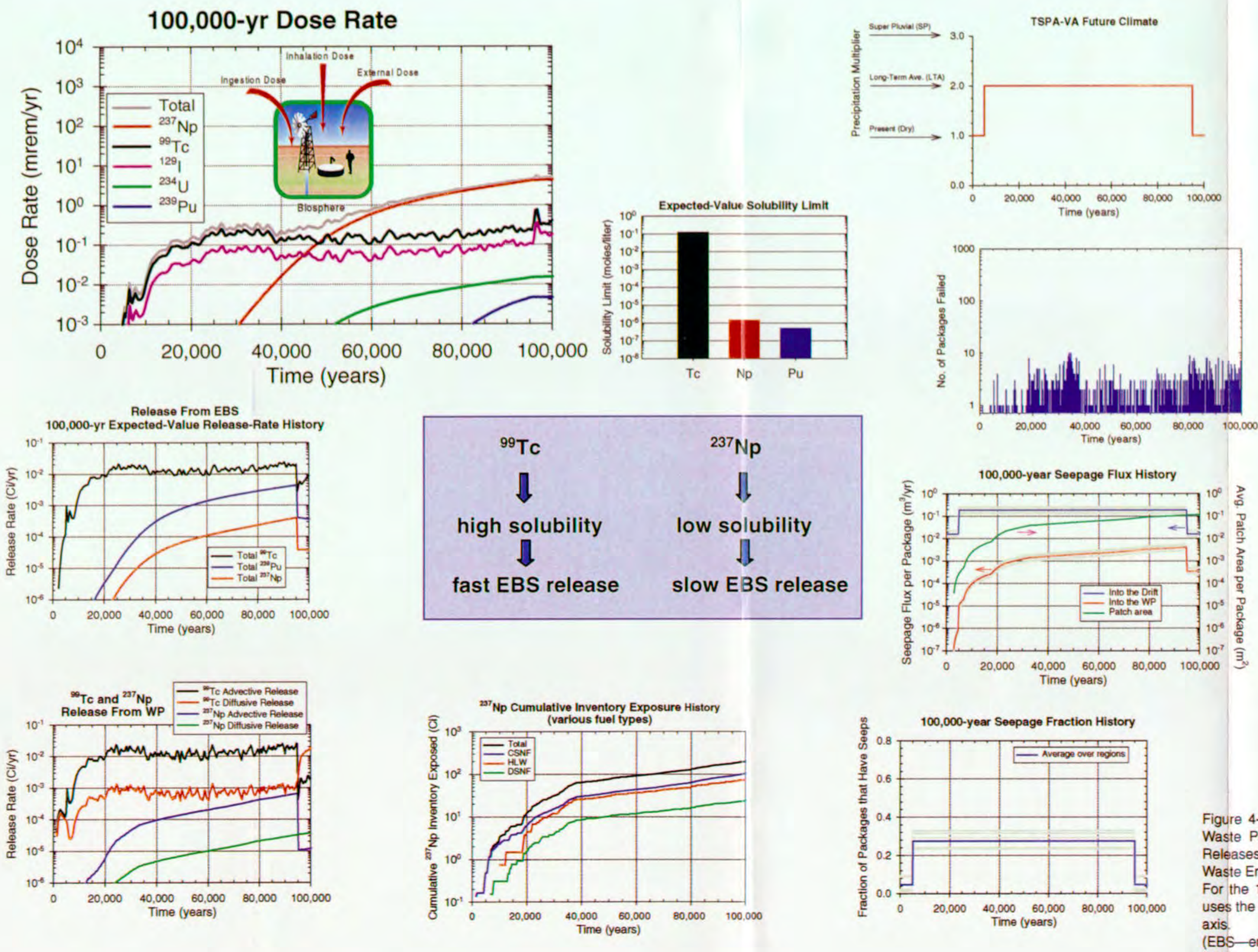

Figure 4-16. Effects of Climate Change, Seepage Flux, and Waste Package Degradation on Engineered Barrier System Releases and Dose Rates in the First 100,000 Years After Waste Emplacement

For the 100,000-year seepage flux plot the patch area curve uses the right-hand axis and other two curves use the left-hand axis.

(EBS—engineered barrier system; WP—waste package; CSNF—commercial spent nuclear fuel; HLW—high-level radioactive waste; DSNF—DOE spent nuclear fuel)

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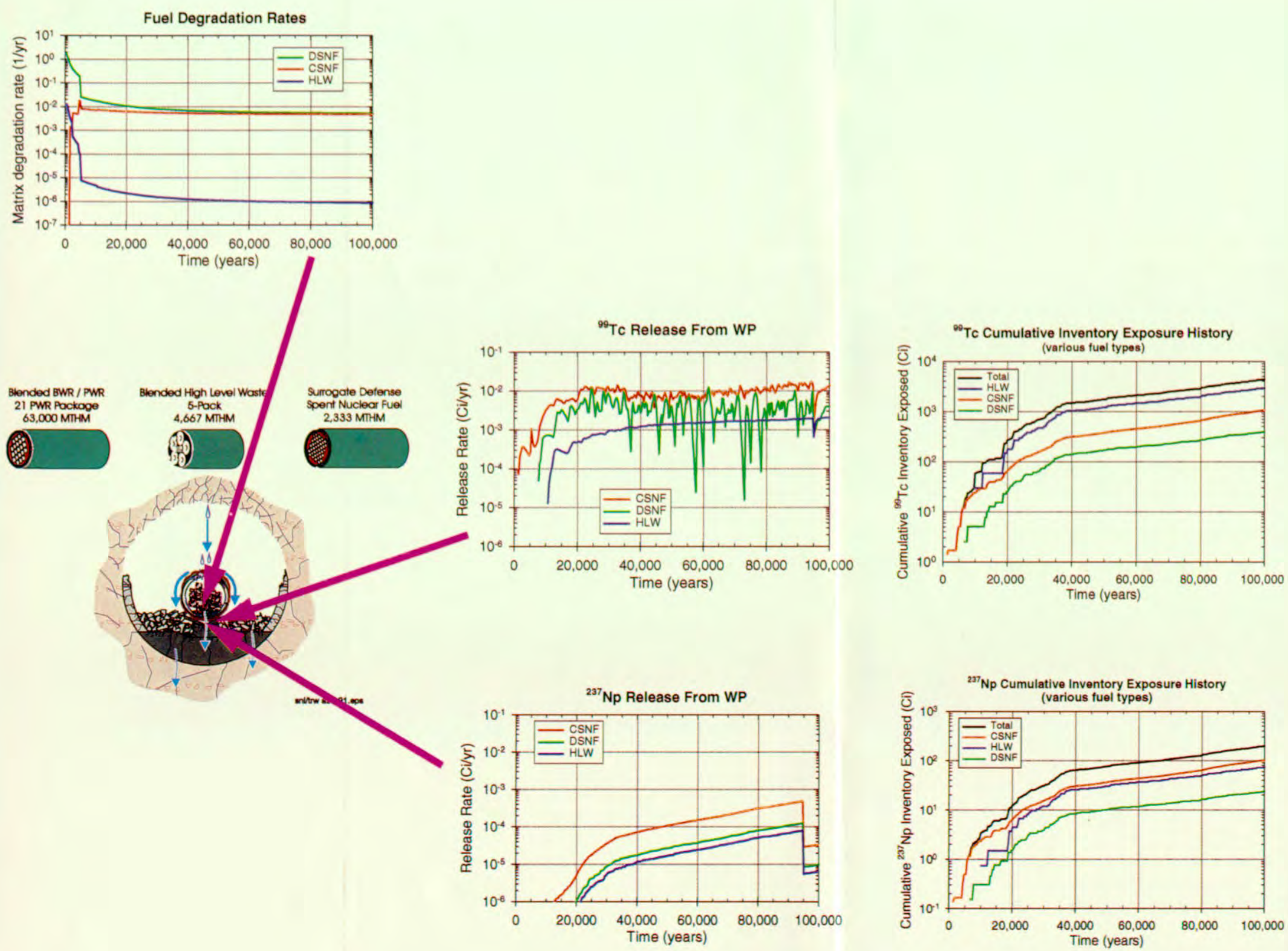

Figure 4-17. Effects of Matrix Degradation Rate and Inventory Exposure Rate on the Releases of the Three Different Fuel Types: Commercial Spent Nuclear Fuel, U.S. Department of Energy Spent Nuclear Fuel, and High-Level Radioactive Waste (WP—waste package; CSNF—commercial spent nuclear fuel; DSNF—DOE spent nuclear fuel; HLW—high-level radioactive waste)

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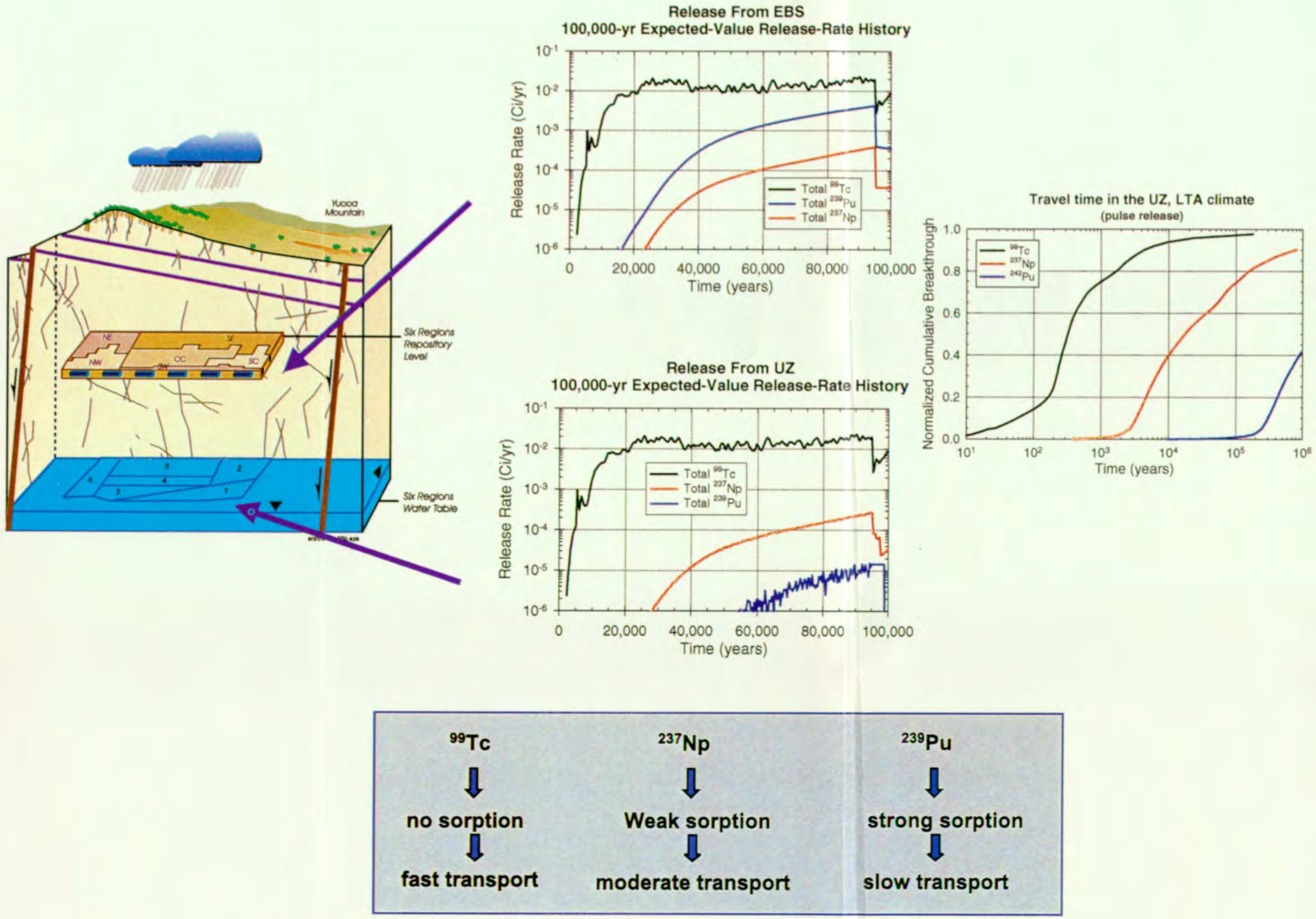

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Figure 4-18. Performance of the Unsaturated Zone During the First 100,000 Years After Waste Emplacement with Respect to Technetium-99, Neptunium-237, and Plutonium-239
Shown is the importance of sorption in the unsaturated zone. (EBS—engineered barrier system; LTA—long term average; UZ—unsaturated zone)

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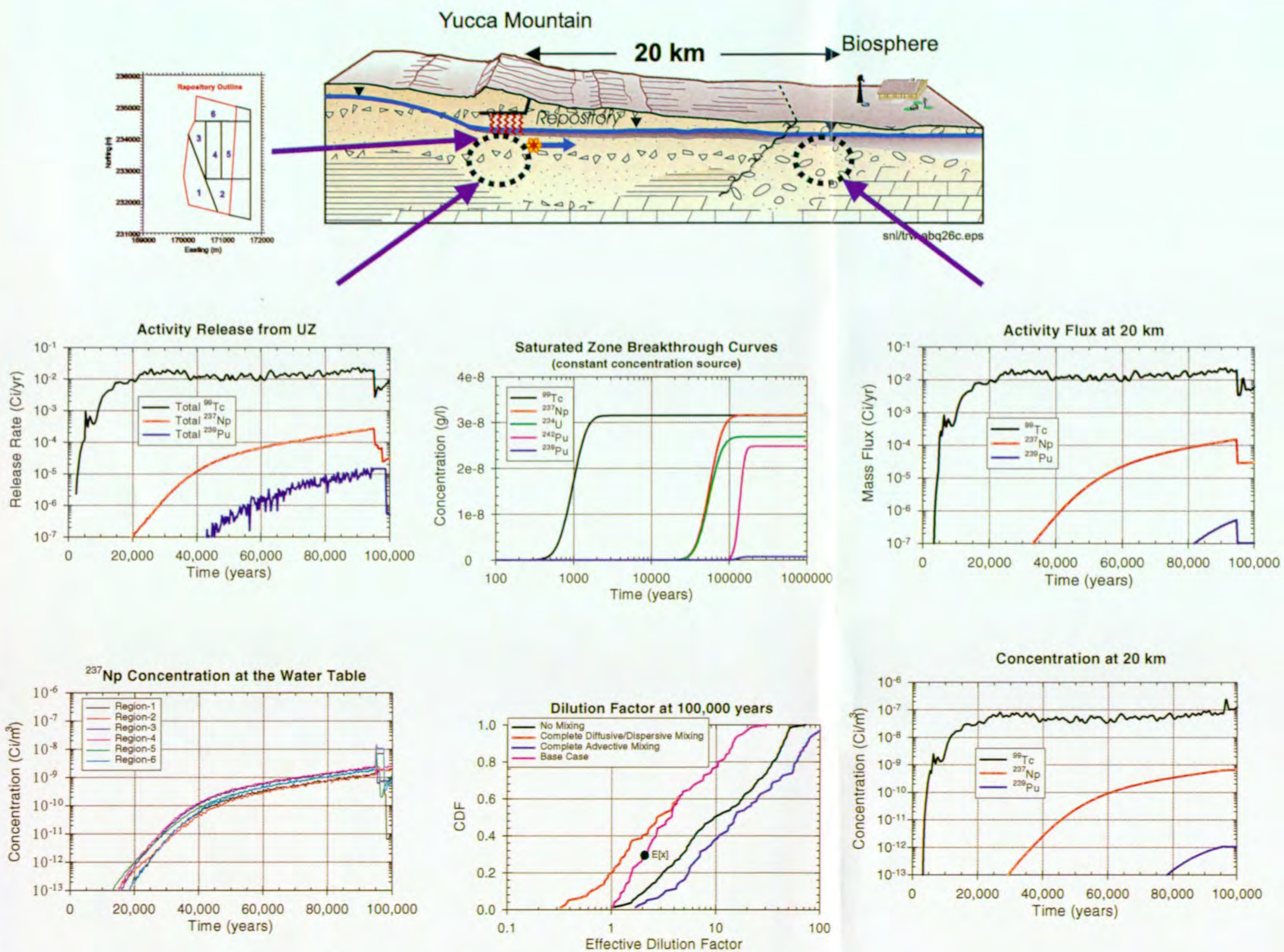

Figure 4-19. Performance of the Saturated Zone During the First 100,000 Years After Waste Emplacement with Respect to Technetium-99, Neptunium-237, and Plutonium-239
Shown is the importance of sorption and dilution in the saturated zone. (UZ—unsaturated zone; CDF—cumulative distribution function)

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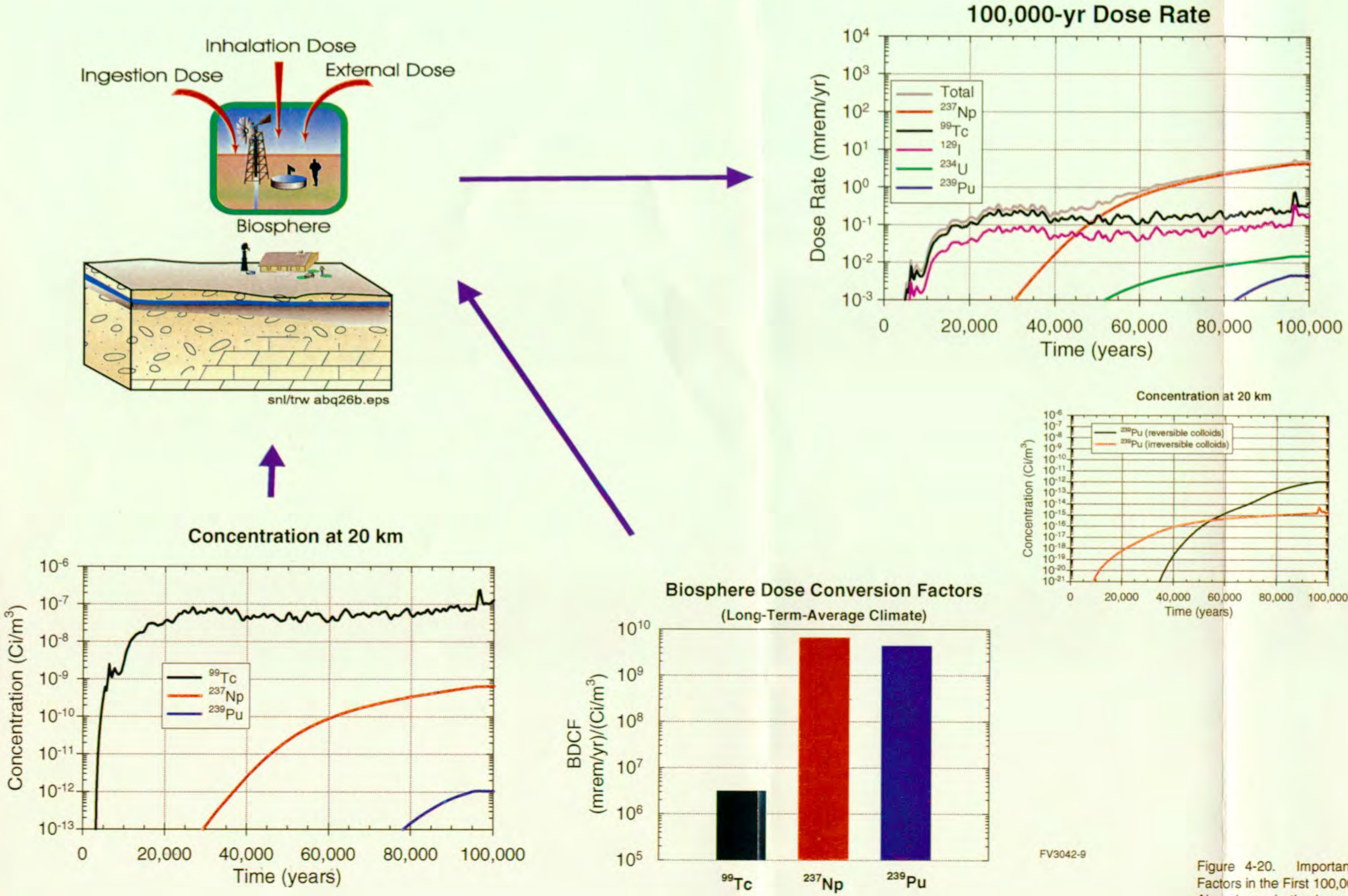

Figure 4-20. Importance of Biosphere Dose Conversion Factors in the First 100,000 Years After Waste Emplacement Also shown is the impact of colloidal transport of irreversibly sorbed plutonium-239. (BDCF—biosphere dose conversion factors)

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technetium-99 is very high in these failed packages because flowing water is not removing the inventory. A much higher concentration gradient is created between the inside and outside of the packages, causing a much higher diffusive mass release. As discussed in the next section regarding 1 million-year releases, when the climate goes back from dry to long-term average a burst of technetium-99 is released. This burst represents the technetium-99 that has built up during the long-term average climate within the fraction of packages that are not dripped on during the dry period, even though they have failed by corrosion. When this incremental fraction of packages (about 0.225) is suddenly dripped on again during the new long-term average climate, all of the technetium-99 is abruptly transported away. This technetium-99 available for rapid transport has accumulated during the dry climate because the commercial spent nuclear fuel degradation rate is very high. All of the technetium-99 available from this degradation is present in a highly soluble form that can dissolve at a very high concentration in the newly available flowing water during the long-term average climate.

The second factor controlling the neptunium-237 release is the cumulative amount of failed inventory, shown in Figure 4-16 for the three inventory types: commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste. For the latter two, the cumulative failed-inventory curve is found by multiplying the cumulative failed-package curve by the inventory per package.[34] However, for commercial spent nuclear fuel there is a second barrier to consider: the Zircaloy fuel cladding.[35] During the first 100,000 years, this cladding is considered to be very robust in the base case, expected-value realization. At most only about 3 percent of the cladding will fail by various mechanisms by 100,000 years (see Section 3.5), meaning that to obtain the cumulative exposed inventory in commercial spent nuclear fuel, the cumulative package-failure history must be multiplied by both the neptunium-237 inventory per package and by the clad failure fraction.[36] In Figure 4-16 the neptunium-237 cumulative inventory-exposed history has the same shape as both the waste package and engineered barrier system release histories. On the other hand, the seepage flux into the packages also has the same steadily increasing curve (ignoring the climate change at 95,000 years), which is caused by the steadily growing patch area curve. It appears as if either

[33] The fact that packages keep failing at the same rate during the dry climate is a conservative assumption. Within the RIP model, there are four environmental package groupings: packages that are never dripped on (55 percent of all packages, in the expected-value base case); packages that are dripped on during all climates (4.5 percent); packages that are dripped on only during the long-term average and superpluvial climates (22.5 percent); and packages that are dripped on only during the superpluvial climates (18 percent). However, only two of these groupings are represented in the waste package degradation model: packages that are dripped on from time zero to 1 million years, and packages that are never dripped on. It is thus a conservative assumption (i.e., erring on the side of maximum releases) to assume that the long-term average fraction of packages (i.e., the fraction that is dripped on during long-term average and superpluvial climates) has the same failure rate during the long-term average climate and the dry climate. On the other hand, a somewhat nonconservative assumption is that the superpluvial fraction of packages fails according to the non-seep failure curve. This assumption is used because these packages are not dripped on at all until at least 250,000 years after repository closure.

[34] The average neptunium-237 inventory per package is 11.4 Ci per package for commercial spent nuclear fuel, 0.735 Ci per package for high-level radioactive waste, and 0.153 Ci per package for DOE spent nuclear fuel.

[35] Some of the inventory of DOE spent nuclear fuel has intact cladding (such as naval fuel and about 50 percent of N-Reactor fuel); however, it is conservatively assumed in TSPA-VA that all the DOE spent nuclear fuel cladding with the exception of naval spent nuclear fuel is failed at emplacement. This assumption is made in order to bound the effect of DOE spent nuclear fuel on repository performance. The performance of naval spent nuclear fuel cladding is discussed in Sections 3.5.3.5. and 5.5.5.

[36] This maximum fraction of 3 percent clad failure over 100,000 years is a combination of creep failure, stainless-steel-rod failure, Zircaloy corrosion failure, and mechanical failure (see Section 4.1.8).

the seepage curve or the cumulative inventory curve could be responsible for the shape of the neptunium-237 release curve. However, the latter is responsible because the seepage is already high enough to have reached the solubility limit, as demonstrated in the discussion above regarding the advective and diffusive release curves. The steadily increasing cumulative inventory curve means that more packages, seepage, and inventory are becoming available with time for transport of more neptunium-237 away from the entire repository.

Figure 4-17 shows the performance of the waste package and engineered barrier system for the three inventory types: commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste. One implication from this figure, especially when the technetium-99 cumulative inventory exposure curves are compared, is that the exposed DOE spent nuclear fuel inventory is nearly as large as the commercial spent nuclear fuel inventory at 100,000 years—the ratio of DOE inventory to commercial inventory is about 0.4.[37] Also, this ratio is roughly maintained when translated to releases from the waste packages, because the dissolution rates for DOE spent nuclear fuel and commercial spent nuclear fuel are comparable for most of the 100,000 years, as shown in Figure 4-17. In contrast, the exposed technetium-99 high-level radioactive waste inventory is about twice as much as the exposed commercial spent nuclear fuel inventory. However, the technetium-99 high-level-waste activity released is about five times lower than the commercial-spent-nuclear-fuel activity released because the high-level-waste degradation rate is lower than the commercial and DOE spent nuclear fuel degradation rates. The exposed neptunium-237 inventory and its activity released from the engineered barrier system has a similar relationship to that of technetium-99 because of fuel degradation rate.

Returning to Figure 4-16, another point about the engineered barrier system behavior is the plutonium-239 release curve. The radionuclide plutonium-239 is another actinide that is solubility limited in ambient Yucca Mountain pore waters. Therefore, its release behavior from the waste package and engineered barrier system should be similar to neptunium-237, which is exactly the behavior shown in Figure 4-16. However, as discussed in Section 3.5, colloidal transport of plutonium-239 is also included in the TSPA-VA model, so the shape of the engineered barrier system release curve for plutonium-239 might be expected to be different from the neptunium-237 release curve. It is not different because in the TSPA-VA model the fraction of *irreversible* colloids (unaffected by interaction with rock matrix or fractures anywhere in the system) is at most only 1×10^{-4} times the concentration of reversible colloids (and only 1×10^{-7} for the expected-value base case realization being described here—see Section 4.1.10). Furthermore, since the *reversible* colloid concentration is linearly proportional to the dissolved aqueous concentration (see Section 3.5), it is necessarily true that the shape of the plutonium-239 release curve is the same with or without colloids. Also, for the expected-value base case the proportionality factor, K_c (the ratio of plutonium mass on colloids to plutonium mass dissolved in the aqueous phase), is only about 0.33, so the plutonium-239 release curve must be dominated by

[37] Roughly, the total technetium-99 commercial spent nuclear fuel inventory can be computed from the product of the number of commercial spent nuclear fuel packages (7,760) times the cladding fraction (at most only 0.03 of the cladding fails during the first 100,000 years) times the technetium-99 inventory per package (118 Ci). The product equals 27,470 exposed Ci. Similarly for the technetium-99 DOE spent nuclear fuel inventory: 2,546 DOE spent nuclear fuel packages times technetium-99 inventory per package (2.55 Ci) equals 6,492 exposed Ci. The TSPA-VA model conservatively assumes no intact cladding on the DOE spent nuclear fuel. Therefore, the ratio of DOE spent nuclear fuel to commercial spent nuclear fuel inventory is about 0.24. (Actually, the number of packages used in this computation should be the number of failed packages at a given time. Because the commercial and DOE spent nuclear fuel fail at about the same rate, the ratio of the inventories can, instead, be computed from the ratio of the total emplaced packages. Also, the 3 percent clad fraction used in this example is only an approximation and changes with time during the 100,000 years, growing from 1.25 percent at initial package failure to about 3 percent 100,000 years after package failure.)

dissolved plutonium-239, not colloidal plutonium-239.

Figure 4-16 also shows the performance of the engineered barrier system, specifically the concrete invert. Although a pulse-release breakthrough curve is not shown, a comparison of total neptunium-237 releases from the engineered barrier system with advective neptunium-237 releases from the waste package indicates that there is a transport delay of neptunium-237 in the invert. The neptunium-237 distribution coefficient (K_d) in the invert is about 100 mL/g (and the plutonium K_d is about 600 mL/g). This K_d for neptunium makes invert travel time in the dry climate very slow (on the order of 100,000 years through the 1 m of invert), while in the long-term average climate the travel time is about 7,500 years. Therefore, because there is a climate change at 5,000 years after closure, the effective travel time of neptunium-237 through the invert is about (5,000 + 7,500 =) 12,500 years. (The neptunium moves only a negligible distance during the 5,000 years of dry climate).

Figure 4-18 shows the performance of the unsaturated zone base case model over the 100,000-year time span, during which the long-term average climate predominates. In Sections 3.1 and 4.1, the unsaturated zone expected-value long-term average base case was explained as having an average of about 40 mm/year percolation through the unsaturated zone at the repository horizon. Part of this flow is diverted laterally by the perched-water zone beneath the northern portions of the repository. The net effect of the diversion, and also sorption of the radionuclides onto the rock matrix, is shown in the unsaturated zone activity-release plot in Figure 4-18. This plot represents the summed activity release over all six regions at the water table for the three indicated radionuclides. Comparison of the engineered barrier system and unsaturated zone release rates shows that the unsaturated zone has little delaying effect on the transport of the nonsorbing technetium-99. For the weakly sorbing neptunium-237, the unsaturated zone appears to delay its movement by about several thousand years. For the strongly sorbing plutonium-239, the unsaturated zone delays its releases by tens of thousands of years.[38] (Using the 50 percent point on the pulse-release breakthrough curves as the metric to gauge travel time, technetium-99 delay is 300 years, neptunium-237 delay is 20,000 years, and plutonium-242 delay is apparently greater than 1 million years. Plutonium-242 breakthrough is shown in the plot because plutonium-239 has a very short half-life of 24,000 years, while plutonium-242 has a half-life of about 387,000 years. However, even with this half-life, the 50-percent-breakthrough metric is in error because the normalization factor for the plot is the initial number of particles, many of which have decayed before traversing the entire unsaturated zone. The exact amount of error can only be determined by simulating plutonium-242 or plutonium-239 transport without decay, that is, taking into account sorption, but not decay.)

Figure 4-19 shows the performance of the saturated zone over the 100,000-year time span for this expected-value base case realization. Shown are two sets of plots; release rate graphs at both the base of the unsaturated zone and the 20-km (12-mile) distance in the saturated zone, and concentration plots at the water table beneath the repository and at the 20-km (12-mile) distance. Also shown again (see Figure 4-15) are the saturated zone generic breakthrough curves and the dilution-factor cumulative distribution functions. The saturated zone demonstrates similar behavior to the unsaturated zone for the three radionuclides; technetium-99 is only slightly delayed, neptunium-237 is moderately delayed, and plutonium-239 is strongly delayed. Based on the breakthrough curves shown, the delay of the 50 percent concentration point is about 50,000 years for neptunium-237 and about 110,000 years for plutonium-239. Finally, as with the 10,000-year technetium-99 transport discussed in Section 4.2.1, neptunium-237 transport through the six streamtubes similarly

[38] The plutonium-239 release curve is erratic at the base of the unsaturated zone. This nonphysical behavior is caused by the discrete nature of the particle tracking model when releases are very low, resulting in the observance of individual particles passing out of the domain.

experiences an effective dilution factor of about four when sampled in a well-withdrawal scenario at 20 km (12 miles).

Figure 4-20 illustrates how the relative concentrations of the three radionuclides in the groundwater translate to relatively different doses as a result of the disparate biosphere dose conversion factors. In particular, the biosphere dose conversion factors (mrem/year/Ci/m^3) of neptunium-237 and plutonium-239 are about 1,000 times higher than the technetium-99 biosphere dose conversion factor. This implies that although the neptunium-237 activity concentration in the groundwater (Ci/m^3) is much lower than technetium-99, its dose rate is about 10 times higher because of larger impact on the human body.[39] Similarly, although the plutonium-239 activity concentration in the saturated zone is very low at 100,000 years (1×10^{-12} Ci/m^3 or 0.001 pCi/L), its dose reaches several millirem per year because of its relatively high biosphere dose conversion factor.

The final plot in Figure 4-20 shows the small effect of including a model for irreversible sorption of plutonium-239 onto colloids, where the ratio of the irreversible fraction to the reversible fraction is 1×10^{-7} in the base case expected-value realization. This model accounts for the early breakthrough of plutonium colloids observed in the Benham test. The scale on this plot is not low enough to show the first breakthrough of these irreversible colloids at the 20-km (12-mile) boundary. However, examining the modeling results shows a first breakthrough at about 1,500 years, which represents the very fast fracture transport through the natural system of the releases from the juvenile-failure package that fails at 1,000 years.

4.2.3 One-Million-Year Dose Rates

Within the first 1 million years after closure, technetium-99 and neptunium-237 are the dominant radionuclides to reach the biosphere; technetium-99 dominates dose rates up until about 50,000 years, after which neptunium-237 reaches the biosphere in appreciable amounts and dominates thereafter. After a long period of delay in the natural barriers, plutonium-242 finally appears at the biosphere in appreciable amounts beginning at about 200,000 years and then remains as the second most important radionuclide, after neptunium-237, for the remainder of the 1 million-year period. In the two superpluvial periods that occur, plutonium-242 has a relatively sharp peak that contributes about the same as neptunium-237 to the total peak dose during those climatic periods. There are relatively large inventories of all three radionuclides (technetium-99, neptunium-237, and plutonium-242) in the commercial spent nuclear fuel packages. The key qualities of technetium-99 and neptunium-237 have already been discussed at the beginning of Sections 4.2.1 and 4.2.2. For the 1 million-year performance period, their behavior is again used to explain the performance of the repository system and subsystems. In addition, the behavior of the rather long-lived plutonium-242 is used to explain the behavior of strongly sorbing radionuclides during the 1-million-year period.[40]

Figures 4-21 through 4-25, discussed in detail in this section, show the primary causes for the shape and timing of the 1-million-year dose rate graphs. Figure 4-21 shows the 1-million-year performance of the engineered barrier system. Again, the key factors are climate change and waste package failure history. In contrast to the 10,000-year and 100,000-year time spans, however, an additional parameter is necessary to explain the shape of the curves after 200,000 years: cladding failure rate. This parameter is discussed in detail below.

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[39] The mass concentrations (g/m^3) for technetium-99 and neptunium-237 are actually much closer to each other than are their activity concentrations, about 7×10^{-6} g/m^3 and 1×10^{-6} g/m3 respectively; however, the specific activity (Ci/g) of neptunium-237 is much lower than that of technetium-99.

[40] Plutonium-242 has a half life of about 387,000 years.

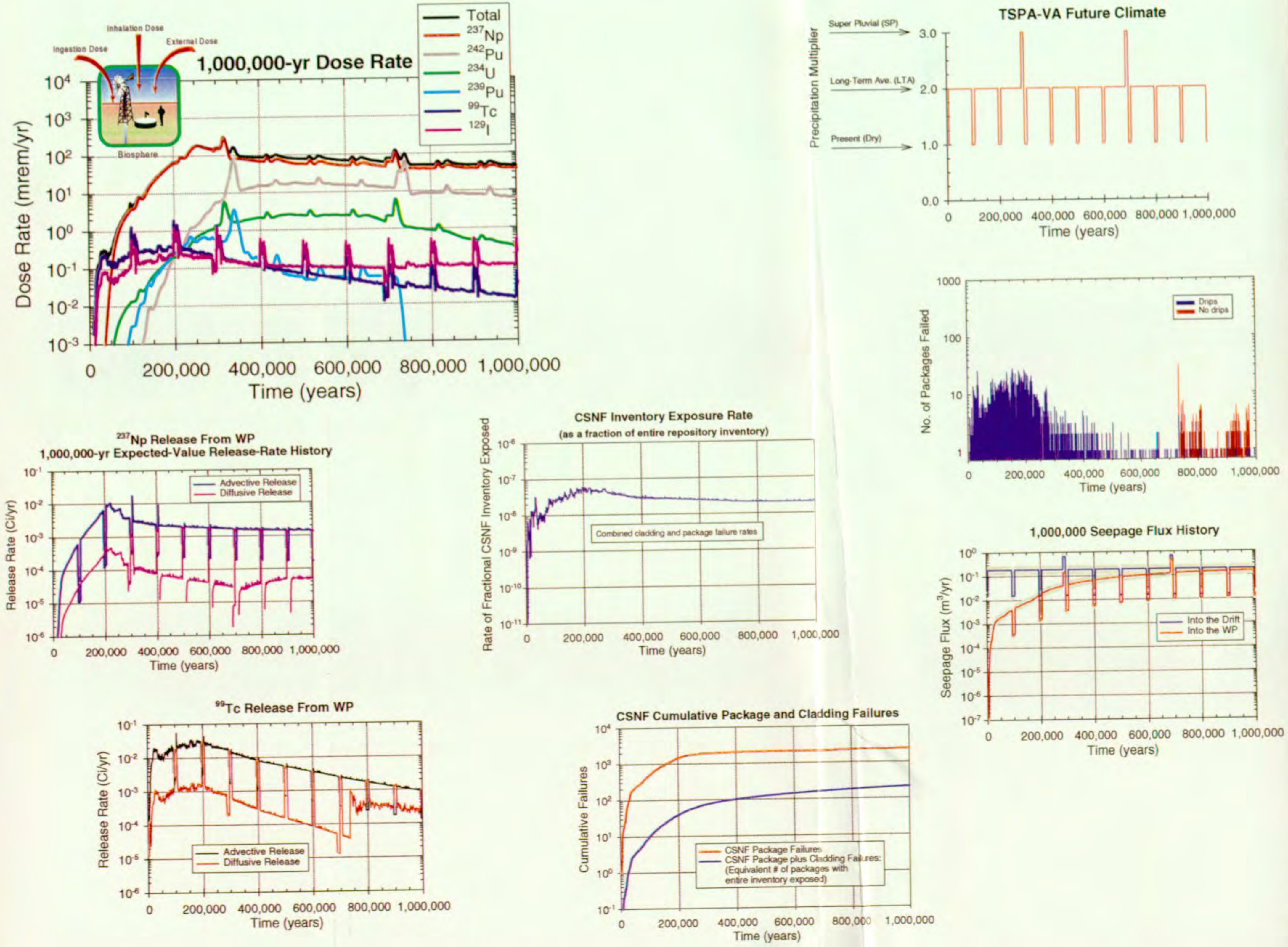

Figure 4-21. Effects of Climate Change, Seepage Flux, Waste Package Degradation, and Cladding Degradation on Waste Package Releases and Dose Rates in the First 1 Million Years After Waste Emplacement
(CSNF—commercial spent nuclear fuel; WP—waste package)

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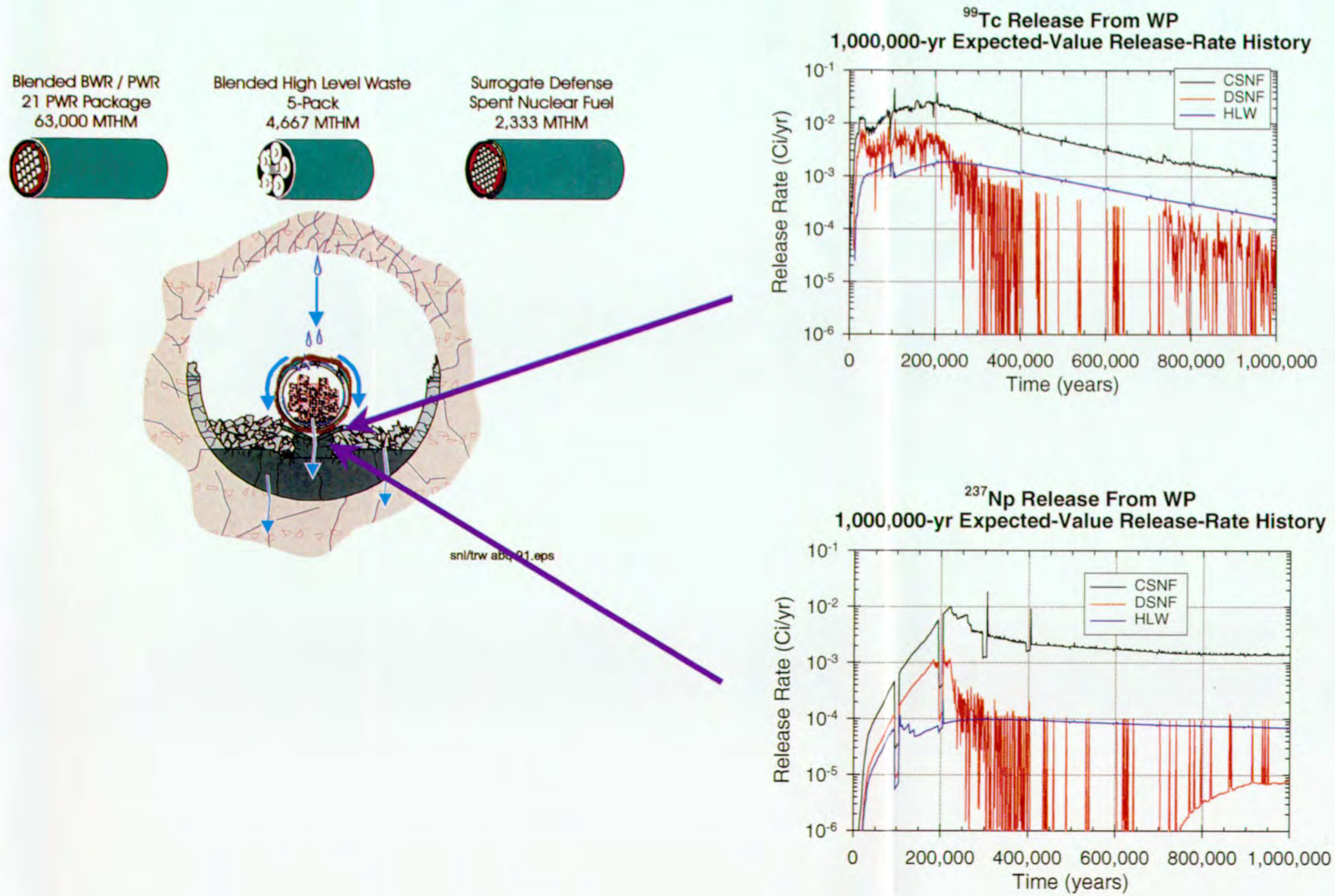

Figure 4-22. Waste Package Releases During 1 Million Years After Waste Emplacement for the Three Inventory Types: Commercial Spent Nuclear Fuel, U.S. Department of Energy Spent Nuclear Fuel, and High-Level Radioactive Waste
(WP—waste package; CSNF—commercial spent nuclear fuel; DSNF—DOE spent nuclear fuel; HLW—high-level radioactive waste)

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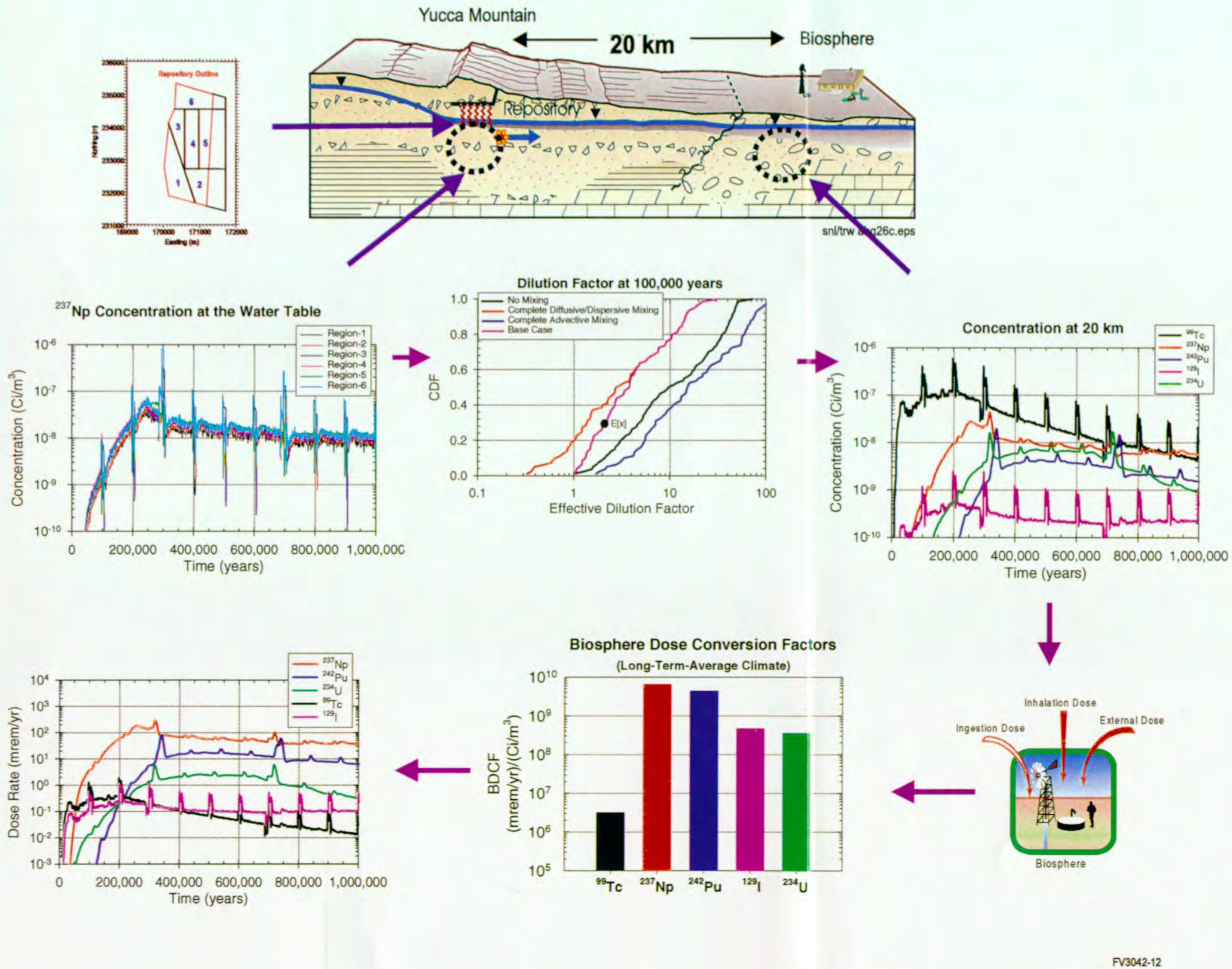

Figure 4-23. Effects of Saturated Zone Dilution and Biosphere Dose Conversion Factors on 1 Million Year Dose Rates
The insert at upper left shows the six water table regions.
(BDCF—biosphere dose conversion factors)

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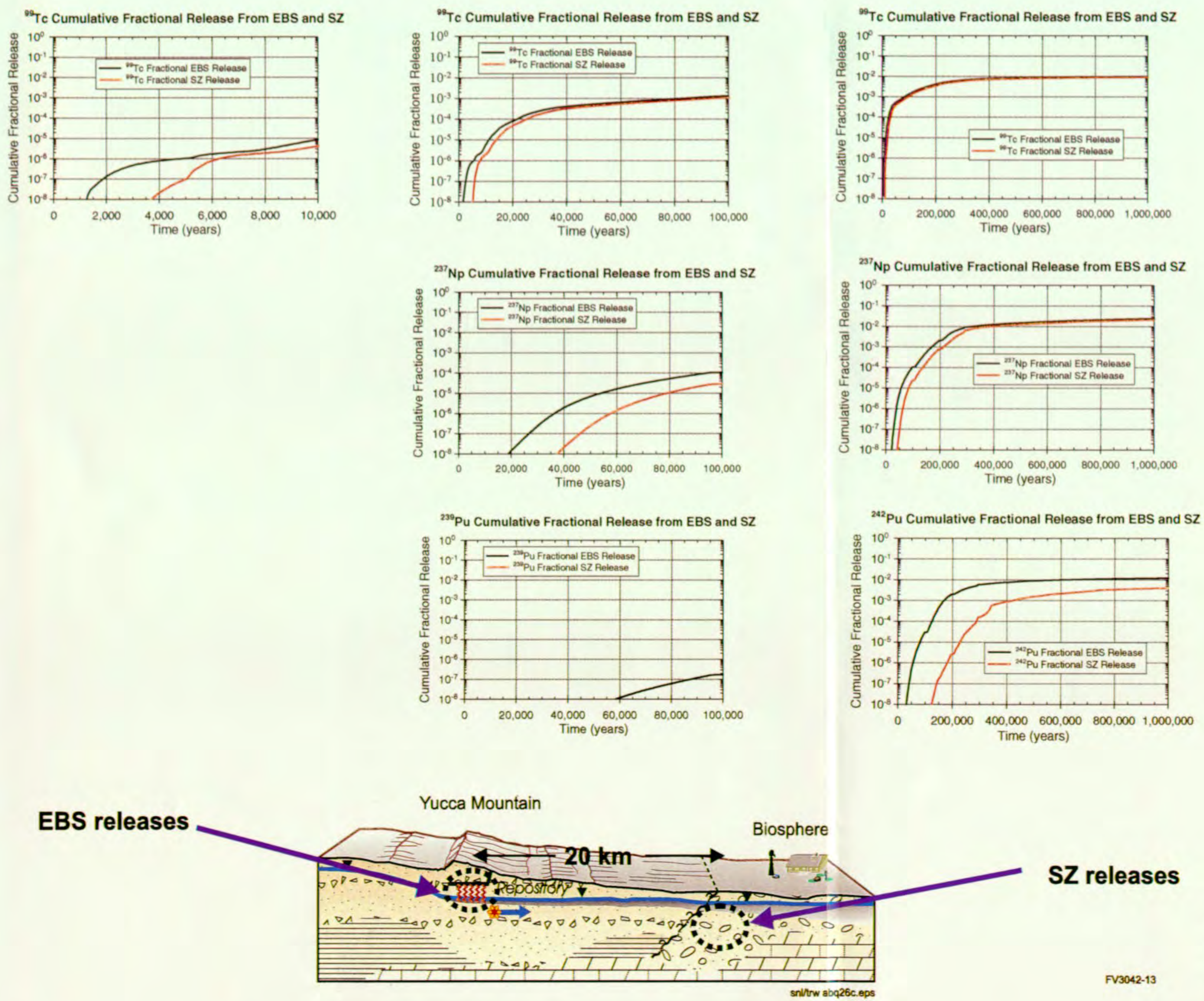

Figure 4-24. Cumulative Fractional Inventory Releases of Technetium-99, Neptunium-237, Plutonium-239, and Plutonium-242 from the Engineered Barrier System and the 20-km Saturated Zone Boundary, Normalized to the Initial Inventory
Shown are the effects of sorption and decay of radionuclides. (EBS—engineered barrier system; SZ—saturated zone)

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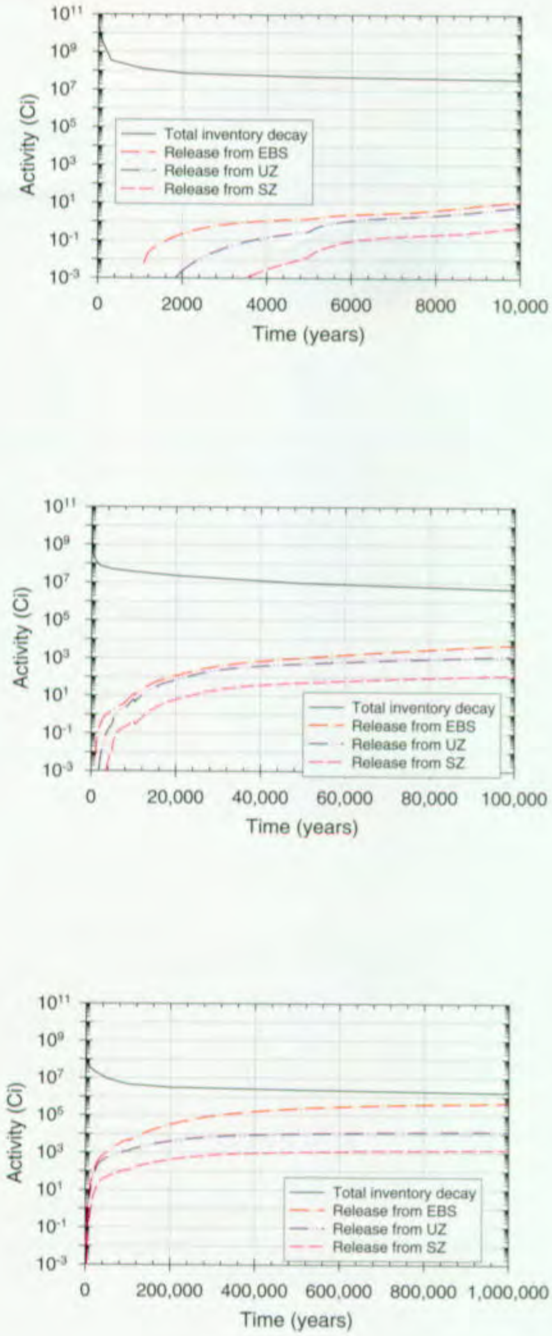

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Figure 4-25. Plots of Cumulative Inventory Released from Different Boundaries of the Repository System Along with the Inventory Decay Curve over a Period of 1 Million Years
(EBS—engineered barrier system; UZ—unsaturated zone; SZ—saturated zone)

As in the 100,000-year time span, the jaggedness of the instantaneous package-failure history explains the jagged appearance of the graphs for technetium-99 waste package releases and biosphere dose rates—for the same identical reason, namely, that technetium-99 is a "release-rate-limited" radionuclide. That is, the radionuclide has such a high solubility that its release is limited by the rate at which packages fail, not the rate at which water moves it out of the packages. The instantaneous package failure history shown in Figure 4-21 is a histogram of package failures, with the interval size being equal to the time step size of 1,000 years. Because this time step is larger than for the 100,000-year simulation, some of the peaks on the histogram are larger than the 100,000-year package-failure histogram in Figure 4-16. Instead of 10 packages being the maximum in any interval, there are up to 30 or more in an interval.

Another feature of the waste package failure history is the failure of the no-drip packages beginning at about 730,000 years. This is the group of packages discussed earlier that are never dripped on; about 55 percent of the total packages in the repository. Because they are never dripped on, their corrosion rate is much lower than for dripped-on packages and, therefore, they do not begin to fail until a very long time after emplacement. Even when they do fail, there is no flowing water to sweep out the inventory, so the only release mechanism is diffusive transport. This effect is shown on the technetium-99 diffusive release curve and slightly evident on the neptunium-237 release curve; however, the effect of these no-drip packages is barely discernible on the dose rate curve in the biosphere.

As with the 100,000-year time histories discussed in Section 4.2.2, neptunium-237 releases are different from technetium-99 releases because of neptunium-237's low solubility, at least for the first 200,000 years. However, after 200,000 years neptunium-237 actually switches from being a solubility-limited radionuclide to being a release-rate limited radionuclide—limited by the cladding failure rate. A number of the plots on Figure 4-21 explain this effect. First, after about 200,000 years the seepage flux through the packages is very high—high enough to sweep out all of the released neptunium-237, even though it has a very low solubility in the flowing water. Second, the commercial spent nuclear fuel cumulative package-failure curve flattens out at about 200,000 years, showing that very few additional commercial spent nuclear fuel packages fail for the remainder of the 1 million years. However, the combined commercial spent nuclear fuel cumulative package-cladding failure curve (the integrated or "convolved" product of package failures with cladding failures) continues to increase after 200,000 years, indicating that more inventory is being exposed. However, the rate at which this cumulative package-cladding failure curve increases is declining with time after 200,000 years. This effect is evident on the instantaneous "commercial spent nuclear fuel inventory exposure rate" curve, which represents the rate at which inventory is exposed to transport as a function of time (where the inventory is represented as a fraction of the entire repository inventory).[41] This inventory exposure curve exactly mirrors the shape of the neptunium-237 advective release curve between 200,000 years and 1 million years, and also the shape of the neptunium-237 dose rate curve. This implies that neptunium-237 is no longer solubility limited after about 200,000 years, but is "cladding release rate" limited. In other words, the release rate or exposure rate of neptunium-237 by failing cladding is slow enough and the seepage flux is high enough that the low solubility of neptunium-237 no longer limits its release from the waste package. This effect is also shown on the diffusive release curve for neptunium-237, which looks very much like the technetium-99 diffusive release curve after 200,000 years. In particular, during the dry climate states, the diffusion releases of both neptunium-237 and technetium-99 attain the same rate as the long-term average advective release rate. This is because the concentrations of both radionuclides build up during the dry climate (i.e., neither were solubility limited during the long-term average climate, so both of their concentrations in the waste packages build up during the dry climate, causing an increased diffusive flux). On the neptunium-237 release plot, this behavior is evident in the dry climate states after 200,000 years. However, in the two dry climate states before 200,000 years, neptunium-237 is solubility limited and its diffusive release curve shows no abrupt changes when the dry climate is established. Also, before 200,000 years, the neptunium-237 release curve shows a sharper rise and peak than the commercial spent nuclear fuel inventory exposure curve because the neptunium-237 release follows the package-failure cumulative distribution function up until that point. The technetium-99 release rate curves also follow the commercial spent nuclear fuel inventory exposure curve at all times, as is expected for a radionuclide that is not solubility limited.[42]

Figure 4-22 shows the relative contributions to engineered barrier system releases from the three main inventory types over the 1 million-year time frame. Following are the important points about these plots:

- DOE spent nuclear fuel is the second highest contributor to neptunium-237 and technetium-99 releases during the first 200,000 years. Commercial spent nuclear fuel is the highest contributor. After that time, the DOE spent nuclear fuel inventory begins to run out because very few DOE spent nuclear fuel packages fail after

[41] This inventory exposure rate curve is the derivative of the cumulative package-cladding failure curve normalized to (i.e., divided by) the total number of commercial spent nuclear fuel packages.

[42] An unusual feature of the neptunium-237 diffusive release curve is the series of gaps that appear in the curve each time the climate switches from dry to long-term average, and also when the climate switches from long-term average to the second superpluvial at about 730,000 years. The gaps are caused by the negative diffusion rate at these points, which cannot be plotted on a log scale. Apparently, when an abrupt change to a wetter climate occurs, neptunium-237 is rapidly flushed from the waste package, causing near zero concentration in the waste package and allowing "backwards" diffusion from the invert into the package. Although not shown in this document, the comparable advective and diffusive releases from the engineered barrier system do not exhibit these negative diffusion rates.

200,000 years. (Individual package failure is readily discernible on the plot.)

- High-level radioactive waste never contributes very much to neptunium-237 and technetium-99 releases compared to commercial spent nuclear fuel.

For performance over the 1-million-year period, the unsaturated zone is of very little consequence, so no figures are shown for it. On the other hand, the saturated zone is important because of dilution, which proves to be the second most important uncertainty (see Section 4.3).[43] Figure 4-23 shows how dilution in the saturated zone affects concentration, specifically neptunium-237, and how the biosphere dose conversion factors for the various radionuclides are used to convert saturated zone concentrations to dose rates.

4.2.4 Cumulative Activity Releases from the Repository and the Engineered Barrier System

Another way of examining repository performance is by looking at cumulative activity release (Curies) versus time at the edge of the engineered barrier system compared to the cumulative activity release in the saturated zone at 20 km (12 miles). First, this method is another means of comparing natural system performance with engineered system performance. Second, the method illustrates the large fraction of the total radioactivity that is actually retained in the repository system at various times. The expected differences in the engineered barrier system and saturated zone cumulative release curves are because of retardation (sorption) and decay. If radionuclides did not decay, eventually the engineered barrier system and saturated zone cumulative curves would become equal. Figure 4-24 shows cumulative fractional activity releases of technetium-99, neptunium-237, plutonium-239, and plutonium-242 at these two spatial locations for the time frames of 10,000 years, 100,000 years, and 1 million years. Cumulative fractional release for any radionuclide at a given spatial location and a given time is defined as the cumulative activity that has traveled past that spatial location divided by the initial inventory emplaced for that radionuclide. In the 10,000-year time frame, nearly all the technetium-99 radioactivity is retained in the repository—less than 0.001 percent has been released from either the engineered barrier system or the saturated zone.[44] After 100,000 years, less than 0.2 percent of the technetium-99 has been released from the repository system (saturated zone at 20 km, or 12 miles) and less than 0.02 percent of the neptunium-237 has been released. The amount of plutonium-239 released beyond 20 km (12 miles) in the saturated zone does not appear on this graph but is slightly less than 1×10^{-9} percent. At 1 million years, about 1 percent of the technetium-99, 2 percent of the neptunium-237, and 0.4 percent of the plutonium-242 activities have reached the 20-km (12-mile) boundary. Much of the 99 percent of the technetium-99 that is never released is a result of radioactive decay throughout the repository system.

Figure 4-25 is another way to show cumulative inventory released. It plots the absolute, rather than the fractional inventory, released (i.e., the release curves are not scaled to the initial inventory). Also, it shows total inventory released rather than individual radionuclides. Total releases from three system boundaries are shown: edge of the engineered barrier system, base of the unsaturated zone, and 20 km (12 miles) downgradient in the saturated zone. By way of comparison the total inventory decay curve is also shown. This curve is the activity versus time for all the emplaced inventory if it were never mobilized by water or gas, that is, if it were always isolated. Figure 4-25 demonstrates that by 10,000 years the inventory that has escaped from the engineered barrier system is less than 0.0001 percent of the total isolated inventory. In other words, 99.9999 percent of the inventory remains in the engineered barrier system up until 10,000 years after closure.

[43] The neptunium-237 biosphere dose conversion factor is the third most important uncertainty in the 1 million-year time frame.

[44] The percent of neptunium-237, plutonium-239, and plutonium-242 released is negligible.

At 100,000 years the plots indicate that 0.1 percent of the inventory has escaped the engineered barrier system and 99.9 percent remains isolated in the drifts. At 1 million years about one-half of the inventory has escaped the engineered barrier system, but 99.9 percent remains contained in the entire repository system—within the natural and engineered barriers, that is, it has not passed beyond the 20-km (12-mile) "boundary."[45]

4.2.5 Summary

The purpose of Section 4.2 has been to demonstrate the influence of the various subsystem and component processes on the overall system performance. The illustrative realization used for this purpose is the "expected-value" realization, that is, the realization that chooses the expected value of all input parameters. The behavior of this realization cannot necessarily be considered to represent the overall mean behavior of the repository, as determined by multiple realization simulations, because of the nonlinear behavior of the models and processes. However, as will be seen in Section 4.3, this expected-value realization does lie near the central tendency of the overall probability distribution—generally somewhere between the median and the mean.

Based on the analysis in Section 4.2, nonsorbing radionuclides, such as technetium-99, dominate peak dose rates at "early" times, perhaps up to 50,000 years after repository closure. After that, the weakly sorbing neptunium-237 controls peak dose rates, with some additional contribution from plutonium-242.

During the first 10,000 years after repository closure, the key factors controlling repository behavior are the failure of the single juvenile (or early) failed package and the few (about 17) corrosion-failed packages, and the climate change at 5,000 years. The main factor affecting releases from the waste packages is the seepage of water into the waste packages, which is controlled by the corroded pit and patch area. During this time frame the seepage flux into the corrosion-failure packages is low enough that technetium-99 releases become solubility-limited in those packages. During this 10,000-year span after closure, travel time in the unsaturated zone and saturated zone during the dry climate state is also long enough to enhance performance.

During the first 100,000 years after repository closure, the major factors controlling performance are the instantaneous waste package failure rate (controls technetium-99 releases), the cumulative waste package and cladding failures (controls neptunium-237 releases), the fraction of packages encountering seeps, and the seepage rate into the packages. The seepage rate is important because neptunium release is solubility limited, so the absolute amount of neptunium-237 escaping any package is linearly proportional to the seepage flowing through the package.

During the first 1 million years after repository closure, the major factors controlling performance are the combined cumulative package and cladding failures, and the superpluvial climate change. The latter causes the peak in the total and neptunium dose-rate curves at about 300,000 years.

Dilution in the saturated zone is important at all time frames, but because of conservative assumptions about its magnitude, it does not play as large a role as in previous TSPAs (for example, see CRWMS M&O 1995). Also, sorption in the unsaturated and saturated zones is important for all of

[45] The curve in Figure 4-25 that is labeled "release from EBS" represents a total over the 39 major radionuclides modeled (see Chapter 6 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* [CRWMS M&O 1998i]). The curve labeled "release from UZ" represents the total over the nine radionuclides discussed in this document. However, as shown in Chapter 6 of *Total System Performance Assessment- Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) the total releases from the 39 radionuclides at the base of the unsaturated zone are essentially identical to the releases from the nine radionuclides. The curve labeled "release from SZ" is also the release from the 9 radionuclides and would be expected to be identical to the 39 radionuclides.

the strongly sorbing radionuclides, such as plutonium, and provides a very long delay (up to hundreds of thousands of years) in transport to the 20-km (12-mile) boundary.

Commercial spent nuclear fuel dominates technetium and neptunium releases, but DOE spent nuclear fuel releases are not insignificant and comprise about 25 percent of the total.

4.3 PROBABILISTIC RESULTS OF THE BASE CASE

For the TSPA-VA, a probabilistic approach is used for assessing the long-term performance of the Yucca Mountain repository. This approach uses a linked system of deterministic models to represent the repository and its associated geologic system, and a Monte Carlo technique to propagate parameter uncertainty through to the calculation of peak radiation dose rates at the specified location 20 km (12 miles) from the repository. A full analysis includes the following:

1. Selecting imprecisely known model input parameters to be sampled
2. Constructing probability distribution functions for each of these parameters, incorporating available data and subjective information to capture uncertainty
3. Generating a sample set by selecting a parameter value from each distribution
4. Calculating outcomes for the sample set

Steps 3 and 4 are repeated many times to produce a distribution of peak dose rates that represents the spectrum of repository performance. The distribution of peak dose rates is normally presented as a CCDF that gives the probability of exceeding a given peak dose rate. The range or spread of peak dose rates represents the amount of uncertainty in the results. In addition to quantifying uncertainty in the results of the performance assessment, another important component of the probabilistic approach used in the TSPA is the sensitivity analysis. Sensitivity analysis is used to identify the relative importance of uncertain input parameters to the calculated uncertainty in repository performance. This information is useful for selecting and assigning priorities to future modeling, site characterization, and design activities so that uncertainty in estimates of long-term performance may be decreased.

This section summarizes the probabilistic results obtained from the base case Monte Carlo simulations. Results are presented in Sections 4.3.1 and 4.3.2. Section 4.3.1 focuses on the calculated peak dose rates to a human located 20 km (12 miles) from the repository. Dose rates are presented in the form of CCDFs for three periods: 10,000 years, 100,000 years, and 1 million years. Factors are also discussed that cause uncertainty in the calculated peak dose rates. Auxiliary results are provided about which radionuclides contribute most to dose and the range in times at which peak dose rates occur. In Section 4.3.2, the contributions of individual parameters to uncertainty in peak dose rates are discussed. Uncertain parameters that are most important to subsystems, such as radionuclide releases from the engineered barrier system or from the unsaturated zone, can also be analyzed, and one example is briefly discussed. For more information on the methods used for probabilistic analysis, see Section 11.3 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

4.3.1 Uncertainty Analysis

In this section, uncertainty results of the Monte Carlo simulations are reported for three periods: 10,000 years, 100,000 years, and 1 million years. The following is a brief description of how the probabilistic results are obtained.

The TSPA results can be symbolically represented by a simple function of the form

$$\text{Dose Rate} = f(x_1, x_2, \ldots, x_{nc}, s_1, s_2, \ldots, s_{nu}, t),$$

where dose rate represents the dose rate to a human at the 20-km boundary as a function of time t.

The variables $x_1, x_2, ..., x_{nc}$ are precisely known input parameters such as the acceleration of gravity and the density of water, and nc is the number of such inputs. Variables $s_1, s_2, ..., s_{nu}$ are imprecisely known or uncertain parameters such as corrosion rate and waste form dissolution rate, and nu is the number of such parameters. In the TSPA-VA base case, nc = 1,006 and nu = 177. Note that the 1,006 certain parameters and 177 uncertain parameters include only those that are used within RIP; many additional parameters are used in the various component models such as TOUGH2 and WAPDEG. A description of the uncertain parameters is provided in Chapter 11, Appendix A of CRWMS M&O 1998i. The imprecisely known inputs are the parameters that cause uncertainty in the performance assessment results.

The uncertainty in parameters $s_1, s_2, ..., s_{nu}$ is characterized by a sequence of probability distributions

$$D_1, D_2, ..., D_{nu},$$

where D_j is the probability distribution for variable z_j. In some cases, the definitions of these distributions are also accompanied by specifications of correlations that further define the relations between the s_j. For the TSPA-VA base case, the seepage parameters for different repository subregions are correlated, seepage and net infiltration for different climate states are correlated, and biosphere dose-conversion factors for different radionuclides are correlated (see Sections 3.1.2.2, 3.1.2.4, and 3.8.2).

Once distributions D_j are specified, the next step is to determine the uncertainty in the performance assessment results that arises from uncertainty in $s_1, s_2, ..., s_{nu}$. First, a sample

$$S_k = [s_{k1}, s_{k2}, ..., s_{k,nu}], \ k = 1, ..., nu,$$

is generated from the specified distributions $D_1, D_2, ..., D_{nu}$, where nk is the size of the sample, or number of realizations. In the TSPA-VA, this sample is generated using Latin Hypercube sampling and the sample size is nk = 100 for most simulations (see Section 4.3.1.2). A performance assessment calculation is then conducted for each sample S_k, with all precisely known input parameters $x_1, x_2, ..., x_{nc}$ held fixed at their known values. The performance assessment calculations yield a series of curves that show how the dose rate changes with time. These curves are referred to as dose-rate time histories and can be represented symbolically as

$$(\text{Dose Rate})_k = f(x_1, x_2, ..., x_{nc}, s_{k1}, s_{k2}, ..., s_{k,nu}, t), \quad k = 1,2, ..., 100,$$

where each $(\text{Dose Rate})_k$, k = 1,2, ..., 100, is the result of a complete calculation of all the linked TSPA components.

Once the dose-rate history for each realization is obtained, the peak dose rate that occurred during the simulation period for each realization is determined by taking the highest point on each dose-rate history curve. Once a peak dose rate is obtained for each realization, the next step is to graphically represent the results in the form of a CCDF, which shows the probability that a dose rate is greater than a given value and is constructed by first ordering the peak dose rates from smallest to largest values, with a probability of 1/nk = 1/100 assigned to each peak dose rate. The probability that a dose rate is greater than a specified dose rate is determined by summing the probabilities associated with all dose rates larger than the specified one. Representative dose-rate time histories for the 100,000-year simulation period are presented next, along with distributions of peak dose rates for the 10,000-, 100,000-, and 1-million-year simulation periods.

4.3.1.1 Uncertainty Analysis Results

Dose-rate histories for the 100,000-year simulation period are presented at the top of Figure 4-26. As noted in the figure caption, 20 realizations in the 100,000-year simulation period did not produce a dose. Also, in the 10,000-year simulation period, 27 realizations have zero dose, but in the 1-million-year simulation all realizations produced a dose. For illustration, four selected realizations are

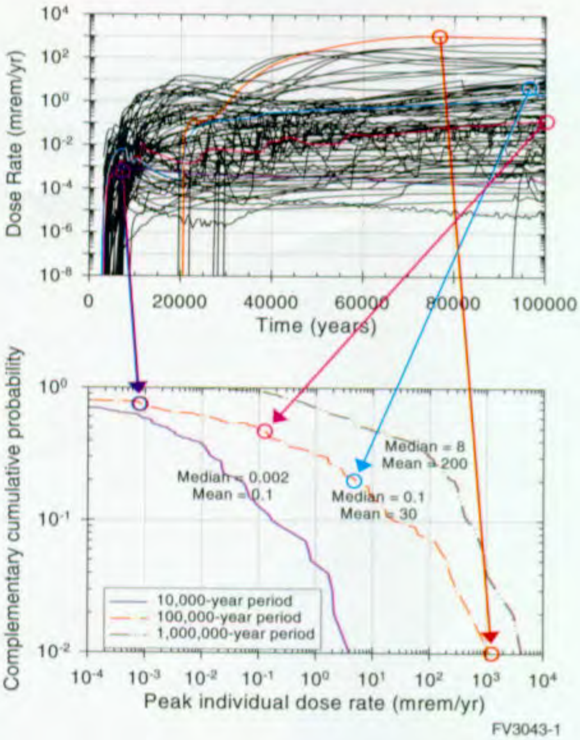

Figure 4-26. Base Case Distribution of Peak Dose Rates for Three Periods and Their Relation to Dose-Rate Time Histories
Twenty dose-rate histories do not appear on the top plot because they have no computed dose for the first 100,000 years.

highlighted; they will be discussed in more detail later in this section. Figure 4-26 shows that the majority of dose rates tend to rise quickly between 3,000 and 10,000 years. A few realizations begin to produce doses in the 20,000- to 30,000-year time interval, with one realization not having any dose until after 90,000 years. In general, four types of behavior through time are shown:

- After the initial steep rise, many of the realizations show a steady increase in dose rate with time that continues throughout the simulation period. This type of behavior reflects significant numbers of waste package failures over time because of corrosion in combination with relatively large fraction of waste packages contacted by seeps.

- Other realizations reach their peak early and their dose rate gradually decreases with time over the simulation period. This type of behavior reflects the impact of juvenile waste package failures, a lower number of waste package failures because of corrosion, and a lower fraction of waste packages contacted by seeps.

- In some realizations, the dose rate remains relatively flat with time after the initial rise. This behavior is caused by juvenile failures in combination with a small number of corrosion-induced waste package failures that are spread over the simulation period.

- In some realizations, dose rate tends to alternately increase and decrease throughout the simulation period. This behavior is primarily caused by a combination of low seepage fraction and discrete waste package failures well separated in time, with a waste package failure roughly coinciding with each dose-rate peak on the history curve.

The distributions of peak dose rates to an individual located at 20 km (12 miles) from the repository are presented at the bottom of Figure 4-26 for the 10,000-, 100,000-, and 1 million-year simulation periods. Peak dose rate is defined as the maximum dose rate that occurs for a given curve during the entire simulation period. This definition is illustrated by indicating the maxima of the four selected realizations and showing where they fall on the 100,000-year CCDF. All realizations are weighted equally, so the lowest probability shown on the curves is 1/100. The median and mean peak-dose-rate values are also shown for each simulation period. The figure shows that peak dose rates continue to increase as time increases. However, the 100,000-year curve shows about a factor of 100 increase in dose rates over the 10,000-year curve, whereas the 1-million-year curve shows only a factor of five to ten increase in dose rates over the 100,000-year curve. The time histories for the 10,000-year and

1-million-year periods are shown for reference in Figure 4-27.

Time histories of some statistical measures of the peak-dose-rate distribution are shown in Figure 4-28. To generate these plots, the statistical measures were calculated from the 100 dose-rate values at each time step in the analyses. For example, the average of the 100 dose rates was calculated and plotted as the "mean" curve and the fifth lowest dose rate among the 100 realizations was determined and plotted as the "5th percentile" curve. Note that in the 10,000- and 100,000-year simulation periods the 5th-percentile dose rate is zero and therefore is not plotted. The mean dose rate in all three periods is much higher than the median value, indicating that the mean values tend to be dominated by a few low-probability high dose rates.

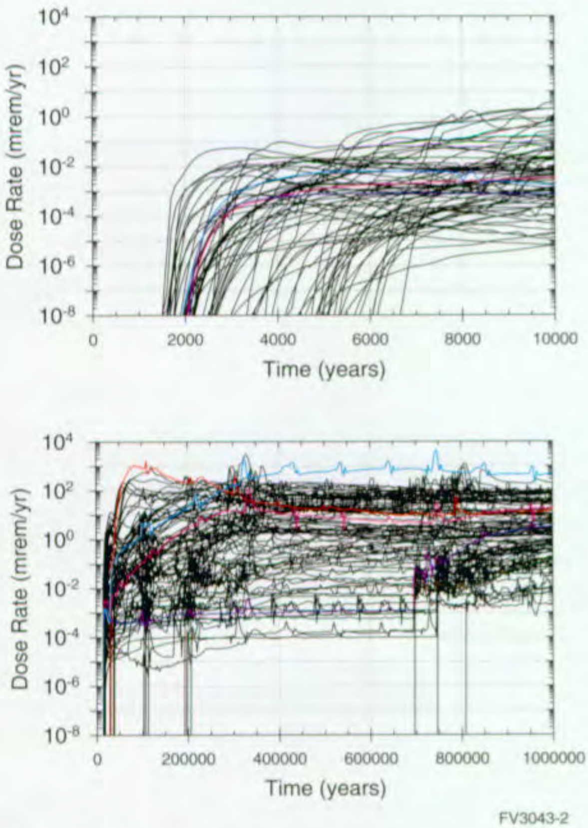

Figure 4-27. Dose-Rate Time Histories for 10,000- and 1-Million-Year Periods
Twenty-seven dose-rate histories do not appear on the top plot because they have no computed dose for the first 10,000 years.

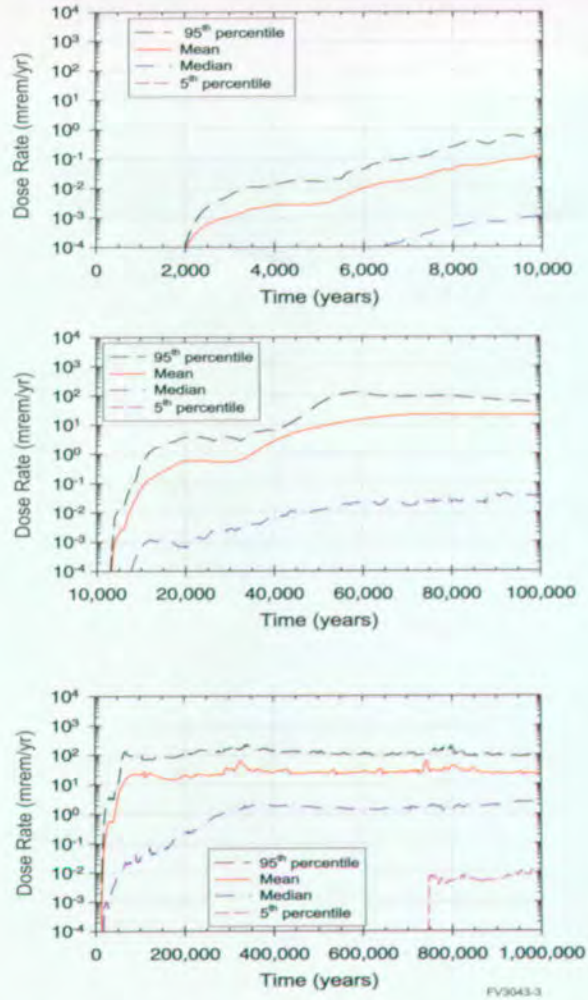

Figure 4-28. Time Variation of Statistical Descriptors of the Calculated Dose-Rate Distribution
Shown are time histories for mean and median dose rate, plus the 5th and 95th percentiles of the dose-rate distribution. The fifth-percentile curve is missing from the top two plots because the fifth-percentile dose rate is zero for those periods.

Figure 4-29 presents the average radionuclide contribution to the peak dose rate. Average contributions are determined by calculating the relative individual radionuclide contributions to the peak dose rate for each realization and averaging the relative contributions across all realizations. In the 10,000-year simulation period (see the top of Figure 4-29), contributions to peak dose rate are dominated by technetium-99 and iodine-129, with a small contribution from carbon-14. In addition, 27 percent of the realizations have zero calculated

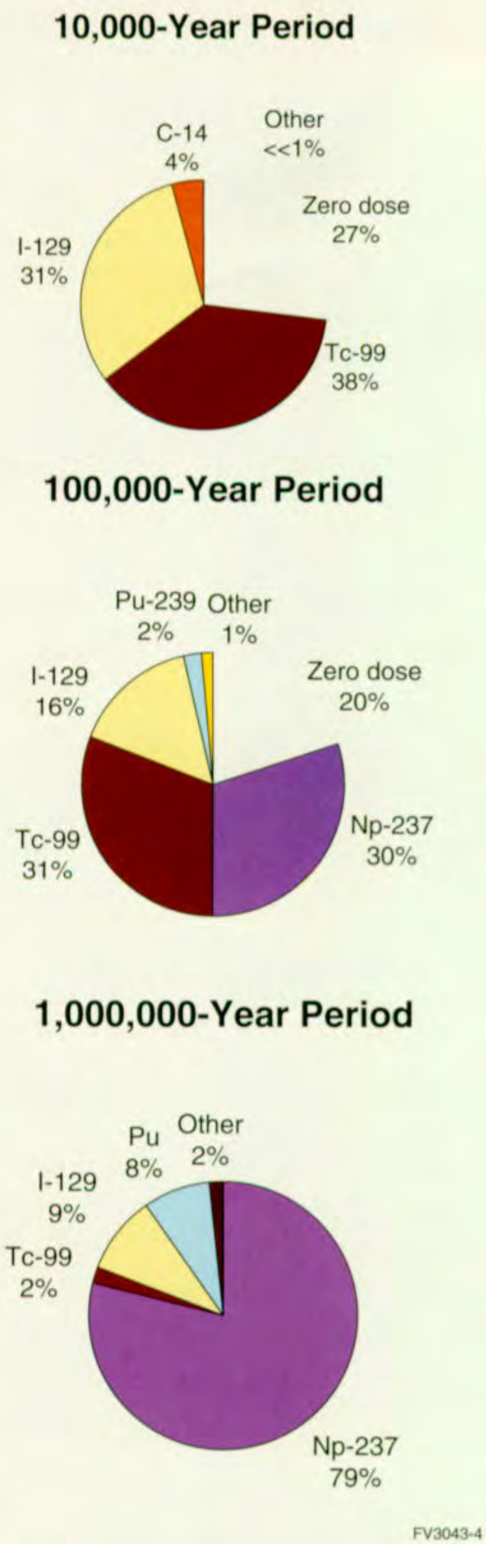

Figure 4-29. Average Contribution to Peak Dose Rate of Different Radionuclides for Three Periods Note that for the earlier two time periods many realizations have no computed dose.

dose. In the 100,000-year period (see the middle chart), contributions to peak dose rate are dominated by technetium-99, neptunium-237, and iodine-129, with 20 percent of the realizations having zero dose. Two percent of the realizations are dominated by plutonium-239. Over 1 million years (see the bottom chart), most peak dose rates are dominated by neptunium-237, with a small number being dominated by iodine-129, plutonium-239, or plutonium-242 (plutonium-239 and plutonium-242 are lumped together in the chart); technetium-99 makes a small contribution to the peak dose rate but is never dominant in any realization.

In general, technetium-99 and iodine-129 are significant contributors to dose because of the following characteristics:

- Large inventories
- Long half-lives (although technetium-99 starts to decay significantly after 100,000 years)
- Highly soluble
- Unretarded during transport

Carbon-14 has a large inventory, is highly soluble, and is unretarded during transport, but its half-life is so short (5,700 years) that it is a significant contributor only for the 10,000-year period. Neptunium-237 has a large inventory and long half-life; however, it is released more slowly from the waste and travels more slowly because of retardation by adsorption. As a consequence, neptunium-237 is not a significant contributor to dose until after 10,000 years. Plutonium (represented by plutonium-239 and plutonium-242) has a very large inventory but normally is released very slowly from the waste because of low solubility and travels very slowly because of high adsorption. These two factors can both potentially be reversed because of colloidal mobilization and transport. With the models and assumptions in the TSPA-VA base case, plutonium is very mobile in colloidal form a small percent of the time. Plutonium-239 has the greater inventory, but it has a half-life of only 24,000 years so it is only important for the first 100,000 years or so.

The timing of the peak dose rates varies for each of the three simulation periods. Figure 4-30 presents scatter plots of the peak dose rate and the time at which peak dose rate occurs for each simulation period. In the plots, each symbol represents a single realization. The scatter plot of 10,000-year peak-dose times (see the top of Figure 4-30) shows that doses do not occur before 3,000 years and that peak dose rates are spread relatively uniformly throughout the remaining 7,000 years. Realizations having later peak times tend to have more waste packages in contact with seeps and therefore more corrosion failures. Juvenile waste package failures may lead to early releases, whereas later releases and doses are generally controlled by waste package failures that occur because of corrosion. Later-peak-dose realizations also tend to have lower net infiltration and, therefore, longer unsaturated zone travel times. Thirty-one realizations reach an apparent peak dose rate at 10,000 years, as indicated by the numerous symbols along the vertical line at 10,000 years; these realizations have not reached a true peak and will continue to increase if the simulation time is extended.

As shown in the middle plot of Figure 4-30, the peak-dose times in the 100,000-year period are clustered in two areas, with few points in between. Before 30,000 years, 16 percent of the realizations have their peak dose rate, and 56 percent of the realizations have their peak after 70,000 years, plus 20 percent have zero dose for the whole period. Many of the early peak dose rates, and especially those before 10,000 years, have only juvenile waste package failures and no corrosion failures for the first 100,000 years. The peak dose rates before 10,000 years are all very low, less than 0.1 mrem/year. For the 100,000-year period, peak-dose times are influenced by the number of packages in contact with seepage water and by waste package corrosion rate. Peak-dose times again tend to increase as both the number of packages in contact with seepage increases and the number of package failures increases. The two are closely related. However, percolation flux through the unsaturated zone and the resulting effect on travel time have less of an influence on peak-dose

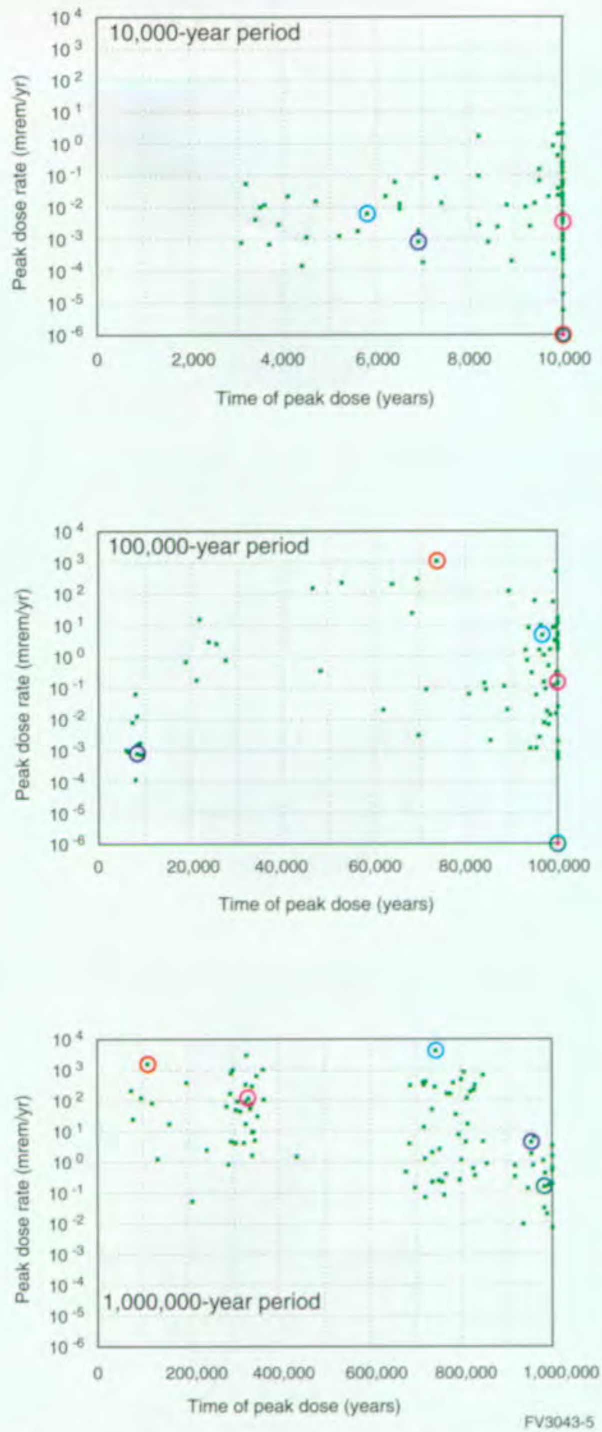

Figure 4-30. Peak Dose Rate Versus Time of Peak Dose Rate for Three Periods
Pink dots at the bottom indicate realizations with zero dose (which are off the scale of the log plot). The colored circles refer to the selected realizations highlighted in Figures 4-26 and 4-27.

time in this longer simulation period. Again, several realizations reach an apparent peak dose rate at 100,000 years, indicating that these realizations have not reached an actual peak during the 100,000-year simulation period.

The clustering of peak dose rates in particular time periods is even more pronounced in the 1-million-year simulation, as shown in the bottom plot of Figure 4-30, with 39 percent of the peaks occurring prior to 400,000 years and 57 percent occurring after 700,000 years. This behavior is mainly because of the climate model, with most of the peak dose rates associated with superpluvial climates. The figure shows two distinct clusters of points: one cluster at about 330,000 years and the other cluster at about 780,000 years. These times correspond to the two superpluvial climates, with the scatter in peak-dose time largely because of the scatter in the sampled time of the superpluvial climates.

The effect of climate changes can be seen clearly in the 1-million-year dose-rate history curves (Figure 4-27, bottom), in the regularly spaced spikes on the curves. These spikes are caused by the climate changes that occur roughly every 100,000 years. The most extreme climate changes occur at the superpluvials, which happen at approximately 300,000 years and 700,000 years. The spikes at those times are larger than the other climate-change spikes, leading to the predominance of those times in the peak-time scatter plot (Figure 4-30, bottom). The climate history for one of the realizations is shown in Figure 4-31, to show that the spikes in the dose-rate curves line up with the times of climate changes. However, the climate changes are at slightly different times in different realizations.

As for the 10,000-year and 100,000-year periods, waste package corrosion rate and seepage fraction influence peak-dose times, but here the relationship is reversed, with higher corrosion rates and more seepage tending to produce earlier peak dose rates. This difference can be explained as follows. Waste package failures and dose rates are still increasing at 100,000 years in most realizations; realizations with higher waste package

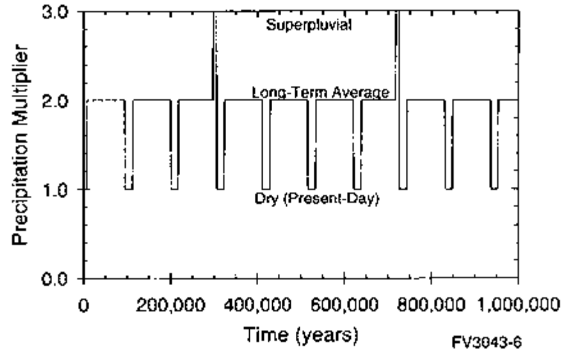

Figure 4-31. Climate History for One Realization
The others look similar. Most of the peak dose rates occur at one of the two superpluvial climates.

failure rates, influenced both by nickel-base Alloy 22 corrosion rate and the fraction of waste packages contacted by seeps, tend to have dose rates increasing faster as well, so their 100,000-year peak is at the end of the period. Those realizations then tend to have their 1-million-year peak dose rates at the earlier superpluvial. However, realizations with low corrosion-failure rates because of low sampled Alloy 22 corrosion rate or low seepage fraction, tend to have their 10,000-year and 100,000-year peak dose rates early because of juvenile waste package failures. These realizations then achieve their 1-million-year peak dose rates late in the period (or even after 1 million years), after they finally have significant numbers of corrosion-failed waste packages.

To further illustrate some of the key processes that affect calculated dose rates and their peak times, five realizations were selected for more detailed discussion. The five realizations were chosen to have a spread of peak dose rates and peak-dose times and, therefore, a variety of different behaviors. The selected realizations are indicated with different colors in Figure 4-30, and their time histories are highlighted with the same colors in Figures 4-26 and 4-27.

Additional information is given in Figure 4-32, which shows the number of failed waste packages over time for the selected realizations. In Figure 4-32, the pink and blue realizations have one juvenile waste package failure, the cyan realization has two juvenile failures, and the red

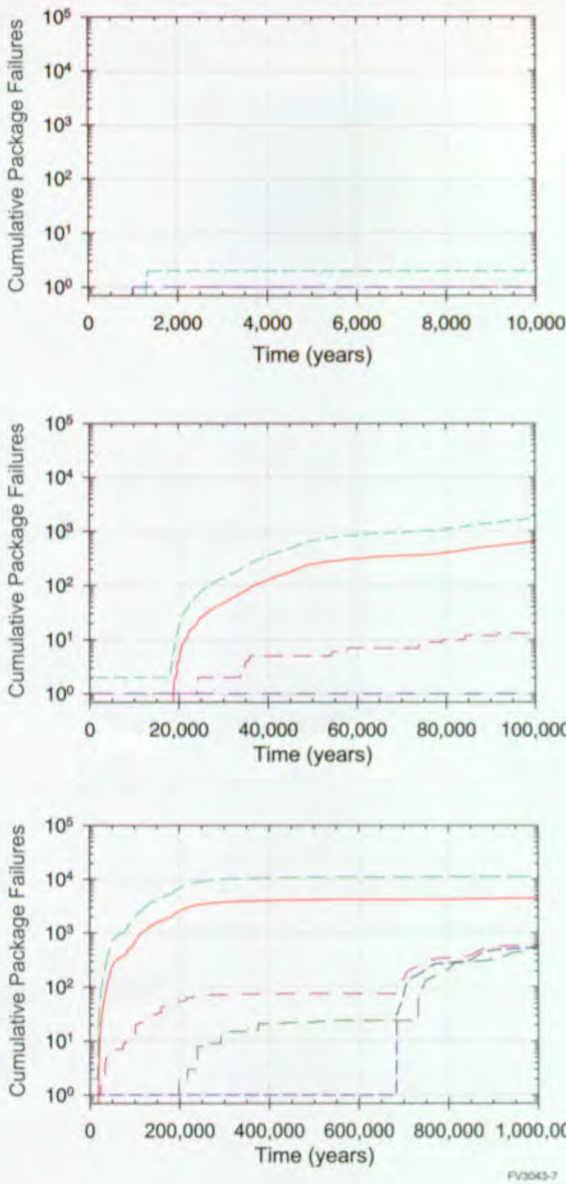

Figure 4-32. Number of Failed Waste Packages over Time for the Selected Realizations
The realizations were chosen to have a spread of peak dose rates and peak dose times and, therefore, a variety of different behaviors.

and dark green realizations have no juvenile failures. The dark green realization has no waste package failures for nearly 200,000 years. The blue realization has no additional waste package failures other than one juvenile failure for almost 700,000 years. The pink realization has a relatively small number of additional failures, and the red and cyan realizations have many corrosion failures. These numbers of waste package failures are strongly reflected in the relative dose rates in Figure 4-26. The blue realization has a low dose rate throughout the entire 100,000-year period, the pink realization is higher, and the red and cyan realizations are higher yet. However, the number of waste package failures is not the only determinant of dose, because the red realization has the highest dose rate even though the cyan realization has more waste package failures. The primary reason that the red realization has such a high peak dose rate even though it does not have the greatest number of waste package failures is that it is one of the realizations with highly mobile plutonium colloids; 97 percent of its peak dose rate is from plutonium-239. The significant falloff in dose rate after 100,000 years in that realization (Figure 4-27, bottom) is because of plutonium-239 decay.

Figure 4-32 shows that many realizations have an increase of waste package failures starting at about 700,000 years. In fact, the red and cyan realizations have a similar increase in the number of failures but it cannot be seen on the log scale because they already have so many waste package failures. The explanation for this behavior is that around 700,000 years the waste packages with no seeps dripping on them start failing. Until that time, only waste packages contacted by seeps have failed. Because the waste packages failing after 700,000 years are dry, the failures are not reflected in significantly increased dose rates in some cases (Figure 4-27, bottom). Dry waste packages have lower rates of release (diffusive only) than wet ones (diffusive plus advective). The fraction of waste packages contacted by seeps is reflected directly in the waste package failure curves. For example, the cyan realization has a seepage fraction for the long-term-average climate of 95 percent, which allows almost all of the waste packages to fail within 300,000 years. The number of failed packages is over 10,000. The red realization has a seepage fraction for the long-term-average climate of about 36 percent, so only about a third of the waste packages fail before 300,000 years in that realization. The other three selected realizations all have seepage fraction less than one percent for the long-term-average climate, so less than one percent of the waste packages fail in those realizations, until the dry ones start failing at late times.

4.3.1.2 Precision of the Base Case Complementary Cumulative Distribution Functions

The number of realizations to use in a Monte Carlo simulation is a significant issue in terms of reliable analyses and proper allocation of resources. Therefore, the number of runs required to reliably predict peak dose rates was examined. To verify whether 100 realizations are sufficient, the 10,000-year and 100,000-year base case simulations were repeated with 1,000 and 300 realizations, respectively. The resulting distributions of peak individual dose rates are compared with the 100-realization base case results in Figure 4-33—the distributions for each time period nearly overlay. The 100-realization distributions do not go below a probability of 0.01 because each predicted dose rate has a probability of occurrence of 1/100, or 0.01. Similarly, the 1,000- and 300-realization distributions display minimum probabilities of 0.001 and 0.003, respectively. In Figure 4-33, peak dose rates do continue to increase as probability decreases. Increased dose rates at these low probabilities are caused by combinations of extreme uncertain-parameter values that are sampled from the tails of the parameter probability distributions. However, 100 realizations appear to provide a good compromise between cost and precision.

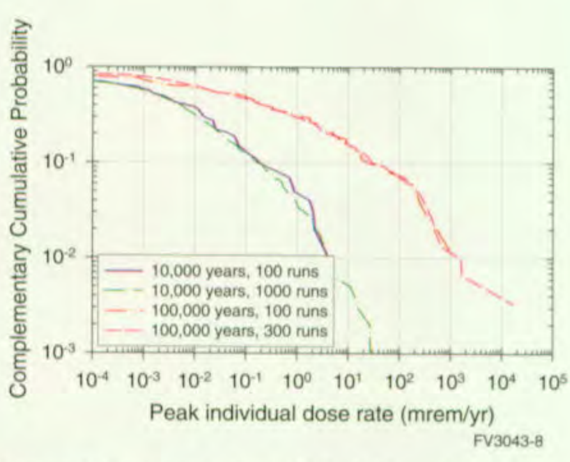

Figure 4-33. Comparison of Peak Dose Rate Distributions Determined from Simulations with Different Numbers of Realizations
The distributions are found to be quite similar, indicating reasonable stability with respect to number of realizations.

4.3.2 Sensitivity Analysis

The primary objective of sensitivity analysis is to identify those uncertain parameters that have a strong influence on the repository performance measures and to quantify the strength of their influence. The simplest tool for sensitivity analysis is the scatter plot, which is a plot of sampled numerical values of an uncertain variable, or independent variable, used in the computations versus the calculated results such as peak dose rate, or dependent variable. If little or no relationship exists between an independent-dependent variable pair, their scatter plot will resemble a random distribution of points. However, if a notable relationship exists between them, the plotted points will cluster and exhibit a recognizable form.

Scatter plots are valuable for identifying a relationship between variables, but they do not quantify the intensity of that relationship. Multiple linear regression modeling is used to quantify relationships between dependent and independent variables. By using a regression model, it is possible to identify variables that contribute most to the calculated uncertainty in the peak dose rates. A good discussion of multiple linear regression modeling and other sensitivity-analysis techniques, and their application to the performance assessment of the Waste Isolation Pilot Plant, is given by Helton (1993).

For the TSPA-VA, the primary technique for regression modeling is stepwise linear regression. In the stepwise approach, a sequence of regression models is constructed starting with a single selected input parameter, and including one additional input variable at each successive step until all significant input variables have been included in the model. This approach avoids having to treat all of the independent uncertain variables simultaneously in a single model.

A measure of goodness of fit of a linear-regression model is provided by the coefficient of multiple determination, R^2. An R^2 value near 1 indicates a good fit, meaning that the model is accounting for most of the uncertainty in the performance measure being analyzed. To avoid poor linear fits with nonlinear data, regression analyses were performed on rank-transformed data. By rank-transformed data it is meant that the actual data values of both outputs and inputs are replaced with their corresponding ranks; that is, the smallest value for a given variable is assigned the value of 1, and the next largest value is assigned a value of 2, and so on up to the largest value.

Two importance indicators are used in the TSPA-VA sensitivity analysis, partial rank correlation coefficient and R^2-loss. Both of these indicators are outputs from stepwise regression modeling. The partial rank correlation coefficient for a particular input variable measures the correlation between the output and the selected input variable, after the linear influences of the other variables in regression have been eliminated (Helton 1993, p. 352). The square of a partial rank correlation coefficient also represents the gain in R^2 of the regression model, expressed as a fraction of the currently unexplained uncertainty, as the selected variable is brought into regression (RamaRao et al. 1998). Partial rank correlation coefficients are also useful because their sign (+ or −) indicates whether the selected input variable has a positive or negative effect on the performance measure. The second importance indicator used, R^2–loss, represents the loss in R^2 of the current n–variable regression model, if the variable of concern is dropped from the regression model, creating a model with n–1 variables, where n is the total number of variables in the final regression model. Therefore, a high value of R^2–loss indicates that the removed variable is important.

Parameter rankings are based on R^2-loss values that were computed by stepwise rank regression analysis. All parameters with an R^2-loss greater than or equal to 0.04 are reported. Partial rank correlation coefficients are also provided to indicate if a parameter has a negative or positive effect on the performance measure in question. A negative effect on a performance measure indicates that as the uncertain input parameter increases, the performance measure decreases. In contrast, if an input parameter has a positive effect, the performance measure increases as the input parameter increases.

4.3.2.1 Sensitivity Analysis Results

Sensitivity-analysis results are presented for 10,000-year peak dose rates at the top of Figure 4-34. Parameters most important to uncertainty in peak dose rates are depicted in bar-chart form with the most important parameter represented by the largest bar, the next most important variable represented by the next largest bar and so on. The important variables are as follows:

1. Fraction of waste packages contacted by seepage water
   (R^2–loss = 0.18; partial rank correlation coefficient = 0.68)

2. Mean corrosion rate of Alloy 22
   (R^2–loss = 0.13; partial rank correlation coefficient = 0.62)

3. Number of juvenile waste package failures
   (R^2–loss = 0.11; partial rank correlation coefficient = 0.60)

4. Saturated zone dilution factor
   (R^2–loss = 0.04; partial rank correlation coefficient = −0.42)

The regression model from which these results are drawn has a total of ten input variables with a total R^2 of approximately 0.80. While a higher R^2 would be desirable, it is considered adequate for the purpose of inferring the importance of the dominant input variables.

Regression results for the 100,000-year peak dose rates are presented next, in the middle chart of Figure 4-34. The important variables are as follows:

1. Fraction of waste packages contacted by seepage water
   (R^2-loss = 0.29; partial rank correlation coefficient = 0.77)

2. Mean corrosion rate of Alloy 22
   (R^2-loss = 0.20; partial rank correlation coefficient = 0.70)

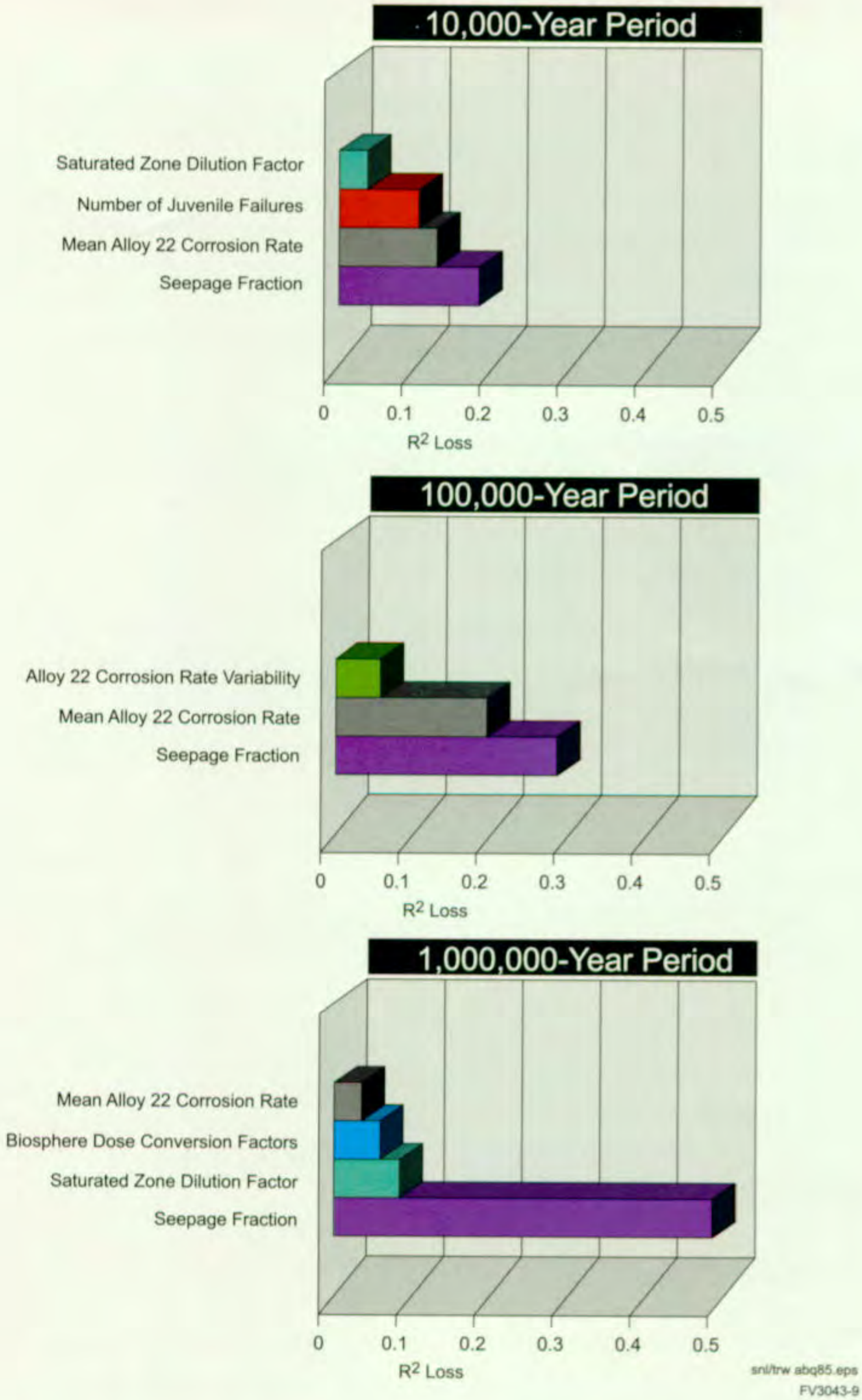

Figure 4-34. Most Important Parameters from Stepwise Regression Analysis for Three Periods
These charts show the relative importance of various parameters to the calculated uncertainty in peak dose rate for the three time periods. Importance of an individual parameter is shown by R^2-loss, the reduction in goodness of the regression fit when the parameter is left out of the calculation, as described in the text.

3. Variability of the Alloy 22 corrosion rate
(R^2-loss = 0.06; partial rank correlation coefficient = 0.49)

The regression model for this case also has ten variables and an R^2 of 0.80.

Peak-dose-rate regression results for a 1 million-year period are presented in the bottom chart of Figure 4-34. The important variables are as follows:

1. Fraction of waste packages contacted by seepage water
(R^2-loss = 0.51; partial rank correlation coefficient = 0.86)

2. Saturated zone dilution factor
(R^2-loss = 0.09; partial rank correlation coefficient = –0.56)

3. Biosphere dose-conversion factors
(R^2-loss = 0.07; partial rank correlation coefficient = 0.51)

4. Mean corrosion rate of Alloy 22
(R^2-loss = 0.04; partial rank correlation coefficient = 0.41)

The regression model for this case has ten variables and a final R^2 of 0.82.

For peak dose rates, the fraction of waste packages contacted by seepage water is the most important parameter in each of the three simulation periods, and its effect becomes more dominant over time. The positive effect indicated for this variable by its positive partial rank correlation coefficient happens because an increase in the number of waste packages contacted by water leads to an increase in radionuclide releases from the repository. The reason for the great importance of seepage fraction is two-fold. First, it has a direct effect on the number of waste packages that fail, as discussed in Section 4.3.1.1; and second, it has a very large uncertainty (see Figure 3-13). Other parameters either have less impact on dose rate or less uncertainty.

The second most important variable in the 10,000-year and 100,000-year cases is the mean corrosion rate for Alloy 22, the inner barrier in the reference-design waste package. The positive effect for this parameter arises because increasing the Alloy 22 corrosion rate produces earlier waste package failures and, therefore, more of the waste packages fail and release radionuclides within 10,000 years or 100,000 years. The corrosion rate for Alloy 22, however, is less important in the 1-million-year simulation period as indicated by its relatively low R^2-loss value in the bottom chart of Figure 4-34. This reduced importance is because the majority of the wet waste packages fail during the 1-million-year period, even for low Alloy 22 corrosion rates; failures of dry waste packages are only starting to be a factor near the end of the million-year period (see the discussion in Section 4.3.1.1). As a consequence, waste package parameters appear to become less important in this longer simulation period because the quantities of radionuclides released from the repository are primarily controlled by the inventory, the fraction of waste packages in contact with seepage water, and the seepage flow rate into the waste packages.

The second most important variable in the 1-million-year case is the saturated zone dilution factor. This factor accounts for the reduction in radionuclide concentrations caused by transverse dispersion that occurs during transport through the saturated zone. The negative effect indicated for this variable (partial rank correlation coefficient = –0.56) is caused by the fact that, as dilution during transport increases, peak radionuclide concentrations at the 20-km (12-mile) distance decrease and, consequently, doses decrease.

The next most important variables in the 10,000-year and 100,000-year simulation periods are the number of juvenile waste package failures and variability in the inner-barrier Alloy 22 corrosion rate, respectively. Both of these waste package variables have a positive effect on peak dose rates because, as noted previously for increased Alloy 22 corrosion rates, they lead to earlier waste package failures and, therefore, increased radionuclide releases from the repository during the given time period. The third most important variable for the 1-million-year simulation period is the biosphere dose-conversion factor. Note that the biosphere dose-conversion factors for all radionuclides were correlated for the TSPA base case; that is, they are all either low or high together, rather than allowing one radionuclide to be high while another is low (see Section 3.8.2).

Examination of scatter plots provides an additional way to visualize the effects of uncertain variables on peak dose rates. Two important parameters discussed above are fraction of waste packages contacted by seepage water and Alloy 22 mean corrosion rate. Scatter plots for these two parameters are shown in Figures 4-35 and 4-36 for the 100,000-year simulation period.

Although there is a lot of scatter in the values, a pattern of higher dose rates for higher parameter values is clearly discernible in both scatter plots. As suggested by the regression-analysis results discussed above, the relationship between fraction of waste packages contacted by seepage water and peak dose rate is even more well defined for the 1-million-year simulation period, as shown in Figure 4-37. This strong relationship is reflected in the high R^2-loss value of 0.51 as compared to the

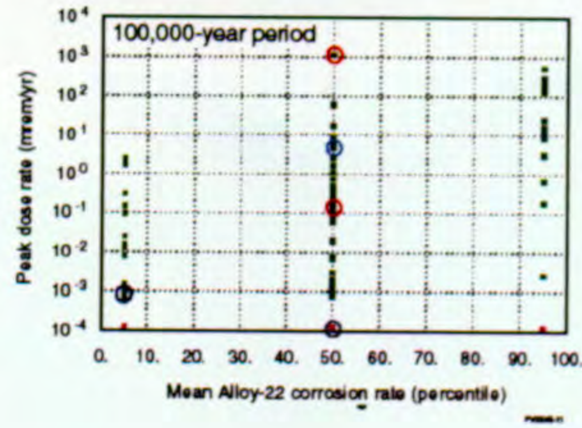

Figure 4-36. Scatter Plot of Mean Alloy 22 Corrosion Rate over 100,000 Years
Scatter plot of mean Alloy 22 corrosion (in terms of percentile from the uncertainty distribution) against peak dose rate for a 100,000-year period. Pink dots at the bottom indicate realizations with zero dose (which are off the scale of the log plot). The colored circles refer to the selected realizations highlighted in Figures 4-26 and 4-27.

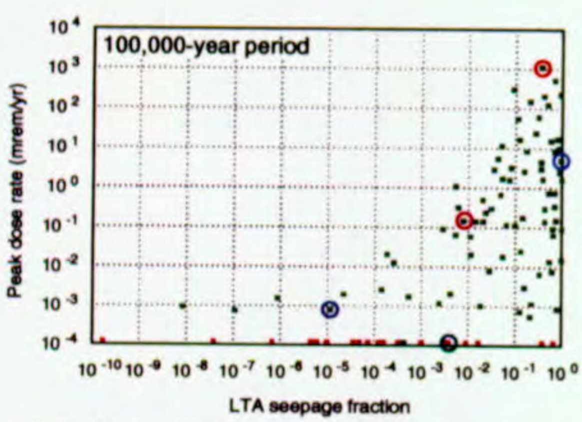

Figure 4-35. Scatter Plot of Seepage Fraction over 100,000 Years
Scatter plot of seepage fraction (fraction of waste packages contacted by seeps) for the long-term-average climate against peak dose rate for a 100,000-year period. Pink dots at the bottom indicate realizations with zero dose (which are off the scale of the log plot). The colored circles refer to the selected realizations highlighted in Figures 4-26 and 4-27.

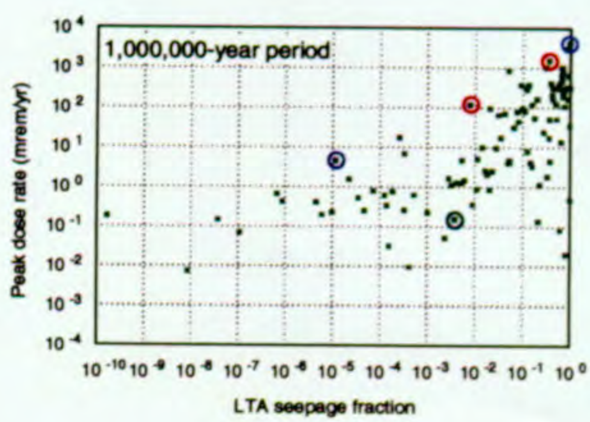

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Figure 4-37. Scatter Plot of Seepage Fraction over 1 Million Years
Scatter plot of seepage fraction (fraction of waste packages contacted by seeps) for the long-term-average climate against peak dose rate for a 1 million year period. The colored circles refer to the selected realizations highlighted in Figures 4-26 and 4-27.

R^2-loss value of 0.29 in the 100,000-year simulation period.

Dose rates are actually time-dependent quantities. As described in Section 4.3.1, a peak dose rate represents the highest dose rate that occurred during the simulation period for a given realization. The sensitivity analysis results presented thus far are for the peak dose rate only and, therefore, do not give explicit information about how parameter importance changes with time, although some insight is gained about changes with time by considering three different periods. To examine more directly how parameter importance changes with time, regression analyses were performed on dose rates at selected simulation times. The resulting values of partial rank correlation coefficients plotted as a function of time are presented in Figure 4-38. This figure shows that the number of juvenile waste package failures is the dominant variable at early times. However, over the long term, the importance of early releases decreases as more waste packages fail. The fraction of waste packages contacted by seepage water begins to increase in importance at around 5,000 years and becomes the dominant variable after approximately 8,000 years. The Alloy 22 corrosion rate also begins to increase in importance at around 5,000 years and remains important throughout the 1-million-year time period, with its importance decreasing somewhat

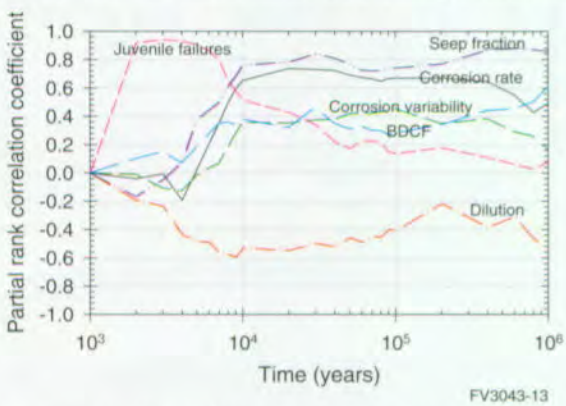


Figure 4-38. Partial Rank Correlation Coefficient as a Function of Time for Six Uncertain Parameters
This figure illustrates another way to show the change of importance of parameters through time from that shown in Figure 40-34. (BDCF—biosphere dose conversion factor)

after 400,000 years. Biosphere dose-conversion factors increase greatly in importance at late times.

The saturated zone dilution factor appears in the time-dependent regression model with a negative partial rank correlation coefficient because of its role in reducing radionuclide concentrations and dose rates. Some of the other variables also have small negative partial rank correlation coefficients before 5,000 years. These small negative values are probably spurious (that is, arising simply because of chance correlations among the randomly sampled variables) because they correspond to R^2-loss values smaller than 0.005 and because it is unlikely that these variables have a negative effect on dose rate. Note that the values plotted in Figure 4-38 do not correspond to the values listed earlier because the ones in the figure are for correlations with dose rate at particular times, whereas the earlier ones were for correlations with the peak dose rate over a period of time.

Regression analyses have also been performed for intermediate performance measures, including radionuclide releases from the engineered barrier system to the unsaturated zone and from the unsaturated zone to the saturated zone. These performance measures are also time-dependent and can be presented in the same way that the time-dependent dose rates were presented. Analyzing intermediate performance measures is useful for isolating individual components of the total system and assessing the importance of variables among the smaller corresponding set of uncertain parameters. As an example, results are presented for releases from the engineered barrier system to the unsaturated zone in Table 4-1. Results for releases from the unsaturated zone to the saturated zone are similar because they are not strongly influenced by uncertain unsaturated zone transport parameters. The results shown are for the specific simulation time of 1 million years. Again, the dominant variable is the fraction of waste packages contacted by seepage water. The second and fourth variables, the number of cladding failures because of corrosion and the fraction of plutonium colloids transported with irreversible sorption, did not appear to have a significant effect on dose rate but

Table 4-1. Importance Ranking of Inputs for Engineered Barrier System Releases at 1 Million Years

Rank	Variable	R^2–loss	Partial Rank Correlation Coefficient
1	Fraction of waste packages contacted by seepage water	0.45	0.84
2	Number of cladding failures because of corrosion	0.07	0.52
3	Alloy 22 mean corrosion rate	0.06	0.51
4	Fraction of plutonium colloids transported with irreversible sorption	0.04	0.43

Regression Model: Input variables = 11, R^2 = 0.82

are important to releases from the engineered barrier system.

4.3.2.2 Impact of Parameter Uncertainty Ranges on Sensitivity Analyses

Sensitivity analyses based on linear regression must be carefully interpreted. A regression-based sensitivity analysis will not give an uncertain variable a high importance ranking if its sample input values do not produce a relatively wide range of calculated output values. In other words, a parameter will not be assigned a high importance ranking unless it accounts for a large fraction of the variance in the output measure being analyzed. Therefore, the magnitude of peak dose rate may be strongly affected by a certain parameter, but if the range of uncertainty associated with that parameter is relatively small, the parameter may not be found to be important. As an example, neptunium solubility was not found to be an important variable in the above analysis, but because neptunium-237 often dominates the calculated dose rates (see Figure 4-29) neptunium solubility clearly does significantly influence total peak dose rate. The reason neptunium solubility did not appear as an important variable in the regression-based sensitivity analysis is largely due to the wide uncertainty ranges associated with seepage fraction and waste package corrosion.

One approach to understanding the effects of parameters that have very wide uncertainty ranges is to perform sensitivity analyses in steps. That is, perform an initial regression analysis, such as the one presented in Section 4.3.2.1, and identify important uncertain parameters. Then, assign single values, such as mean or median values, to some or all of the important uncertain parameters identified in the first step and repeat the analysis. By assigning single values to uncertain parameters identified in the first step, important parameters with less uncertainty may be identified in the second step.

This type of phased approach to sensitivity analysis was conducted for the TSPA-VA base case. For the secondary sensitivity analysis, the "expected" values (see Sections 2.3.3.3, 4.1, and 4.2) were used for the following:

1. Infiltration and mountain-scale unsaturated zone flow
2. Seepage fraction (fraction of waste packages contacted by seepage) and seep flow rate
3. Alloy 22 mean corrosion rate and variability

As for the base case analysis, 100 realizations were simulated for each of three periods: 10,000 years, 100,000 years, and 1 million years. For all parameters other than those being held fixed, the same sample values as for the base case were used. The distributions of peak dose rates for an individual located at 20 km (12 miles) from the repository are given in Figure 4-39. Comparison with the base case distributions in Figure 4-26 shows that the modified-parameter case has a much narrower spread of peak dose rates as expected, since uncertainty in seepage fraction, corrosion rate, and infiltration are not present. Note that in the base case many realizations had no calculated dose for the

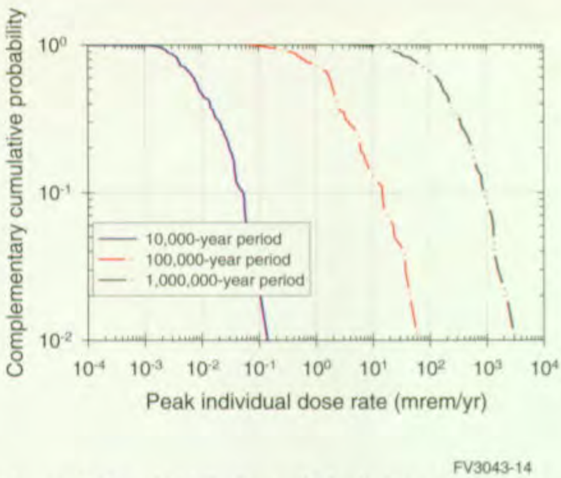


Figure 4-39. Distribution of Peak Dose Rates for Three Time Periods for the Modified-Parameter Case
Because some of the most important parameters were held fixed in this case, the spread of peak doses is much less than in the base case (Figure 4-26).

first 100,000 years, but in the modified-parameter case all realizations produced a dose within the first 10,000 years. Note that if this process of assigning expected values to important variables and conducting simulations were repeated a number of times, the distribution of peak dose rates would continue to narrow and converge towards the expected-value dose rate.

Regression-based sensitivity-analysis results are presented in Figure 4-40. Parameters most important to uncertainty in peak dose rates are depicted in bar-chart form with the most important parameter represented by the largest bar, the next most important variable represented by the next largest bar and so on. Saturated zone dilution factor and biosphere dose-conversion factors appeared on the base case bar chart (Figure 4-34), but in this analysis they account for more of the uncertainty because the most important parameters in the base case are held constant. In addition, a number of parameters are listed in Figure 4-40 that were not revealed as contributing significantly to the base case peak-dose uncertainty. The new parameters are the solubilities of technetium (for 10,000 years) and neptunium (for 100,000 years), the seepage collection factor (for 10,000 years and 100,000 years), the fraction of the saturated zone transport path in alluvium (for 100,000 years and 1 million years), the fractions of Zircaloy cladding failing because of corrosion and mechanical stresses (for 1 million years), and the saturated zone longitudinal dispersivity (for 1 million years).

The effects of these parameters on doses are straightforward. An increase in solubility can lead to faster dissolution and an increase in radionuclide releases from the repository. The seepage collection factor is used to adjust the amount of seepage water that gets into waste packages and contacts waste (see Sections 5.5.1 and 5.4.3); higher values of this factor lead to higher releases for solubility-limited radionuclides. The fraction of the saturated zone transport path in alluvium affects doses because of significantly higher radionuclide sorption in alluvium relative to the volcanic units, so that more transport through alluvium can lead to lower doses. The two cladding-failure parameters help determine the amount of commercial spent nuclear fuel exposed to water; higher values mean more waste exposed and thus the potential for higher releases and higher doses. (Note, however, that their effect is time-dependent, with most of the corrosion and mechanical cladding failures occurring after 100,000 years; see Section 3.5.2.1.) Lastly, longitudinal dispersivity in the saturated zone affects doses because higher dispersivity implies more spreading during transport and therefore lower concentrations and lower doses. Note that transverse dispersion has much greater potential for lowering doses than does longitudinal dispersion; transverse dispersion is accounted for in the analyses by means of the saturated zone dilution factor (see Section 3.7.2).

4.3.3 Summary

The Monte Carlo method is used to propagate parameter uncertainty through to the uncertainty in peak dose rate to an individual. The uncertainty in calculated peak dose rates is quantified by the CCDF. The range or spread of peak dose rates in each distribution represents the amount of uncertainty in the results. Some statistics of the calculated peak dose rates for each base case simulation period are summarized in Table 4-2.

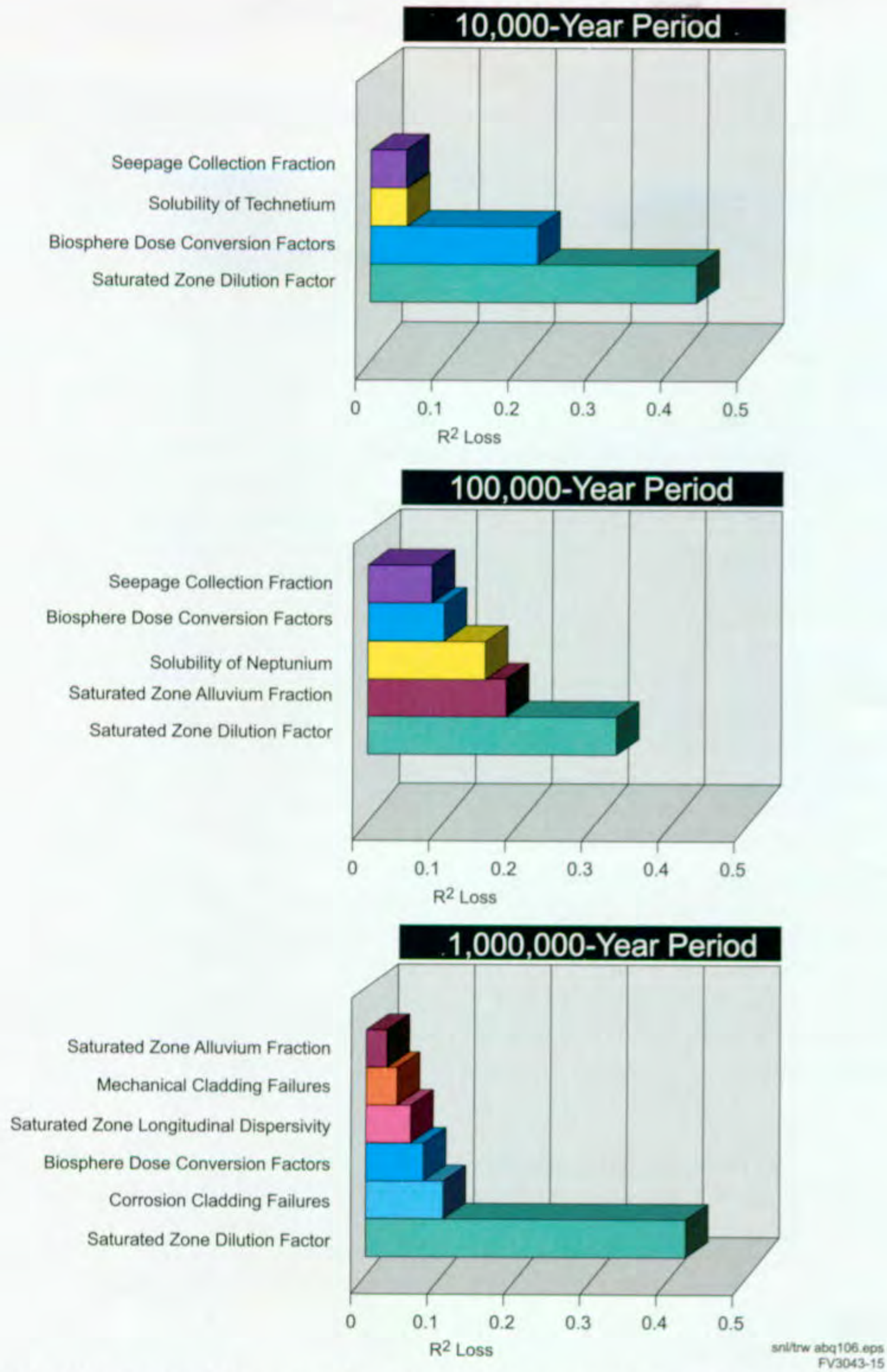


Figure 4-40. Most Important Parameters from Stepwise Regression Analysis for the Modified-Parameter Case
These charts show the relative importance of various parameters to the calculated uncertainty in peak dose rate when some of the most important parameters are held fixed. Importance of an individual parameter is shown by R^2-loss, the reduction in goodness of the regression fit when the parameter is left out of the calculation.

Table 4-2. Summary of Calculated Peak Dose Rates at 20 km (12 miles)

Simulation Period (years)	5th Percentile (mrem/year)	Mean (mrem/year)	50th Percentile (mrem/year)	95th Percentile (mrem/year)	Coefficient of Variation
10,000	0	0.1	0.002	0.8	3.8
100,000	0	30	0.09	200	4.4
1,000,000	0.07	200	8	1,000	2.8

As shown, the peak dose rates are found to increase substantially with increasing simulation time. However, the coefficient of variation, or standard deviation divided by mean, indicates that relative uncertainty in the calculated dose rates does not increase appreciably in the longer simulation periods (that is, uncertainty in dose rate increases and average dose rate increases, but their ratio does not). Thus, uncertainty in peak dose rate does not increase without bound with increased time.

The times at which peak dose rates occur were examined and found to be influenced by a number of factors, including the fraction of waste packages contacted by seepage water, Alloy 22 corrosion rate, net infiltration rate, juvenile waste package failures, and seepage flux into the waste packages. Very early dose-rate peaks are associated with juvenile waste package failures, and late dose-rate peaks are associated with superpluvial climates.

The sensitivity-analysis results indicate that the uncertainty in calculated peak dose rates in all three simulation periods is primarily dominated by the fraction of packages in contact with seepage water. Waste package corrosion rate is also an important contributor to uncertainty in peak dose rates. In particular, the mean Alloy 22 corrosion rate is an important contributor to uncertainty, but to a lesser extent than the fraction of waste packages contacted by seepage water, particularly in the 1-million-year period. In the 1-million-year simulation, the saturated zone dilution factor and biosphere dose-conversion factors were found to be more important than the mean Alloy 22 corrosion rate.

4.4 EFFECTS OF DISRUPTIVE EVENTS

Previous work has identified potentially disruptive events that might affect long-term repository performance (DOE 1988, Section 8.3.5.13; Barnard et al. 1992, Chapters 6 and 7; Wilson et al. 1994, Chapters 16 and 17). The four disruptive events considered for TSPA are basaltic igneous activity, seismic activity, nuclear criticality, and inadvertent human intrusion. The methods used to identify and select the disruptive events are described in the first part of this section. How these events are used in this TSPA is discussed in the remainder of the section. Chapter 10 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) discusses the details of the disruptive-events analyses.

Disruptive events are considered to have probabilities of less than 1 (their chances of occurring are less than 100 percent over the life of the repository), in contrast to the expected events and processes (for example, waste package corrosion, thermal effects, groundwater flow and transport). Generally, disruptive events are rare (volcanoes or earthquakes) and have identifiable starting and ending times, in contrast to continuous processes like corrosion. Although criticality does not have a sudden onset, it is included here as an example of an off-normal condition.

The disruptive event modeling studies and analyses presented in this section were prepared with the view of addressing various subissues of the NRC Key Technical Issues on Igneous Activity (NRC 1998d), Structural Deformation and Seismicity (NRC 1997d), and Total System Performance Assessment and Integration (NRC 1998a). Specifically, the information presented is pertinent to the igneous activity subissue on consequences of igneous activity within the repository setting, the structural deformation and seismicity subissue on seismic motion, and the TSPA integration subissue on model abstraction.

4.4.1 Initial Selection of Important Issues

In previous TSPAs, DOE has used generalized event trees for constructing disruptive scenarios (Wilson et al. 1994, Section 3.2). This approach has created a detailed understanding of the processes that could contribute to increased radionuclide releases following a disruptive event. The approach taken in past TSPAs was not intended, however, to demonstrate that these analyses comprehensively analyzed all disruptive scenarios. Therefore, for this TSPA, DOE has begun a scenario development method that will document all features, events, and processes in the analysis. Implementation of this scenario development method is incomplete for this TSPA, partly because new information must be considered as it becomes available.

The ultimate criterion for inclusion of disturbed scenarios in TSPA analyses is whether they are likely to have a significant impact on performance of the repository system over the period of regulatory concern. The criteria for considering inclusion of disturbed scenarios in TSPA analyses are

- Probability–Is the probability of the event occurring large enough that it could affect the chances that performance of the repository would suffer?
- Consequence–Are the potential consequences of the disturbance of sufficient magnitude that repository performance could be adversely impacted?
- Regulatory–Are there regulatory requirements that mandate consideration of disturbances (or specifically exclude certain disturbances)?

To date, basaltic igneous activity, seismic activity, and nuclear criticality have been identified as the disruptive features, events, and processes to be used in scenario construction, based on evaluations using the generalized event trees (Barr et al. 1993; Barr et al. 1996; CRWMS M&O 1997b). Human intrusion has also been investigated, but the method of inclusion in the TSPA for the potential licensing analyses is uncertain, pending clarification of regulatory requirements. Other disruptive events, such as meteorite impact or erosion, have been eliminated using the three criteria listed above. Section 10.2 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) discusses scenario development in detail.

4.4.2 Igneous Activity

Basaltic igneous activity includes volcanic eruptions and intrusive or extrusive events (in which molten igneous material is cooled within the earth or on the earth's surface, respectively). Volcanoes can potentially disperse radionuclides from waste packages directly into the atmosphere, where they can reach nearby populations. Igneous intrusions that penetrate the repository but do not reach the surface could potentially destroy waste packages and make the radioactive contents more readily available for transport by groundwater. Intrusions that occur in the Yucca Mountain region could potentially alter the transport of contaminants in groundwater. These scenarios have been investigated in prior TSPA analyses (Barnard et al. 1992, Chapter 7; Wilson et al. 1994, Chapter 17). The consequences of igneous activity were found to contribute less than one percent to overall repository performance measures (to either releases or doses). However, the probability of occurrence of igneous activity as estimated by the *Probabilistic Volcanic Hazard Analysis* (CRWMS M&O 1996c) is greater than the lower limit of 1×10^{-8} per year adopted by DOE as the level of concern (NRC 1995, page 3-5).

The *Probabilistic Volcanic Hazard Analysis* (CRWMS M&O 1996c) was conducted to assess the probability of disruption of the proposed repository by a volcanic event and to quantify the uncertainties associated with this assessment. The evaluations of members of a ten-person expert panel were elicited to ensure that a wide range of approaches was considered in the hazard analysis. The results of the individual elicitations were combined to develop an integrated assessment of the volcanic hazard that reflects the diversity of alternative scientific interpretations. The assessment focused on the volcanic *hazard* at the site expressed as the probability of disruption of the repository. It provides input to an assessment of volcanic risk, which expresses the probability of increased radionuclide release because of volcanic disruption.

The possible volcanic hazard to the repository is magma feeding a dike or surface eruption. The basic elements that need to be assessed to define the hazard are the spatial distribution and the recurrence rates of future volcanic events in the region. Spatial models represent the future locations of volcanic activity. Temporal models define the frequency of occurrence of volcanic activity and, therefore, the probability of occurrence. Results of the study indicate that spatial issues are more important than temporal issues.

Based on the results of the *Probabilistic Volcanic Hazard Analysis* (CRWMS M&O 1996c), a volcanic event is expected to intersect the repository footprint (the area within the repository outline) at an expected annual frequency of 1.5×10^{-8}. This frequency value has a 90-percent confidence interval of 5.4×10^{-10} to 4.9×10^{-8}. The major contributors to the uncertainty in the frequency of intersection are the statistical uncertainty in estimating the rate of volcanic events from small data sets and the uncertainty in modeling the spatial distribution of future events.

4.4.2.1 Direct Release Scenario

In the direct-release scenario, a volcanic eruption disperses contaminated ash on the ground. The components of this scenario are illustrated in Figure 4-41. The figure is constructed as a series of branch points; if an event or process occurs, the next element of the scenario is evaluated. If it does not occur, either the scenario is terminated with no adverse performance assessment consequences or another scenario is invoked.

The first branch asks whether igneous intrusion occurs in the Yucca Mountain (Figure 4-41a). This is the issue addressed by the *Probabilistic Volcanic Hazard Analysis* (CRWMS M&O 1996c), as discussed above. If igneous activity does occur, a branch on whether the igneous event causes an intrusion follows. If the intrusion is outside the repository footprint, the indirect-effects scenario is used (see Section 4.4.2.3). If the intrusion intersects the repository, the next branch asks if any waste packages are directly contacted. If the intrusion does not contact any waste packages, there is no adverse impact on performance assessment beyond that evaluated in the indirect-effects scenario.

If waste packages are directly contacted by ascending extrusive magma or by a pyroclastic flow, the next branch point is shown in Figure 4-41b. The branch determines whether the conditions allow the waste packages to be breached by the ascending hot material. The processes that can rupture a waste package include rapid corrosion by aggressive magmatic gases, deformation and rupture of the metal waste package, or blowout of the end plates from the high temperatures. The thick outer barrier of the waste package is made of materials that are very susceptible to attack by sulfur compounds normally identified in eruptions. Therefore, we assume that, during the period of eruption (typically between 5 and 40 days for igneous events in the Yucca Mountain region [Jarzemba 1997, Table II]), the outer barrier will be completely removed. The inner waste package wall is made of a superalloy composed of nickel, chromium, and molybdenum, that has been selected for its corrosion resistance. Although experiments have not directly confirmed it, the full 2-cm (3/4-in.) thickness of this superalloy is unlikely to be corroded through because the expected duration of typical Yucca Mountain eruptions appears to be too short. Only if the inner barrier has previously been degraded by normal corrosion processes will an igneous event breach the waste package. If the waste package is not breached, the scenario stops because the encapsulation of the waste packages in basalt is likely to have a beneficial effect on subsequent repository performance.

For those waste packages that are breached, the scenario investigates the mechanisms for removing waste from the opened packages. The moving

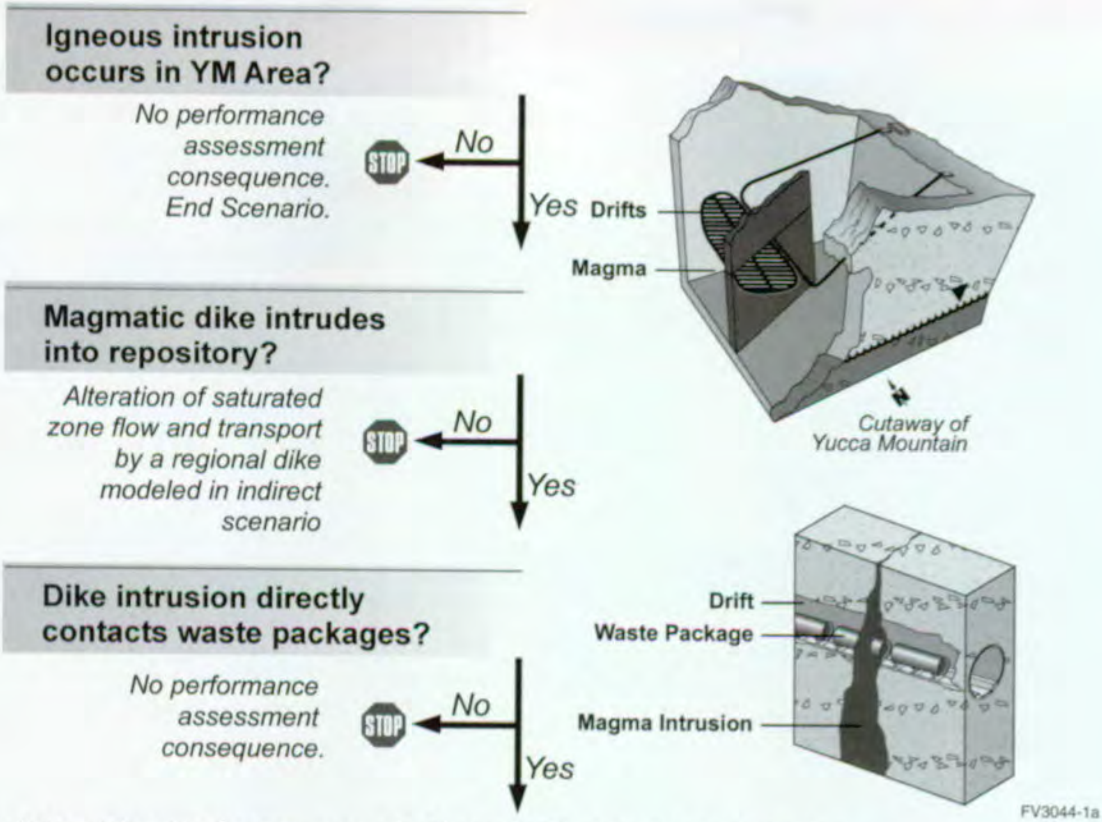


Figure 4-41a. Alternative Consequences of a Magmatic Dike Intruding the Repository
This figure illustrates some of the branch points for a dike interacting with the repository.

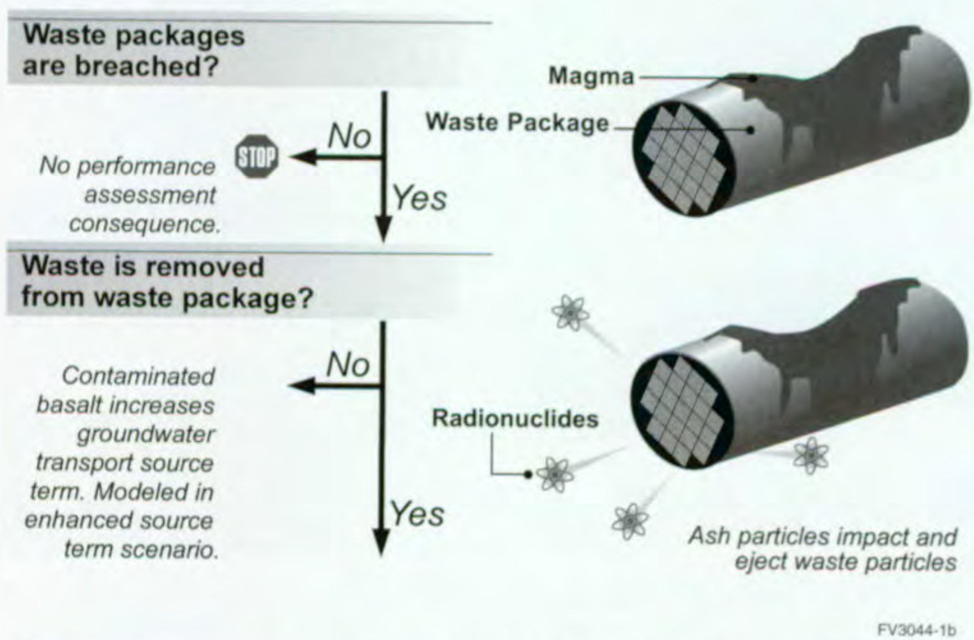


Figure 4-41b. Alternative Consequences of a Magmatic Dike Intruding the Repository-Part 2
If waste packages are directly contacted by magma or a pyroclastic flow what are the possible consequences? The chemically aggressive gases and high temperatures associated with a volcanic event could breach packages.

liquid magma or pyroclasts can cause waste to be ejected from the packages by hitting it. The density of waste material is considerably greater than that of magma (approximately 11 g/cm^3 [Glasstone and Sesonske 1991, Table 8.3] for waste and 0.8 to 2.5 g/cm^3 [Suzuki 1983, Figure 3] for magma). This density difference means that to cause waste to be ejected from the package, the magma material must be at least as large as the waste material that it hits. This relationship holds for the range of ascent velocities for magma that are typical of the Yucca Mountain area volcanoes. The process was simulated by comparing the sizes of magma fragments and waste materials. If the moving igneous material is unable to eject material from the waste package, then the enhanced-source-term scenario is examined (see Section 4.4.2.2).

The ascending magma or pyroclasts can transport the heavy waste material to the surface (see Figure 4-41c) if the upward magma velocity is faster than the downward settling velocity of the waste. An ascent velocity ranging from approximately 10 m/s (20 mph) to over 200 m/s (400 mph) is required to entrain the waste material. This component of the scenario completes the definition of the source term used by the ash-dispersal analysis. If the waste is not moved to the surface, it collects somewhere underground and could contribute to the enhanced-source-term scenario. This type of underground accumulation of waste material has not been evaluated for this TSPA.

The dose to humans from this scenario is calculated based on the dispersal of contaminants in volcanic ash. The ash-dispersal model (Jarzemba 1997, Section IV.B) uses information on eruption characteristics of volcanoes, wind direction and velocity, and ash and waste characteristics.

4.4.2.2 Enhanced Source Term Scenario

When liquid magma intersects the repository drifts, it can flow down the drifts and engulf the waste packages. In this environment of high temperatures and aggressive magmatic gases, even the corrosion-resistant parts of the waste packages are expected to fail. When this happens, the liquid

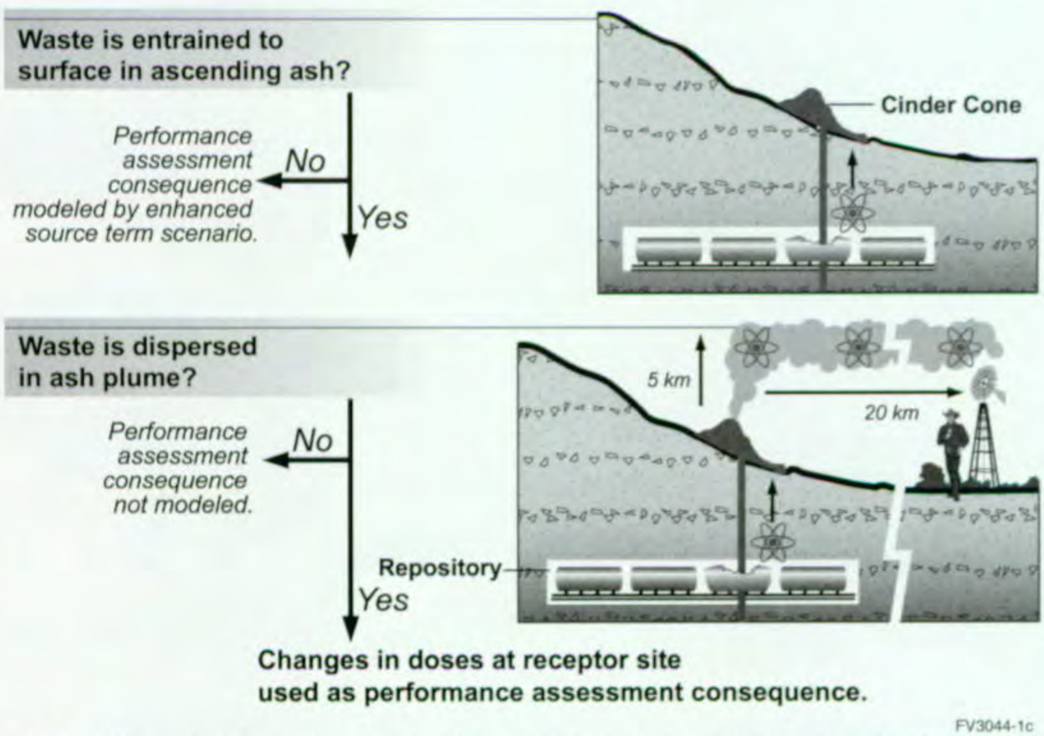


Figure 4-41c. Alternative Consequences of a Magmatic Dike Intruding the Repository-Part 3
If waste is incorporated in the eruption, what are the potential consequences? Contaminated ash may be transported to the surface and dispersed in an ash plume.

magma can dissolve some of the UO_2 in the spent nuclear fuel (Westrich 1982, Figure 3). When the basalt cools (in about 10 years for a 2-m [7 feet] wide intruded dike) groundwater returns to the drifts. As the basalt cools, it cracks, allowing the water to dissolve the UO_2 and other radionuclides in the contaminated basalt. The groundwater then carries the contaminants through the geologic media to the dose-exposure location (the location 20 km [12 miles] south of the potential repository where humans can be exposed to radionuclides). Figure 4-42 illustrates this scenario.

4.4.2.3 Indirect Igneous Effects Scenario

If an igneous event occurs near the Yucca Mountain repository, it could affect repository performance although magma does not contact the waste. This scenario is shown in Figure 4-43. As discussed in Section 3.7, groundwater flow and radionuclide transport through the saturated zone are controlled by the geologic features and their hydraulic properties. If an igneous event occurs in the saturated zone, it could potentially change flow and thus repository performance. The most likely places for dike intrusion are near sites of recent igneous or seismic activity (faults or existing dikes). The dike could change the flow pattern by

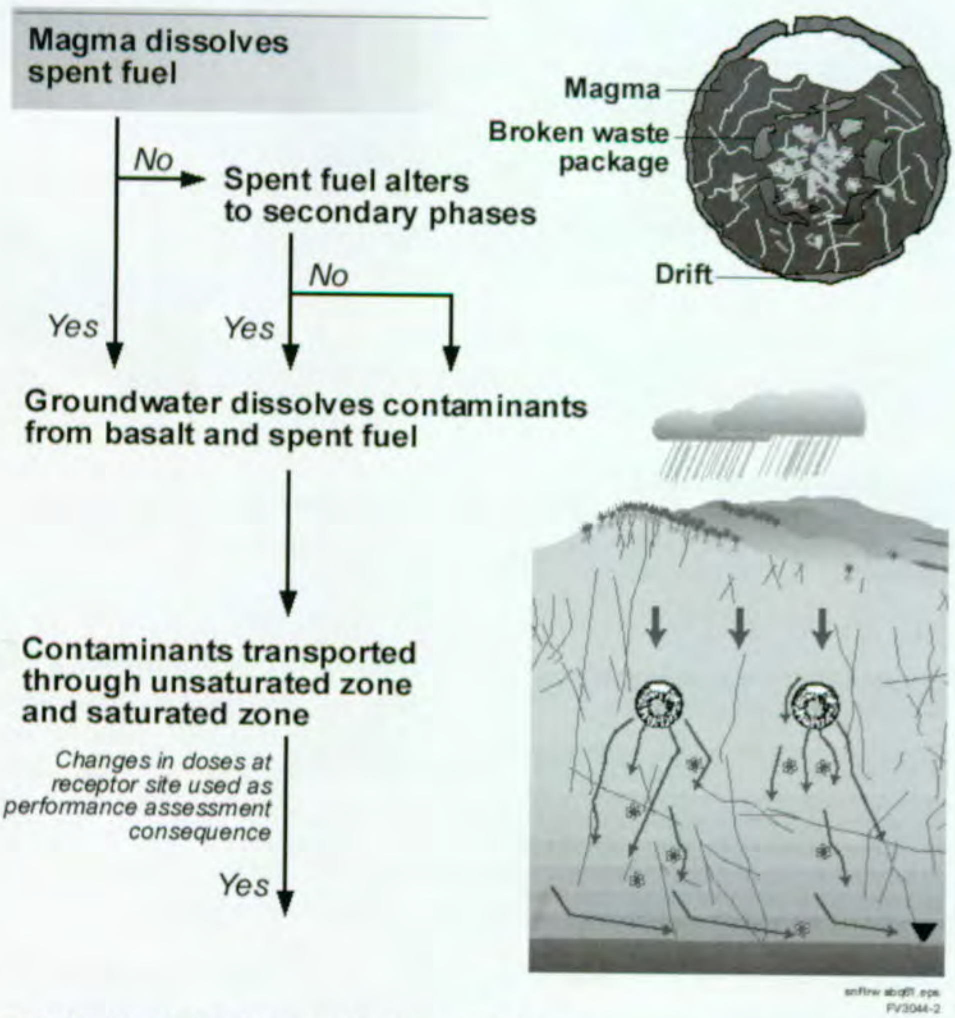


Figure 4-42. Source Term Enhanced by Igneous Activity
What are the possible outcomes if liquid magma intersects the repository, engulfing the waste packages and causing breaches? After the intrusive magma cools, and normal groundwater flow returns, radionuclides could be carried from the repository in groundwater.

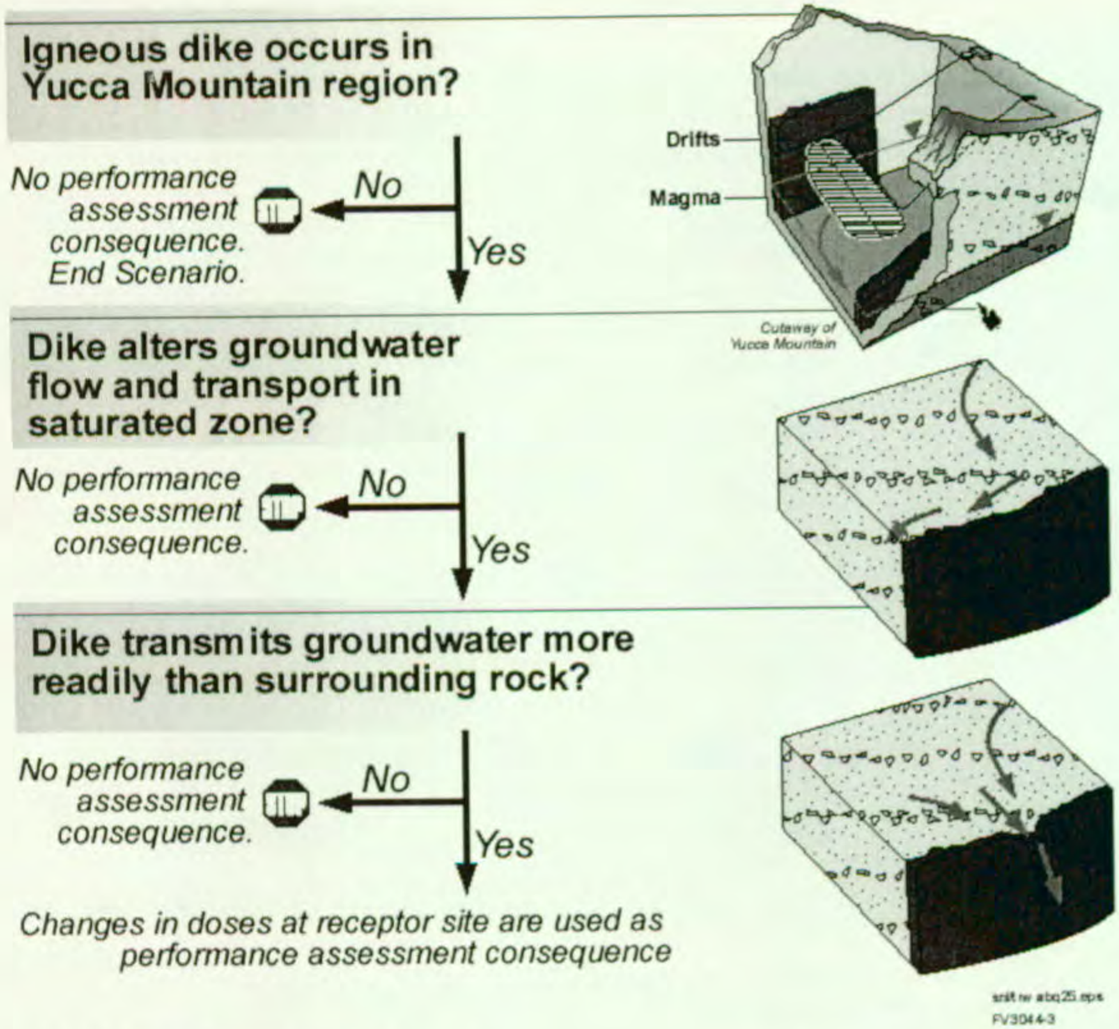


Figure 4-43. Indirect Igneous Effects Scenario
If magma intrudes near the repository, not directly contacting the waste, but possibly altering groundwater flow patterns in the area, could it affect the dose rate at the receptor site? If a magmatic dike alters the flow pattern, redirecting flow, there is the potential for an increase in radionuclide doses to humans.

redirecting flow. This change in flow could potentially increase the radionuclide doses to humans.

4.4.2.4 Results of Igneous Activity Analyses

Modeling of direct-release volcanism shows that there is very little impact from this scenario. When all the processes outlined in Section 4.4.1 are considered, less than 6 percent of all igneous events could cause releases of contaminants into an ash cloud. When combined with the probability that these events happen at all, the contribution to the base case repository performance is negligible. Figure 4-44a compares median dose rates. The median dose rates are also much smaller than the peak dose rates for the groundwater base case. Figure 4-44b compares the dose rates over 1 million years. The maximum dose rate from volcanism is approximately two million times less than the peak dose rates from the base case groundwater transport of radionuclides.

The model for the enhanced-source-term scenario examines the interaction underground of magma with waste packages. The magma is predicted to contact between 0 and 170 waste packages, with 40

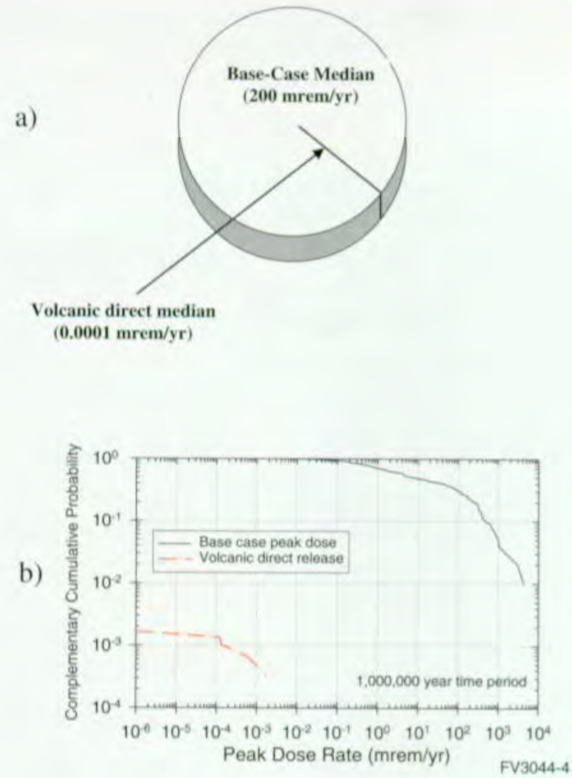


Figure 4-44. Volcanic Eruption Scenario Dose Rates (Top, a) Median dose rates from volcanic eruptions compared with median base-case dose rate. The median dose rates due to releases associated with volcanic eruption are significantly smaller than those modeled in the base case. (Bottom, b) CCDF showing combined peak dose rates and probabilities for the volcanic eruption scenario compared with the base case.

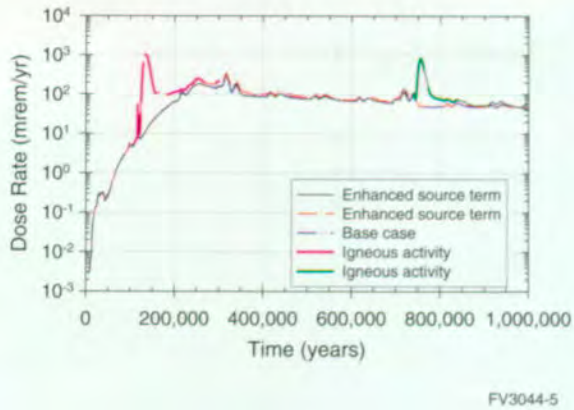
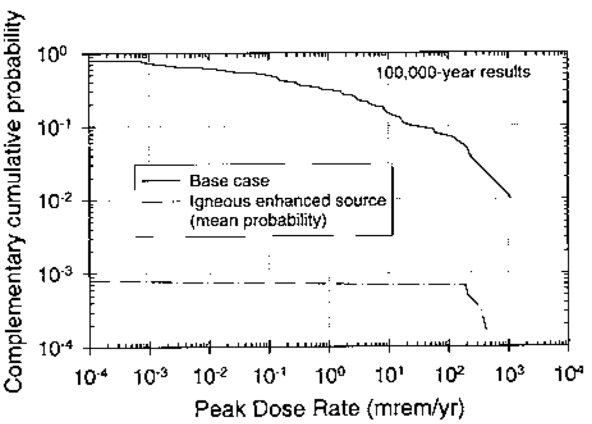


Figure 4-45. Dose-Rate Time Histories from Igneous Enhanced Source Term Scenario Compared with Base Case Dose Rate
Two examples of dike intrusion causing an enhanced source term release to groundwater are plotted, and result in an increased dose rate that occurs tens of thousands of years after the intrusion.

to 50 waste packages being the average. The sudden release of radionuclides in this scenario produces an increased source term for the base case groundwater flow and transport calculations. All other flow and transport modeling parameters are specified the same as for the base case. Two examples of dikes interacting with waste packages are shown in Figure 4-45. One event occurs at about 113,000 years and has a peak dose rate of 4 times the peak dose rate of the base case example shown in Figure 4-45 (that occurs at about 350,000 years).

The other event occurs at about 735,000 years and has a peak dose rate of 3 times the example base case. In both cases, the peak dose rate at the dose-exposure location is tens of thousands of years after the event because of the time needed for groundwater to transport the radionuclides. For all the modeled cases, the increase in dose rates range from less than 2 times to about 8 times the base case, although volcanic events that happen before about 100,000 years produce dose rates many times greater than the base case dose rate for the same time period. As Figure 4-45 shows, after the waste released by interaction with magma has been transported away, the dose histories again track the base case (this can take about 100,000 years). Figure 4-46 shows CCDFs for peak dose rates for 100,000 years (Figure 4-46a) and for 1 million years (Figure 4-46b). In both cases, the CCDFs are significantly less than the corresponding base case CCDFs. The peak dose rates are less than half those of the corresponding base case rates, and the probabilities are approximately one-tenth those of the base case dose rates.

Modeling of the indirect effects of igneous events has shown that recently formed geologic structures (faults or dikes) upstream from the repository have no effect on contaminant transport. In order to calibrate the flow model it was necessary to make the hydraulic conductivity of existing structures very low (Wilson et al. 1994, Table 11-1). Thus, making them even more of a barrier through



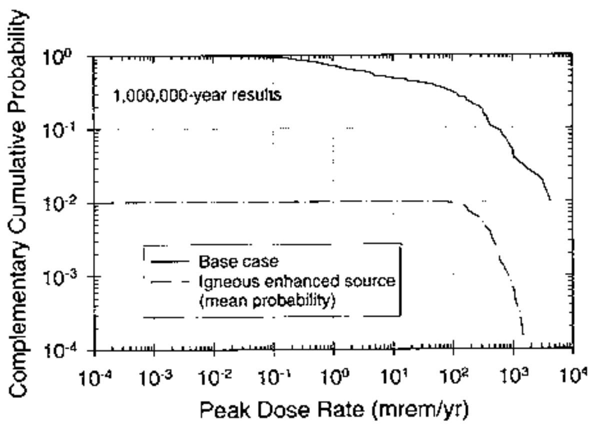


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Figure 4-46. Peak Dose Rate Complimentary Cumulative Distribution Function for 100,000 and 1 Million Years

Peak dose rate CCDFs for 100,000 years (Top, a) and 1 million years (Bottom, b). Probabilities and dose rates for both time periods are considerably less than the corresponding base case dose rates. (CCDF—complimentary cumulative distribution function)

igneous intrusion causes no noticeable change in the flow patterns. When the down-gradient dikes are modeled as being more transmissive, there is a small change in the flow pattern. Generally, the flow is directed more to the east and may be slower, which does not increase the dose rates significantly. Patterns of radionuclide concentration for the base case and for a more transmissive dike located along the Solitario Canyon fault are shown in Figure 4-47.

The probability that any of these igneous-activity scenarios will occur can be estimated. Over 10,000 years, there is less than one chance in 1,000 that there will be any igneous activity at all. If there is activity, there is a very low probability that a volcanic eruption will contain any high-level radioactive waste, because the waste packages still provide enough containment to withstand the forces of the eruption. About 60 percent of the time, an eruption would cause an enhanced source term for groundwater transport of radionuclides (regardless of the time of occurrence). Over one million years there is less than a 10 percent chance that there will be igneous activity. If an igneous event does occur, there is about a 5 percent chance that a volcanic eruption will contain high-level radioactive waste and a 60 to 70 percent chance that there will be more radionuclides in groundwater flow because of the eruption. The details of the igneous-activity analyses are contained in Section 10.4 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i). Table 4-3 summarizes the probabilities of volcanic effects.

4.4.3 Seismic Activity

Seismic activity has not previously been systematically included in TSPA analyses, although ancillary calculations have been made (Gauthier et al. 1996). Potential effects of seismic activity include the following:

- Vibratory ground motion and displacement from earthquakes (can cause rockfall on waste packages or other disruptions to the packages)
- Changes in site hydrologic properties (such as change in flow patterns of groundwater near the waste packages or a change in elevation of the water table)
- Indirect effects such as alteration of groundwater flow and transport paths from faulting near the repository site

Because these events have not been evaluated in a previous TSPA, their potential impact on a repos-

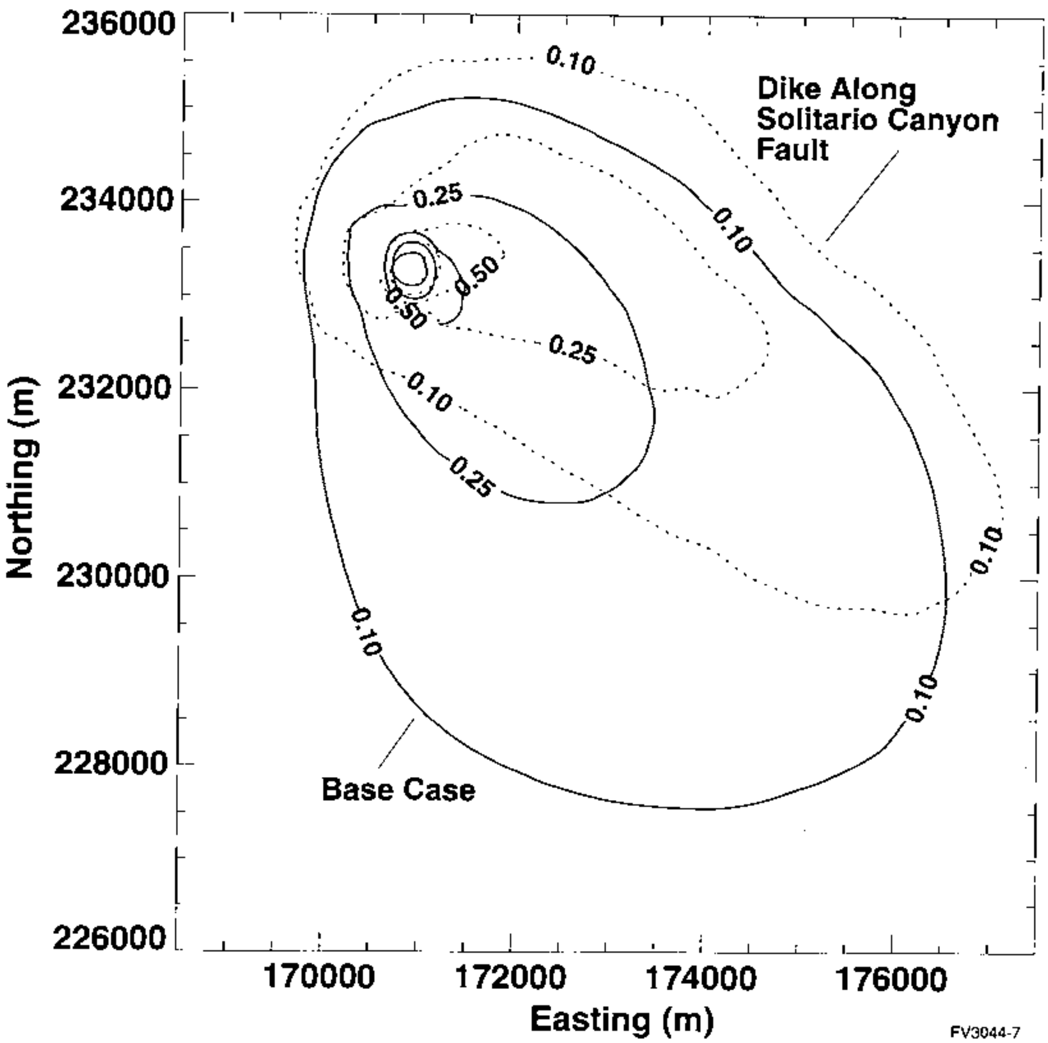


Figure 4-47. Contaminant Concentration Profiles Resulting from a Dike Intruding Along the Solitario Canyon Fault

A transmissive dike along the Solitario Canyon fault area could shift contaminant concentration flow patterns to the northwest (shown by the dotted contours).

itory has not been known until these analyses. The probability of earthquake occurrence is sufficiently high that these scenarios must be considered.

The *Probabilistic Seismic Hazard Analysis* (CRWMS M&O 1998h) led to the determination of fault displacement and vibratory ground motion hazards. The activities performed were as follows:

- Evaluation and characterization of seismic sources including the characterization of potential fault displacement
- Evaluation and characterization of vibratory ground motion, including earthquake source, wave propagation path, and the effects of the country rock
- Probabilistic seismic-hazard analyses for both fault displacement and vibratory ground motion

The hazards results are in the form of annual frequencies that various levels of fault displacement and vibratory ground motion are expected to exceed. The hazard analyses were based on evaluations of the seismic source charac-

Table 4-3. Probabilities of Igneous Impacts on Repository

	At 10,000 years	At 1,000,000 years
Probability of Occurrence	< 0.0001	< 0.1
Probability of Surface Releases	~0	~0.05
Probability of Increased Groundwater Doses	~0.6	0.6 – 0.7

teristics, earthquake ground motions, and fault displacement that reflect interpretations of different scientific hypotheses and models using available data.

The procedures used for evaluating seismic hazards allowed quantitative assessments based on interpretations provided by the experts. (Two expert groups were assembled: a panel of ground motion experts, and panel of seismic-source and fault-displacement experts. The latter panel was divided into teams covering the geology, tectonics, and geophysics of Yucca Mountain.) The quantification incorporates uncertainty in the hazard analyses because of uncertainty in the input interpretations as well as random variability in input parameters. The hazard results are presented as mean, median, and fractile hazard curves representing the total uncertainty in input interpretations.

4.4.3.1 Rockfall Scenario

Falling rock can affect repository performance if damage to the waste packages causes early leakage of radionuclides. Rockfall can either split the waste package, allowing immediate access of air and water, or cause dents in the package, providing locations for accelerated corrosion. Additionally, rockfall can knock waste packages from their supports, enhancing the chances of corrosion on the bottom. If the waste package is prematurely corroded, air and water can reach the waste earlier. Figure 4-48 illustrates this scenario.

Rocks from the drift walls can fall if the stresses on them change. Initially the drifts are lined with concrete, but the concrete is expected to decompose within a few hundred years (the base case assumption and the one used here is that the drifts are not backfilled). As the emplaced waste heats the repository area, the rock stresses change, primarily because of thermal expansion. In some cases, the effect on the rocks surrounding the drifts is a change from compression to tension, potentially causing rockfall.

Seismic activity can also cause rockfall. The magnitude of the sizes of rocks that fall has been correlated with vibratory ground motion. The *Probabilistic Seismic Hazard Analysis* (CRWMS M&O 1998h, Volume 1, Chapter 7) has provided the probabilities of occurrence of various levels of peak ground velocity. It should be noted that earthquake ground motion is greatly reduced underground. An earthquake that can collapse buildings on the surface has much less effect underground. Over any period (such as 10,000 years), there are expected to be numerous small earthquakes (with small ground velocity), but there is also a small probability that there can be large earthquakes. The *Probabilistic Seismic Hazard Analysis* (CRWMS M&O 1998h) provides estimates of these probabilities.

For waste packages to be damaged by falling rock, the following must occur:

- A sufficiently big rock is available
- The waste package wall is so thin that the rock can damage it
- The rock hits a package

The distribution of rocks available to fall has been estimated from a study of fracture spacing in the Exploratory Studies Facility. Most rocks are less than 1,000 kg (1 ton), with many weighing near 50 kg (100 lb). There are very few rocks greater than 3,500 kg (4 tons). The distance the rocks fall is approximately from the drift ceiling to the waste

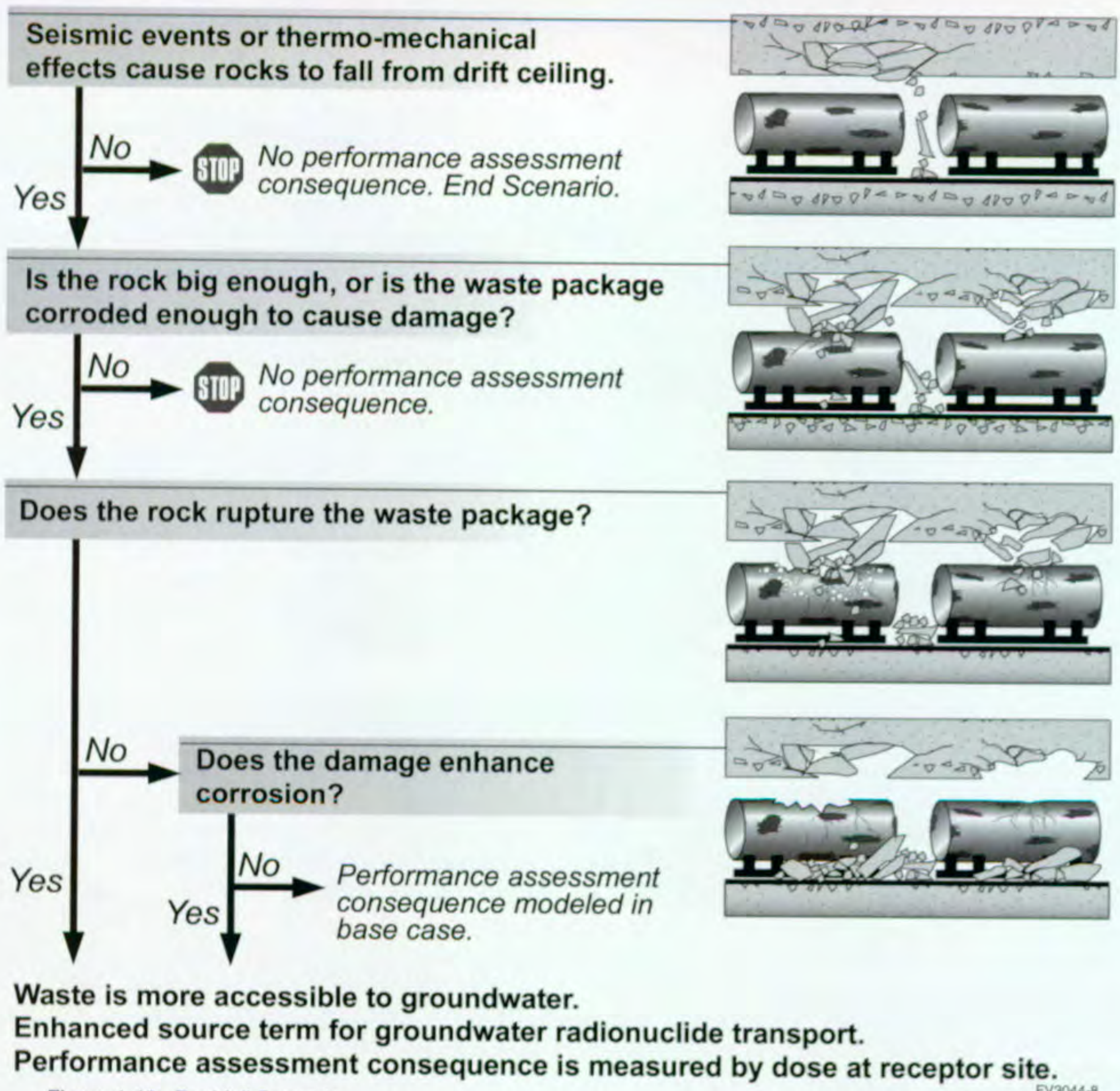


Figure 4-48. Rockfall Scenario
Rockfall could damage packages causing acceleration of corrosion, and if packages are already damaged by corrosion, even faster exposure of waste could occur.

package—approximately 3.5 m (12 ft). When the waste package outer barrier is thick, the rock mass needed to damage it is very large (larger than 3,500 kg (4 tons)). As the waste package walls corrode, the rock mass that can damage it is correspondingly less (as small as 50 kg (100 lb) for severely corroded packages). Therefore, at later times (more than 100,000 years) rockfall damage is more likely because the waste packages have corroded and there is a much longer period of time in which a large earthquake can occur. Lastly, the waste packages occupy about 40 percent of the space along a drift. For the size rocks considered here, any that fall have an approximately 40 percent chance of hitting a waste package (or a 60 percent chance of falling to the drift floor without hitting a waste package).

4.4.3.2 Indirect Seismic Effects Scenario

Seismic activity near Yucca Mountain could affect repository performance even if a fault does not intersect the repository. This scenario is shown in Figure 4-49. As discussed in Section 3.7, flow and transport through the saturated zone is controlled by the geologic structures and their hydraulic properties. If faulting occurs in the saturated zone, it could potentially change water flow patterns and, therefore, repository performance. The most likely places for faulting to occur are at locations of previous igneous or seismic activity. These locations include faults near the repository footprint (for example, Solitario Canyon and Drill Hole Wash) or faults in the Yucca Mountain region (for example, Stagecoach Road). A fault can change the flow pattern by either being a barrier to flow or a conduit for flow. Redirection of flow could increase the dose rate at the receptor location. However, recently observed seismic activity (the Little Skull Mountain earthquake) does not significantly affect flow. Over sufficiently long times, several earthquakes could cause substantial movement along existing faults.

4.4.3.3 Results of Seismic Activity Analyses

Based on modeling results, most waste package failure because of ground motion occurs when the waste package outer wall is completely corroded away. Figure 4-50 shows a distribution of waste package failure times because of rockfall. Analyses of the rock sizes that are required to break open a waste package show that, if the outer barrier is not corroded, a rock larger than any observed in the Exploratory Studies Facility is needed to damage it. When the outer barrier and about half the thickness of the inner barrier are gone, a rock similar to the average-sized rock in the Exploratory Studies Facility can damage a waste package. However, this type of failure requires more than 100,000 years under wet-corrosion conditions. The analysis combined failures from rockfall with those from normal corrosion. Waste package damage from corrosion and rockfall has been used in dose calculations similar to those for base case groundwater flow. The calculations show that there is almost no effect on repository performance over 1 million years. The model for waste form release after the waste package is breached assumes that just one hole is enough to permit substantial amounts of waste to be released (see Section 3.4). The rockfall analysis predicts that much larger (and more numerous) holes occur than for corrosion. However, this does not change the amount of waste released under the existing waste package model.

The results of the indirect-effects modeling are very similar to the modeling of the indirect effects of dikes (see Section 4.4.2.4). There is very little change in the flow pattern when a permeable fault is introduced to the region and no detrimental effect at all if the fault is a flow barrier. There is no difference between this concentration profile and the one for the base case. Modeling has shown that the concentration profiles change only if several faults occur.

Over 10,000 years, the probability of rockfall causing a waste package to split open is essentially zero because the waste package walls are thick enough to withstand hits from most falling rocks. There is less than a 1 percent probability that rockfall will accelerate corrosion during this time period. Over 1 million years, about 30 percent of the waste packages in the repository could be breached by rockfall. When these failures are added to the normal failures from corrosion, they do not significantly change the overall probability of failure because they mostly occur at very late times (more than 500,000 years after emplacement). Section 10.5 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) contains a detailed description of the calculations.

4.4.4 Nuclear Criticality

Isolated nuclear criticality events could occur if the engineered control measures in the waste package fail and other conditions occur (such as the presence of water). Additionally fissile material in the waste can potentially form a critical configuration in the surrounding rock. Criticality has not previously been included in TSPA analyses, but the DOE waste package design team has extensively investigated the possibility (CRWMS M&O 1996b; CRWMS M&O 1997r; CRWMS M&O 1998f). Although criticality is very unlikely because multiple failures in designs and materials must occur, it cannot be ruled out during the period

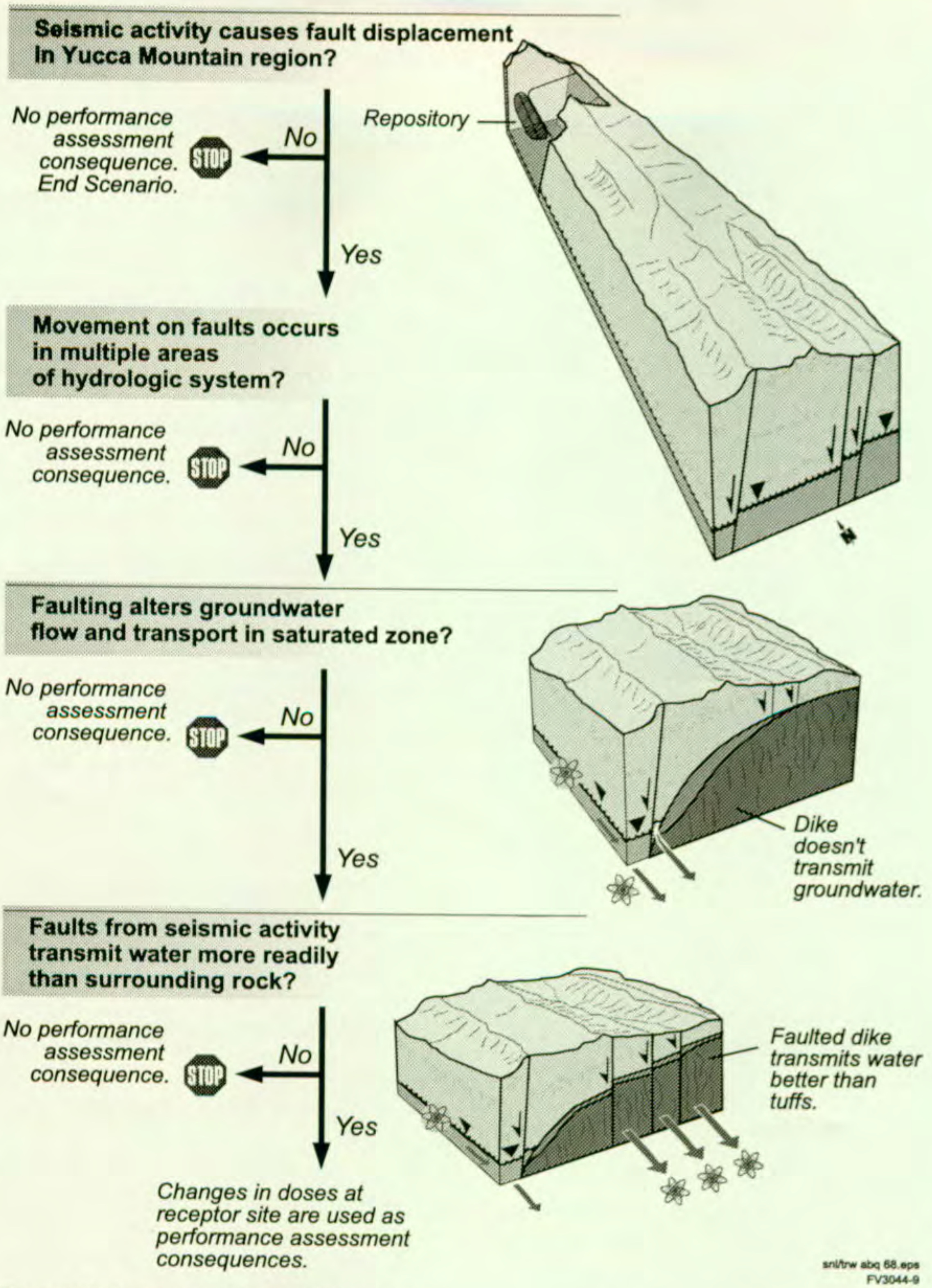


Figure 4-49. Indirect Seismic Effects Scenario
Could seismic activity in the repository area, not directly impacting the site, have an effect on groundwater flow? If groundwater flow patterns were affected in a manner that enhanced flow to the receptor site, there could be a change in dose rate.

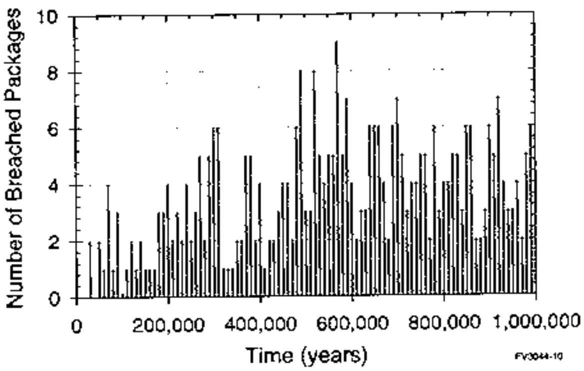


Figure 4-50. Number of Waste Package Failures from Rockfall as a Function of Time
Based on the size of rocks observed in the Exploratory Studies Facility, the packages would have to have lost the outer barrier and half of the inner barrier thickness in order to be significantly damaged by rockfall.

of repository performance. Criticality can occur in two locations in a repository system, inside a waste package or in the surrounding rock. The following three factors are required for nuclear criticality to occur in the Yucca Mountain repository environment (Kastenberg et al. 1996, Chapter 1):

- A sufficient quantity of fissile fuel (uranium or plutonium)
- A moderator (e.g., water) of the fission neutrons
- Insufficient neutron absorbers (e.g., boron) for the amount of fissile material present

In addition, the geometrical configuration of the fuel and moderator influences the likelihood of a criticality.

4.4.4.1 In-Package Criticality Scenarios

There is generally sufficient fissile fuel inside a waste package to form a critical configuration; however, the waste package is designed to provide sufficient neutron absorbers to prevent criticality. Furthermore, waste packages may be loaded in such a manner that in-package criticality, even in the worst degraded configuration, is an incredible event (CRWMS M&O 1997r, p. 75). If water can enter the waste package, a mechanism then exists to corrode the steel containing the neutron absorbers and remove any soluble neutron absorbers. If, in addition, the water fills most of the waste package, a moderator is present to support a criticality. Scenarios for in-package criticality investigate processes for introducing water to a waste package and dissolving out the neutron absorbers (CRWMS M&O 1997b). The processes differ slightly depending on the type of fissile material in the waste package, but the general scenario is the same. Figure 4-51 illustrates this general scenario. Criticality analyses for both commercial spent nuclear fuel (CRWMS M&O 1997r) and a bounding case for DOE-owned aluminum-clad spent nuclear fuel (CRWMS M&O 1997u) have been completed.

If a hole develops in the top of the waste package but not in the bottom, a "bathtub" that can hold water is formed in the waste package. For commercial spent nuclear fuel, the primary neutron absorber is boron carried in stainless steel plates. Certain of the assemblies will be so reactive as to require additional neutron absorber, less soluble than boron. For such a limited fraction of the waste packages, gadolinium, hafnium, or boron carbide could be suitable replacements for, or supplement to, the boron. As the steel rusts away, the boron can dissolve in the water and be flushed from the waste package or it can settle to the bottom of the package. Each spent nuclear fuel assembly is in the proper configuration to support criticality because it was in that configuration when it was in a nuclear reactor. As the waste form degrades, the fuel assemblies may shift. This shift may make the configuration more or less reactive.

The onset of a nuclear criticality is accompanied by an inereased heat output. This heat will cause accelerated evaporation or boiling of the water moderator; the resulting loss of moderator will slow down, or stop the criticality[46]. A steady-state criticality can be maintained at a power level such that the rate of water evaporation, or boiling, will just equal the rate of water flowing into the waste package. Eventually, even the steady-state reactor will shut down because of reduction in the flow of

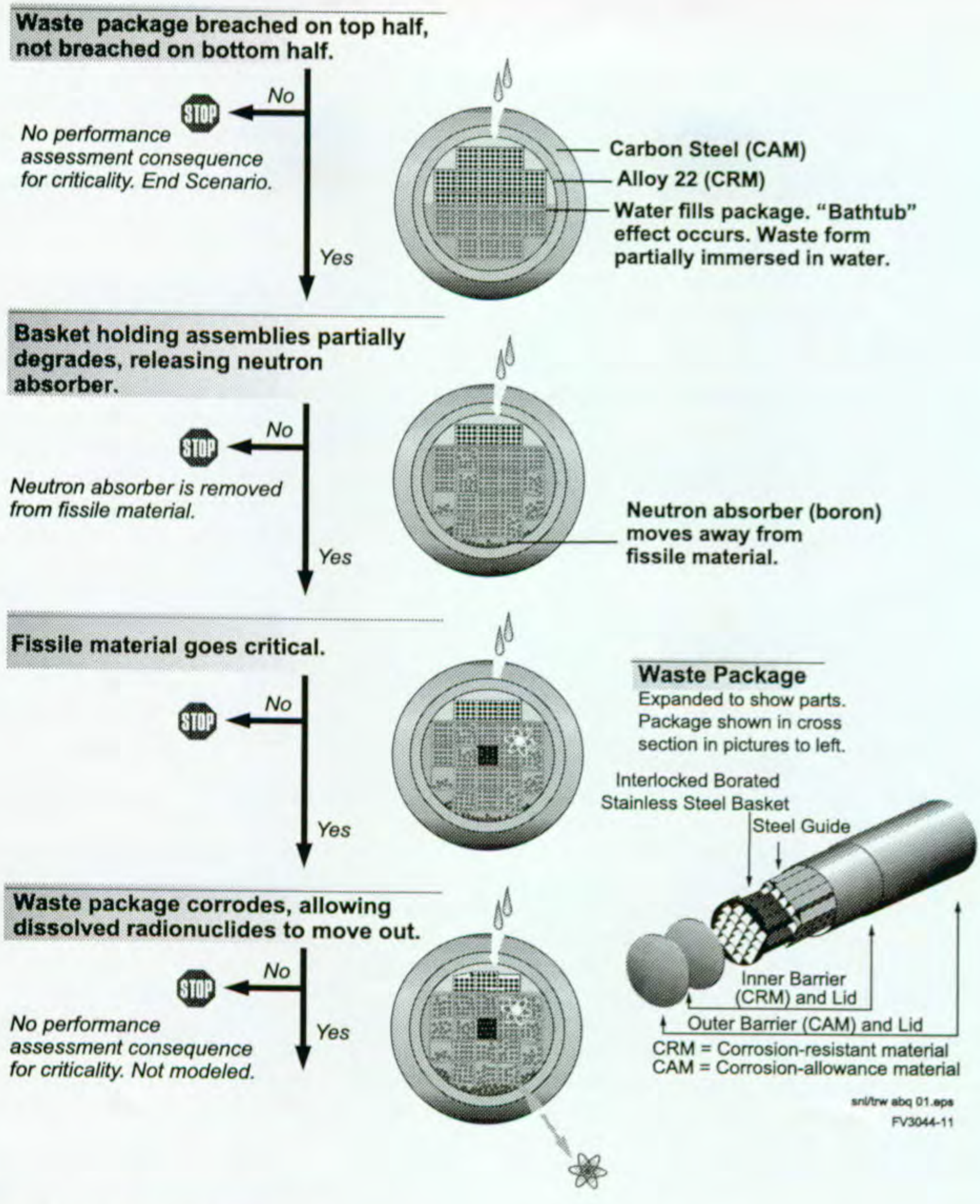


Figure 4-51. In-Package Criticality
Criticality is very unlikely, but a series of sequential steps that could lead to in-package criticality are shown. The result could be additional radionuclides available for transport; however, many would be short-lived and could decay away before reaching a receptor.

water into the waste package, or a loss of standing water due to failure of the package bottom.

Criticality increases the radionuclide inventory (in the waste package where the event occurred) that could be transported by groundwater to the dose-exposure location. The additional fission products initially have high activity but many are relatively short-lived. They can potentially cause increased doses at the dose-exposure location if both short-lived and long-lived radionuclides are transported there. The transport process for these radionuclides is expected to be the same as for the base case, thus the impact of any criticality on performance assessment can be judged by the change in source term. Because transport by groundwater ranges from several hundred to several thousand years (see Figure 3-73), the additional short-lived radionuclides that are produced by the criticality will mostly decay away before they reach the dose receptor point.

4.4.4.2 Out-of-Package Criticality Scenarios

As waste is transported from the waste package, it could accumulate in the surrounding rock. Transport as either solutes or colloids is possible. Mechanisms for accumulation include physical or topographic features (such as accumulation of contaminated water in geologic strata or ponding of water in low spots in the drifts). Flooding of drifts is considered quite unlikely because the calculated seepage rate is very low and the permeability of the rock is high. Other scenarios will be investigated for the LA, including those considering potential criticalities in the repository drifts. When the waste form dissolves, its chemical oxidation state usually increases. Accumulation mechanisms can occur if the contaminated water encounters reducing materials that cause the dissolved fissile materials to precipitate. Organic materials such as buried carboniferous logs are a possible source of reducing agents. The transport process separates the fissile materials and the neutron absorbers that may have left the waste package together. For example, transport of fissile-material colloids may separate them from the neutron-absorbing materials. If the fissile material forms a spherical shape in the rock, criticality is more likely to happen than if the material is in a flat plane. Fission products created by the criticality could be transported to the dose receptor point. Figure 4-52 illustrates some external criticality scenarios.

4.4.4.3 Results of Nuclear Criticality Analyses

The consequences of criticality have been evaluated by the waste package design team and found to be relatively small compared with the measures for nominal repository performance. An analysis of an in-package criticality scenario for commercial spent nuclear fuel has been completed, using conditions and waste characteristics that potentially maximize the effects of the criticality (CRWMS M&O 1997r, Section 6.2.2). Section 10.6 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) contains details of these analyses.

The examples analyzed had criticality durations ranging from 1,000 to 10,000 years (the latter being the longest period that could be supported by a necessary, moist climate cycle). To be conservative, the example criticalities are assumed to start 15,000 years after emplacement, when commercial spent nuclear fuel is the most reactive (CRWMS M&O 1997r, Figures 6.2.2-1 and 6.2.2-2). For further conservatism, a criticality design-basis fuel is used (selected to be more reactive, with respect to criticality, than 98 percent of the commercial pressurized water reactor spent nuclear fuel). Figure 4-53 shows the time histories of the radioactivity for a waste package containing 21 pressurized water reactor design-basis fuel assemblies that undergo a criticality event of 1,000, 5,000 or 10,000 years. Also shown in the figure is

[46] The natural reactor occurring at Oklo, Gabon, depended on unique conditions in the surrounding rock that resulted from sufficient water to moderate the fission reaction. When the water was unavailable, the reactions stopped (Oversby 1996, pp. 37-38; Naudet 1975, pp. 589-601).

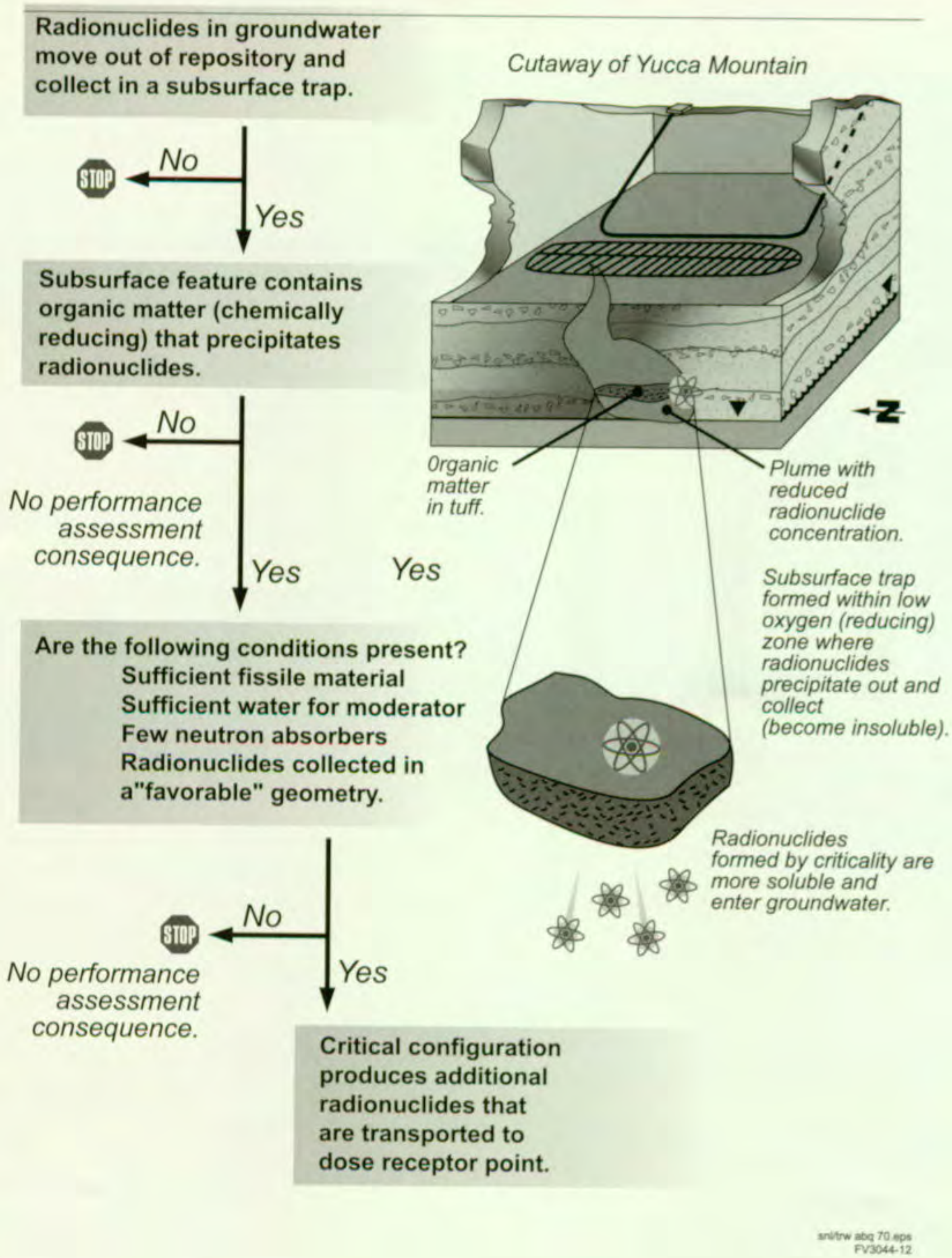
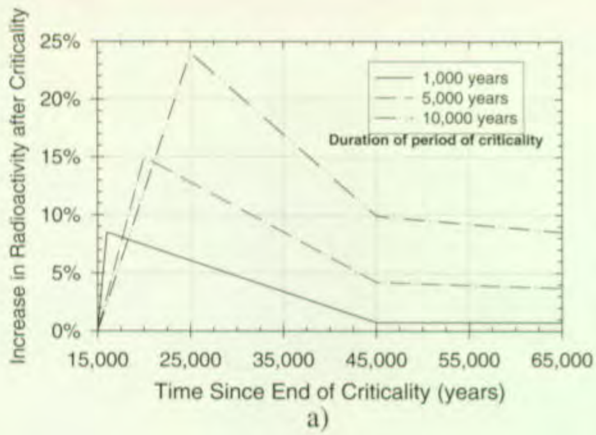


Figure 4-52. External Criticality Scenarios
Criticality in the rock away from the repository is very unlikely, but a series of processes that could lead to out-of package criticality are shown. The result could be additional radionuclides available for transport to a dose receptor.



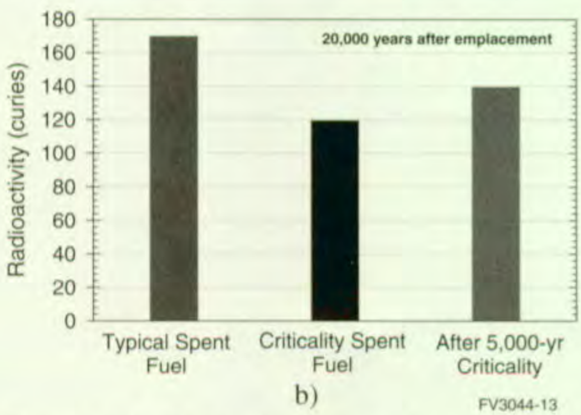


Figure 4-53. Radioactivity Levels for In-Package Criticality Model
(Top, a) Criticalities lasting for three different durations are shown: 1,000 years; 5,000 years; 10,000 years. All start at 15,000 years after emplacement. (Bottom, b) A comparison of total radioactivity developed for different fuel types from a 5,000-year criticality. ("Criticality spent fuel" is the waste that is modeled as going critical for 5,000 years.) The extra radioactivity from this criticality is less than the normal radioactivity from most waste.

the total radioactivity for the same package that does not undergo a criticality, and the radioactivity for a package containing average pressurized water reactor assemblies. It is seen that the design basis fuel has about 24 percent more radioactivity immediately after the criticality shuts down (at 25,000 years) than does the fuel not undergoing a criticality. It should, however, be noted that both these radioactivity curves are below that of the average pressurized water reactor spent-fuel package that does not undergo any criticality. The additional heat output from a steady-state criticality is only about 2 kW per package, which is inconsequential compared with the overall repository heat load. Because of the small increases in radioactivity and heat output, there is no chance that the waste package and engineered barrier system will be mechanically disrupted by a criticality.

Criticalities outside the waste package primarily require some mechanism to accumulate the fissile material in a localized area. To accumulate sufficient fissile material for a criticality, the mechanism must be fairly effective, since the concentration of dissolved uranium and plutonium in groundwater is very low (about 1 part in 100,000 or less, as is discussed in Section 3.3). An analysis of the potential for external criticality from plutonium has been completed (CRWMS M&O 1998f, Section 6). The waste packages contain sufficient neutron absorbers to prevent criticality inside the container. However, at high pH, uranium and plutonium are relatively soluble, while the neutron absorbers are not. These conditions provide the opportunity for the fissile materials to move away from the neutron absorbers and concentrate in the drift or the surrounding rock. Concentration mechanisms include reaction with the tuff or sorption on zeolites. Using the EQ6 reaction-path computer code (Wolery 1992a), precipitation and sorption of plutonium has been estimated. For both processes, the concentration of plutonium is less than 0.01 percent by volume, which is much too low to make criticality possible.

Another mechanism for external accumulation is the analog of the uranium ore deposits found in nature. The conditions for epigenetic (accumulating later than the surrounding host rock) formation of ore bodies are almost totally absent from the Yucca Mountain area (CRWMS M&O 1998f, Section 6.2). Uranium or plutonium precipitation occurs if the dissolved materials encounter a reducing environment. Potential reducing environments include hydrothermal fluids (can be associated with igneous activity, although these conditions have not been observed in the Yucca Mountain region), decayed organic material, or petroleum deposits. None of these features are located in the Yucca Mountain area, nor are they anticipated to occur.

In the extremely unlikely event that an external criticality occurs (with uranium or plutonium), the resulting radionuclide inventory increase is very small. For example, (CRWMS M&O 1998f, Section 9) a plutonium-fueled criticality operating at a power level of 590 watts was simulated for 4,000 years. Of the total radioactivity of 560 curies immediately after shutdown, 480 curies come from radioactive decay of the plutonium already present in the waste form and the balance (80 curies) from the result of the criticality (fission products and transuranic elements), so the criticality contributes only a minor amount to the radioactivity.

It has been suggested (Bowman and Venneri 1996) that a transient external criticality with a large mass of highly enriched fissile material could have sufficient positive reactivity feedback (due to overmoderation, a condition that can increase k_{eff} as water is removed from the system), and could be sufficiently confined by the rock, to produce a power pulse large enough to be classified as an explosion (Bowman and Venneri 1996, p. 280). The following are reasons why such a positive feedback transient event is far less likely than the already incredible external criticality:

- It will be nearly impossible to accumulate a large enough mass because the few waste packages containing highly enriched uranium (or plutonium) will be widely interspersed among the much greater number of packages containing low enriched material. As multiple waste packages fail near the highly enriched material, the effective enrichment would be reduced (Van Konynenburg 1996, p. 306).

- The slow accumulation of such a large mass of fissile material (over a million years) is entirely inconsistent with the rapid reactivity insertion necessary for a transient criticality. With slow accumulation, it would be nearly impossible to accumulate an overmoderated mass without going through a small criticality that would disrupt the mass and consequently preclude the possibility of reaching the overmoderated condition (Van Konynenburg 1996, pp. 316 ff.).

The following is a summary of the physical and chemical processes that act to make repository criticality an unlikely event (either inside or outside the waste package):

- Most of the waste packages cannot go critical because they do not have sufficient fissile material (or have sufficient enrichment) to form a critical mass.

- Of the waste packages that have sufficient fissile material for criticality, commercial spent nuclear fuel waste packages are a major fraction. The maximum fraction of commercial packages with sufficient fissile material for criticality never exceeds eight percent of the total. This amount drops by 30 percent for times beyond 40,000 years (CRWMS M&O 1997r, p. 46). This significantly decreases the criticality probability because only 10 percent of the waste packages are expected to even be breached by 40,000 years (see Figure 4-7).

- Of the waste packages that have sufficient fissile material for criticality once they are breached and degraded, most require some form of water retention to provide moderation. Models of the waste package corrosion process indicate that only 25 percent of the waste packages that are breached will hold a significant amount of water for the time required to flush the neutron absorbers that have been added for criticality prevention.

- External criticalities are extremely unlikely because of the low probability of accumulating significant amounts of fissile material by any credible geologic process. Specifically, there have been no identified deposits of carboniferous logs or other carbonaceous material near Yucca Mountain to cause accumulation to occur.

4.4.5 Human Intrusion

Human intrusion is generally interpreted to mean inadvertent penetration of the repository (such as by drilling operations) that either releases radionuclides at the surface or accelerates radionuclide transport to the dose-exposure location. Regulations specifically exclude consideration of deliberate intrusion of the repository. Human-intrusion drilling scenarios have been investigated in prior TSPA analyses and found to contribute less than 1 percent to the overall repository performance (Barnard et al. 1992, Chapter 6; Wilson et al. 1994, Chapter 16). Because of the uncertainties associated with evaluating future human technology and societies, the National Academy of Sciences has recommended that the probability of occurrence of human intrusion not be considered in any TSPA analysis (National Research Council 1995, Chapter 4). The consequences of human intrusion, in terms of potential increases in long-term doses to the exposed public, are used to measure the resilience of the repository to such disturbances. In the past, human intrusion requirements have been incorporated in NRC and EPA regulations.

4.4.5.1 Technical Bases for Human Intrusion Analyses

The stylized analysis of human intrusion assumes that, as a result of drilling into the repository, a waste package is penetrated by a drillhole. Waste then falls down the drillhole to the saturated zone beneath the repository. There, the flowing water dissolves the waste and carries it to the dose-exposure location. This analysis is meant to show whether a single drilling incident of this type can cause large doses.

The analysis does not consider the process that causes punctures to the waste package or how the waste gets down the drillhole to the water table. Instead, we make some assumptions about the amount of waste that can fall down the hole based on current drilling practices. The analysis assumes that the drilling is for water, an operation in which a 21-cm (8-in.) drill bit is typically used. The drilling operation is assumed to stop when the driller hits water at the water table. At this point, the hole is abandoned; it is neither cased (lined with pipe) nor plugged. Shortly thereafter it fills with rock debris. Consequently, most of the waste that reaches the saturated zone comes from the drilling incident; not from being washed down the hole later.

The waste that falls down the drillhole is assumed to be extensively pulverized. The water dissolves the waste according to the dissolution rate appropriate for saturated zone conditions and the surface area of the specific waste type. The dissolved waste is transported to the dose-exposure location, where it adds to the dose from the base case radionuclide transport. Figure 4-54 shows this scenario.

4.4.5.2 Results of Human Intrusion Analyses

Between 550 and 2,700 kg (1,200 and 6,000 lb) of waste are assumed to fall down the borehole (the amount depends on how much is removed by the drill). Because it is so finely pulverized, the waste can readily dissolve. Two dissolution rates for spent-fuel waste are used, representing the lower and upper ranges expected for saturated zone parameters that control dissolution. Some of the radionuclides included in the analysis are completely dissolved in about 20 years, while the low-solubility ones (like plutonium) continue to dissolve in groundwater for many thousands of years. The drilling incident is modeled as occurring at 10,000 years (the time at which the waste package is likely to be degraded enough that a drill can penetrate it). Figure 4-55 shows the dose-rate time histories for the extreme cases modeled (about 550 kg, or 1200 lb per low dissolution rate and 2700 kg or 6000 lb. per high dissolution rate). The inset graph in Figure 4-55 shows the dose rates from 10,000 to 20,000 years. The peak for the large-mass/high dissolution rate case in this period is approximately 145 times the base case dose rate at 12,000 years. The dose rate for the low mass/low dissolution rate case is approximately 3.7 times that of the base case at the same time. By about 15,000 years the dose rates again closely track the base case. At about 50,000 years the dose rates diverge again, as can be seen in Figure 4-55. After about 150,000 years the dose rates for all human-intrusion cases again track the base case time history.

In terms of the dose to a critical group over 100,000 years, the effects of human intrusion are

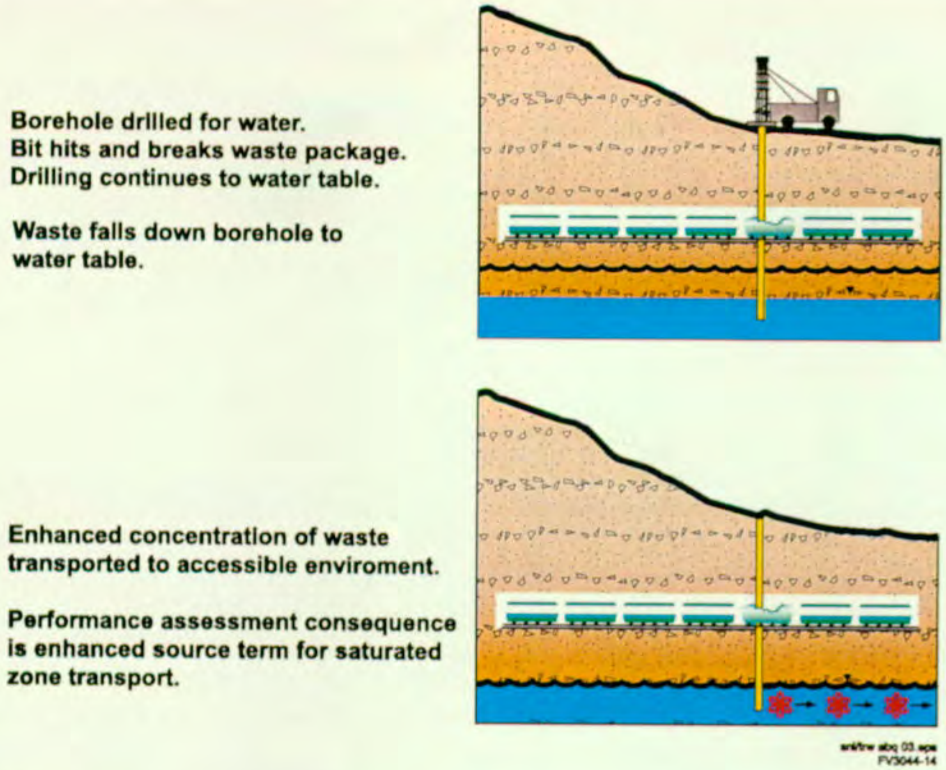


Figure 4-54. Human Intrusion Scenario
This figure is a stylized depiction of what might happen if a waste package were penetrated by a drill bit during drilling for water with drilling continuing to the water table. The assumption is that waste could be carried down the drillhole to the water table. This could add to the dose rate at the receptor.

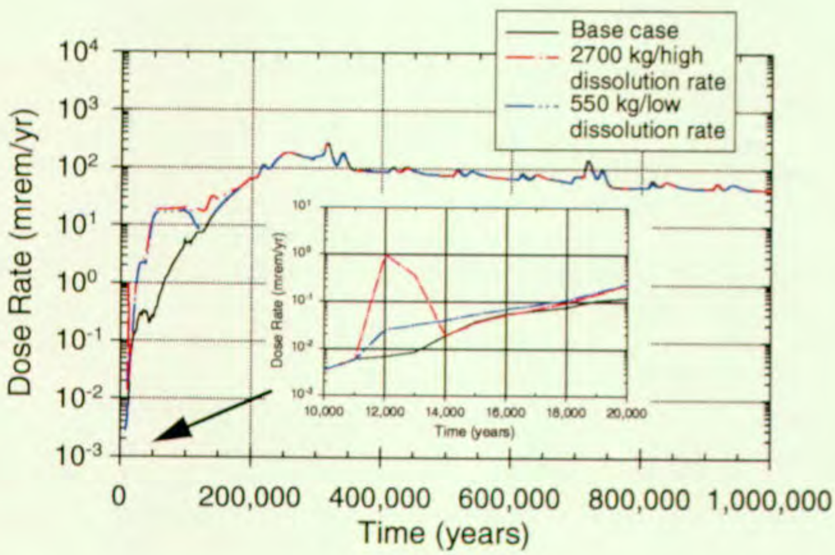


Figure 4-55. Comparison of Human-Intrusion Dose Rates with Base Case Dose Rates
This figure illustrates the case where the extremes of the amount of waste goes downhole and are dissolved in groundwater at the two different rates. The inset graph shows that the fastest moving radionuclides are transported to the accessible environment in about 2,000 years with the next radionuclides arriving in about 12,000 years.

small (an approximate four-times increase over the base case dose rate for about 50,000 years). Over 1 million years, this increase over the base case is unlikely to be significant. At times closer to the human-intrusion incident, the increased dose rates can be much larger than the corresponding base case rates. Section 10.7 of the *Total System Performance Assessment - Viability Assessment (TSPA-VA) Analyses Technical Basis Document* CRWMS M&O (1998i) contains details of the human-intrusion analyses.

4.4.6 Summary

The four scenarios modeled here have produced a wide range of performance impacts on the base case performance of the repository. Results from modeling direct volcanic effects show almost no impact on performance from either consequence or probability. The increased dose rates when a dike creates an enhanced source term can be larger than some base case analyses, but their probability of occurrence is less than 10 percent. The peak dose from volcanism occurs thousand years after the event because of the time required for transport by groundwater to the dose receptor point. Because the volcanic events can happen at any time, the peak dose rate can be much sooner than the peak dose rate for the base case. The models for rockfall do not predict a significant impact on performance. In-package nuclear criticalities have the potential to increase the radioactivity in a small number of emplaced waste packages by up to 25 percent, but the probability of this type of criticality is low, as is the probability of criticality outside the waste package after degradation. Human intrusion produces increased dose rates that are within the range of variability of base case results. In summary, the overall impact of these disturbances does not significantly change our assessment of base case performance.

One area of uncertainty that is apparent from these analyses is the behavior of waste package materials in magmatic environments. If DOE decides to analyze disruptive events in the future, a better understanding of the corrosion and deformation resistance of the waste package inner barrier material can help reduce these uncertainties.

4.5 EFFECTS OF DESIGN OPTIONS

Three enhancements to provide potential improved performance of the engineered barrier system have been identified as "design options" in Volume 2, Section 5.3. These enhancements are as follows:

- Emplaced drift backfill
- Drip shields
- Ceramic coating of the disposal container with backfill

These options are intended to reduce the amount of liquid water in contact with the waste package, which is the principle factor impacting waste package lifetimes. Backfill may cause a reduction in relative humidity for longer periods after waste emplacement, thus causing further delay in the initiation of corrosion. Backfill also may be considered to control the chemistry of the system, thus controlling waste package corrosion, waste from dissolution, engineered barrier system transport and to some extent geologic transport. However, the current analyses do not include such controls. Drip shields and ceramic coating may provide barriers to seepage into the waste package, which can reduce waste form degradation and mobilization. These three enhancements to the engineered barrier system are illustrated schematically in Figure 4-56 and are discussed in detail in Volume 2, Section 5.3. Additional "design alternatives" identified in Volume 2, Section 8.2, are not analyzed for the TSPA-VA, but will be considered as part of the EIS and LA analyses.

4.5.1 Emplacement Drift Backfill

The first enhancement is a design for covering the waste packages with backfill. This design was incorporated into the thermal-hydrologic model and the waste package degradation model to determine the effect on waste package and, ultimately, repository system performance. The backfill was assumed to have the thermal properties of crushed tuff, to be dry at emplacement, and to be introduced 100 years after waste emplacement. The initial moisture in the backfill may cause early corrosion, but this effect is

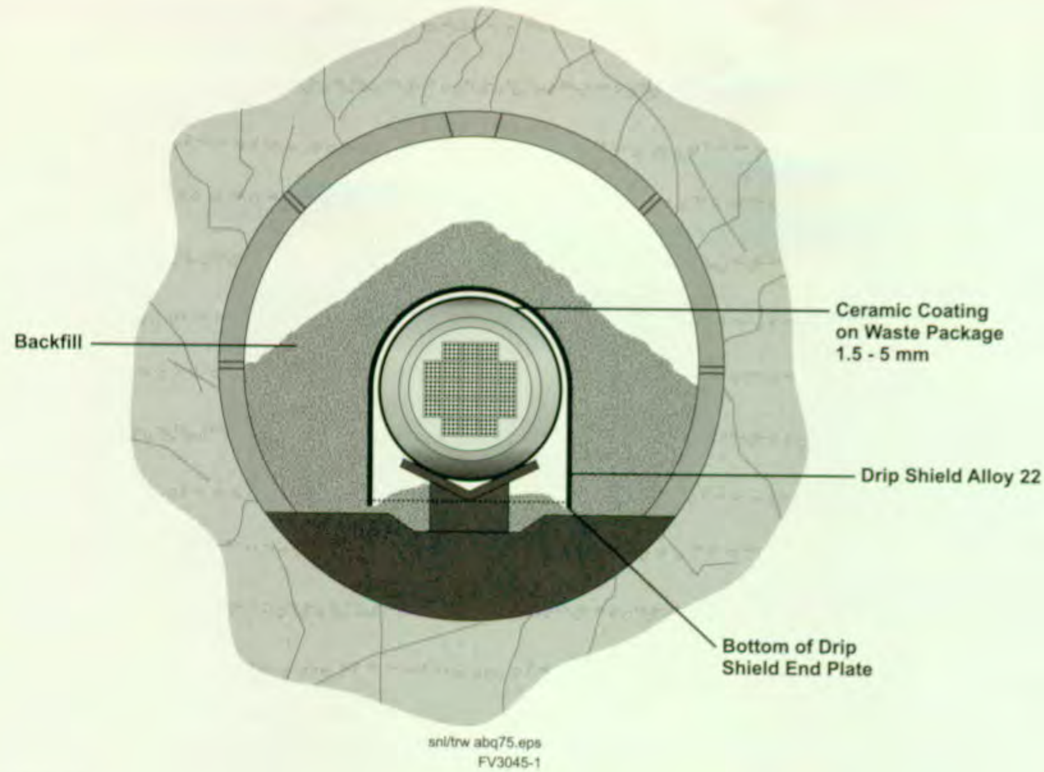
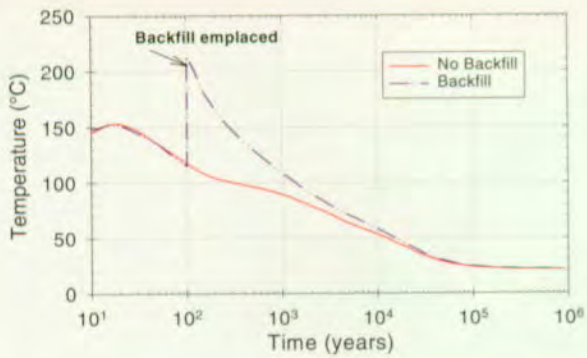


Figure 4-56. Engineered Barrier System with Design Enhancements (Backfill, Drip Shield, and Ceramic Coating)

expected to be small as temperatures in the repository quickly rise to drive off the moisture in the backfill. The primary effect on the system is an increase in temperature at the time of backfilling and a corresponding delay in the relative-humidity increase as the repository cools. While the effect of moisture control is an important potential benefit of backfill, no change in seepage contacting the waste package as a result of backfilling was included in the modeling. The full evaluation of the effects of backfill on system performance have not been completed because the effects of seepage have not been quantified and included in the analysis. For the model assumptions included herein, there is little difference in total system performance between backfill and no backfill cases.

The thermal-hydrologic results and waste package degradation analyses comparing the case with the reference design (no backfill in the emplacement drift) and a partially backfilled emplacement drift are illustrated in Figure 4-57. The large temperature increase at 100 years corresponds to the emplacement time of the backfill. For an emplacement drift without backfill, the dominant mode of heat transfer is radiation, which is an efficient mode of heat transfer between the waste package surface and the drift wall. For a backfilled drift, the resistance to heat transfer from the waste package surface to the drift wall is greatly increased; therefore, the waste package temperature increases. Accompanying the increase in surface temperature is a corresponding decrease in relative humidity around the waste package. These results are presented in Figure 4-57. The degradation of corrosion-allowance material (labeled CAM in the figure) for the backfill case is delayed over 1,000 years from the non-backfill case. The bend in the corrosion-allowance-material degradation curve, after about 80 percent of the waste packages are breached through the outer barrier, is caused by a few of the waste packages that have low relative humidity compared to the rest of the waste packages. This is a result of the thermal-hydrologic variability leading to some zones with low relative humidity. These packages have longer lifetimes for their corrosion-allowance material. The non-backfill case didn't have any such



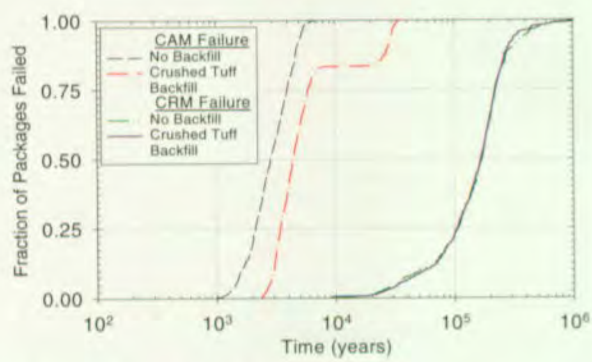


Figure 4-57. Effect of Backfill on Temperature and Fraction of Waste Packages Failed over Time
The top figure shows the average waste package temperature with and without backfill in the NE region (see Figure 3-20) for commercial spent nuclear fuel, long-term average climate, mean infiltration, and base case hydrologic properties. The bottom figure shows the corresponding waste package degradation history. (CAM—corrosion allowance material [outer layer of the waste package], CRM—corrosion resistant material [inner layer of the waste package])

packages. The degradation of corrosion-resistant material (labeled CRM in the figure) is essentially the same for the two cases because the primary factor for degradation of the corrosion-resistant material is whether dripping is occurring (the model assumes the same dripping conditions for both backfill and no backfill conditions). The TSPA results for the two cases are similar because the waste package degradation is similar, so those results are not presented here.

The effect of backfill on seepage is uncertain and additional testing is being conducted to better understand these effects. As noted, the backfill is assumed in these analyses not to alter the seepage onto the waste package. Some of the potential effects of backfill on seepage are as follows:

- Diversion of the seepage around the waste package.
- Reduction of seepage reaching waste package because of evaporation of incoming water that is at a low flow rate.
- Concentration of salts in the backfill from seepage water.
- Promotion of water condensation at the contacts between the backfill and the waste package surface.
- The backfill is not expected to restore the preconstruction unsaturated zone flow conditions. These effects have not been fully evaluated for their impact on overall system performance.

Also, if the drift is backfilled, the effect of rockfall on waste package failure is minimized.

4.5.2 Drip Shields

Performance assessment analyses have been conducted to evaluate the effect of a drip shield on waste package lifetime as well as on overall performance. Several configurations for a drip shield have been identified. However, only a single representative drip shield is evaluated here. The evaluated drip shield design was a 2-cm (0.8-in.) thick sheet made of Alloy 22, the same material as the inner barrier of the waste package. The drip shield was assumed to be placed over the waste package (waste package drip shield) and in the model measured half the surface area of the waste package (the failure of the sides of the drip shield is not important to performance of the waste package). The simulation assumed backfill emplacement at 100 years. The drip shield analyses do not include the potential effects of galvanic coupling of Alloy 22 over the carbon steel, although the design specifies a ceramic coating on the carbon steel that could reduce the effect.

The degradation model for the drip shield assumed that the drip shield upper surface was 100 percent wet in dripping zones. The drip shield was assumed to fail only from general corrosion (no localized corrosion) because there is little potential for crevices on the wetted surface, and there are no significant oxidants (other than oxygen). Failure from rockfall was not considered in these analyses, but is not expected to contribute significantly to drip shield degradation because of backfill protection. Likewise, crevice corrosion between backfill and drip shield is not expected because temperature and oxidant availability would not be sufficient to initiate crevice corrosion of Alloy 22. The drip shield-corrosion rate was assumed to be the same as the benign conditions for Alloy 22 corrosion (moderately acidic pH 3 and moderately oxidizing, see Section 3.4). Although crevices could form between the drip shield and the corrosion-allowance material underneath the drip shield, they were assumed not to form because of the lack of chloride ions required for crevice corrosion (Fontana and Greene 1978, Chapter 3). Potential galvanic coupling between the drip shield and the corrosion-allowance material could reduce the general corrosion rate of the drip shield, but was not considered. The design specifies a ceramic coating on the carbon steel to reduce this potential effect. The waste package was assumed to undergo only humid-air corrosion during the time when the waste package drip shield was functioning. After failure of the waste package drip shield, water was assumed to drip on 10 percent of the waste package surface area because only a small area of the drip shield fails in the model, and only a fraction of the waste package surface area directly under the patch opening in the drip shield will get wet. Very few drip shield patches fail, so 10 percent wetting of waste package surface is conservative. This percentage is in contrast with the base case in that water is assumed to drip on 100 percent of the waste package surface area.

The degradation of the waste package and drip shield in dripping zones was evaluated. The results are presented in Figure 4-58 in a comparison with waste package failure for the base case.

The corrosion of the outer barrier of the waste package (corrosion-allowance material) proceeds at the same rate as the base case even though a drip shield was added. Humid-air corrosion of the outer barrier is unabated by elimination of water dripping onto the waste package. However, the drip shield does cause significant delay in inner-barrier corrosion. The drip shield is expected to be in a more benign environment than some portions of the waste package (because of lack of contact with the corrosion-allowance material), thus its failure is later than the corrosion-resistant material in the case with backfill. Deferring dripping until after drip shield failure significantly extends the lifetime of the waste package. The drip shield enhances the overall waste package lifetime by greater than 100,000 years. The effect on dose from the addition of the drip shield is illustrated in Figure 4-59. Doses are reduced from base case amounts by greater than an order of magnitude for the first 300,000 years of the simulation. However, once a significant number of the waste packages fail after the drip shield fails, the doses approach base case amounts; at about 500,000 years the doses become identical to those from the base case. This analysis indicates that the life span of the drip

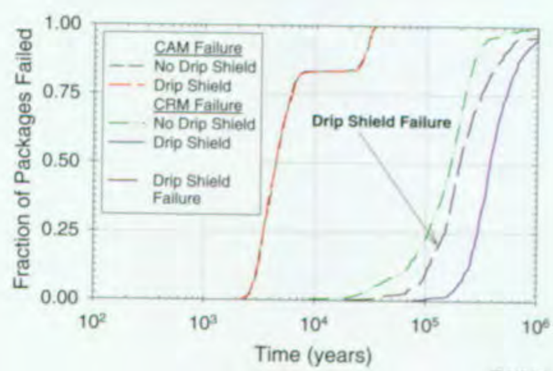


Figure 4-58. Effect of Dripshield on Waste Package Degradation
The drip shield fails after the no-drip shield waste package because of the less aggressive corrosion conditions on the drip shield in comparison with the waste package. This effect leads to significantly later failure of the waste package associated with the drip shield than in the base case.

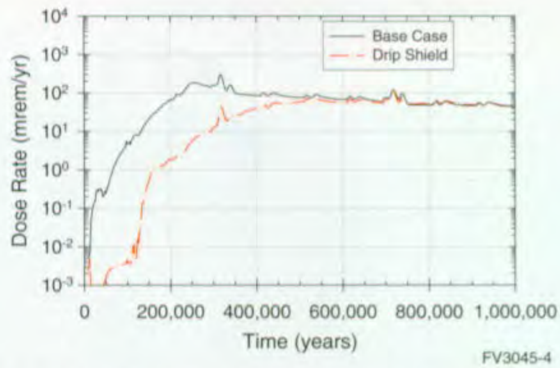


Figure 4-59. Effect of Dripshield on Dose Rate at 20 km (12 miles)
The drip shield substantially improves performance at early times up to 300,000 years because of later waste package failure than in the base case.

shield is the key determinant in providing improved performance. After the drip shield fails, the dose from the repository gradually returns to the dose simulated with non-drip shield conditions.

4.5.3 Ceramic Coating of the Disposal Container with Backfill

A third design enhancement specifies a coating of ceramic material on the waste packages, with the expectation that this coating will delay corrosion of the outer barrier for some period. The design includes backfill on top of the waste packages to prevent damage to the coating by rockfall, so thermal-hydrologic parameters from the backfill case are important in these analyses. The preliminary model for ceramic coating degradation was provided by the repository-design organization. This model was implemented in WAPDEG, a computer program for modeling various designs, waste package materials, and degradation mechanisms (see Section 3.4). The ceramic degradation model is based on a process by which the oxygen transport rate through water-filled pores in the ceramic coating on the carbon steel substrate is retarded. Because the corrosion rate for the carbon steel substrate is proportional to the rate at which oxygen is supplied to the substrate, ceramic coating performance as an oxygen transport barrier is expressed as the substrate-corrosion-rate reduction factor. The factor is expressed with a distribution to represent the uncertainty. The ceramic coating may have very low connected porosity, that may provide additional protection to the waste package (*Outer Barrier Corrosion Model with Ceramic Coating*. Interoffice Correspondence. V. Pasupathi to J. H. Lee, LV.WP.VP.05/98-102, May 24. Transmitted via QAP 3-12 on May 26 by D. Stahl).

Degradation of the ceramic coating was evaluated for two cases. In the first case, degradation of the ceramic coating was evaluated under humid-air conditions that were assumed to be effective for the first 1,000 years of the analysis. The first evaluation was only for 1,000 years and no failures occurred over this period. The relative humidity is low for the first 1,000 years, so the ceramic pores are assumed to not be filled with water. In the second case, degradation of the ceramic coating was evaluated for continual aqueous or dripping conditions. The failure of the ceramic coating for only the second condition is illustrated in Figure 4-60, since the first provides no failure over the period of interest. For humid-air conditions, the ceramic coating is not breached in the period of interest, the first 1,000 years of the simulation, after which aqueous conditions are assumed. For the aqueous conditions, the breach of the ceramic coating does not begin until after 300,000 years. After 1 million years, there is less than a 6 percent failure of the ceramic coating on the waste packages with dripping. For the preliminary ceramic coating model, there are few waste package failures under 1 million years. The ceramic model requires additional data and justification because it is in the early stages of development. The dose rate plot for the ceramic case is also presented in Figure 4-60 in comparison with the base case. The dose for the ceramic case starts much later and is reduced by over an order of magnitude from the base case.

The preliminary ceramic coating model that was used in the current analysis to evaluate the impact of the ceramic coating design option on the long-term waste package performance was developed based on the assumption that stress developed by the volume expansion of the corrosion products of the CAM substrate is the only mechanism leading to the coating failure. The analysis assumed two

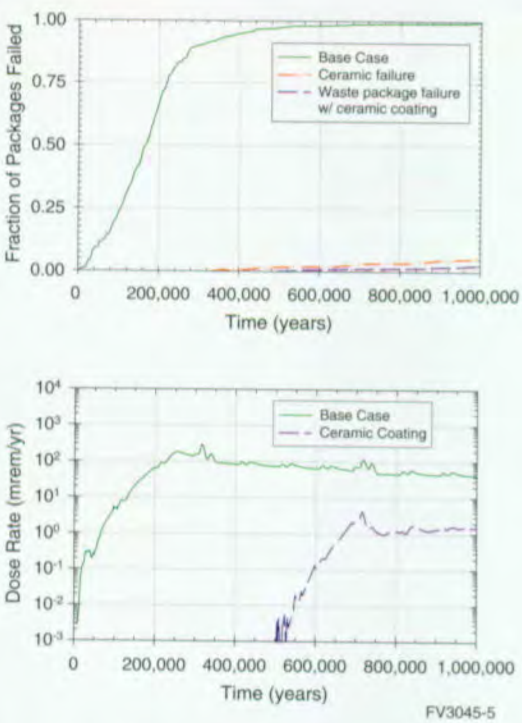


Figure 4-60. Effect of Waste Package Ceramic Coating on Waste Package Degradation

Use of ceramic corrosion factors for humid air and for aqueous conditions leads to a very small number of waste packages with ceramic coating failure. The humid-air corrosion factor was only applied for 1,000 years, so there were no waste packages failing in humid-air conditions. Waste packages affected by aqueous corrosion on the ceramic coating did not begin to fail until after 300,000 years, and only a few percent had failed ceramic coating by 1 million years.

percent interconnected porosity in the coating. As discussed previously, the sensitivity analysis results showed that the ceramic coating itself could last more than 300,000 years if the interconnected pores in the ceramic coating are filled with water, which provides a barrier to the oxygen transport to the corroding carbon steel substrate.

However, in the current analysis, the effect of other potential mechanisms that could cause damage to the proposed thin ceramic coating on the waste package have not been analyzed. Defects and flaws in the coating could be introduced from the fabrication process (application of the coating on the waste package). Differential thermal expansion between the ceramic-coating material and the waste package metal-barrier substrate could potentially cause cracks in the coating. Because of the extremely long time periods considered in the geologic repository, dissolution of the ceramic materials over such long-term periods (especially under locally perturbed acidic or alkaline conditions on the waste package) and its effect on the integrity of the coating need to be considered. In addition, damage to the coating could be caused by improper handling and transport of the ceramic-coated waste package), impact by a sufficiently large rockfall, and ground motion by earthquakes. A complete analysis of the effects of these processes should be conducted before potential benefits of the ceramic coating can be substantiated.

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5. SENSITIVITY ANALYSES FOR COMPONENTS

The individual process model components of TSPA-VA are described in Section 3 of this volume. The construction of Section 5 of this volume parallels that of Section 3. In Section 5, the TSPA-VA is again discussed based on its component parts to assess the sensitivity of the results to changes in the various parameters used to construct the model of that component. An important task for the TSPA is performing the uncertainty and sensitivity analyses. They are one means of showing which parameters in the analysis have the most influence on repository system performance.

In general, the sensitivity analyses do not show, in an absolute sense, which parameters in the analysis are most important to performance. Instead, they show the parameters in which uncertainty most affects the results. Therefore, by holding a parameter constant in a comparative calculation, the relative reduction in uncertainty is obtained. In some cases, if future studies could reduce the range in uncertainty, the parameter might no longer appear as a parameter to which performance is highly sensitive. Conversely, if a parameter or component is assigned an inappropriately low uncertainty range, it might not show up as a particularly important parameter. Uncertainty analyses must be performed with care and implemented in many different configurations. This process allows analysts to gain the necessary understanding about which parameters are most important to actual repository performance. It can also provide DOE with an indication of which parameters might warrant additional study to achieve the confidence necessary for an adequate licensing argument.

5.1 UNSATURATED ZONE FLOW

5.1.1 Sensitivity to Climate

Three sensitivity studies were conducted to look at the effects of base case assumptions about climate. The sensitivity studies address timing and duration of climate conditions; assumptions about more extreme climate magnitudes or the sufficiency of using only three discrete climate states are not addressed. Climate magnitude is not considered, because the range of uncertainty in net infiltration (from one-third to three times the base value) is intended to cover the range of natural variability.

The first sensitivity study examines the duration of each climate state. In the base case, the long-term-average climate lasts about 90,000 years, the dry climate lasts about 10,000 years, and the super-pluvial climate lasts about 10,000 years. There is some information in the paleoclimate record about how long the Great Basin pluvial lakes were present and how deep they were. This and other information indicates that both the dry and super-pluvial climates lasted longer, perhaps as long as 50,000 years. Therefore, the expected-value base case is compared with a case having 50,000-year climate durations for all three climates but which is otherwise identical to the expected-value base case.

The comparison is shown in Figure 5-1. This figure shows that differences are relatively insignificant between the two cases, suggesting that climate duration, at least at the scale of thousands of years, is not important. There is a slight difference, generally less than a factor of two, in the magnitude of the peaks. The peaks mark the transition from one climate to another and primarily represent the effect of abrupt changes in water table elevation and saturated zone groundwater flux. The superpluvials do not occur at the same time in both cases, so the major peaks in dose rate do not occur at the same time.

The second sensitivity study examines the effect of the base case assumption of an instantaneous transition from one climate state to the next. In this sensitivity study, three calculations were performed, identical to the expected-value base case, except that climate was held constant in each. Therefore, one calculation used a dry climate for the entire 1 million years; one, a long-term-average climate; and one, a superpluvial climate. The results are shown in Figure 5-2. The figure shows that the base case and the case using only the long-term-average climate are virtually identical. The only difference is that the case using the long-term-average only does not show fluctuations in the

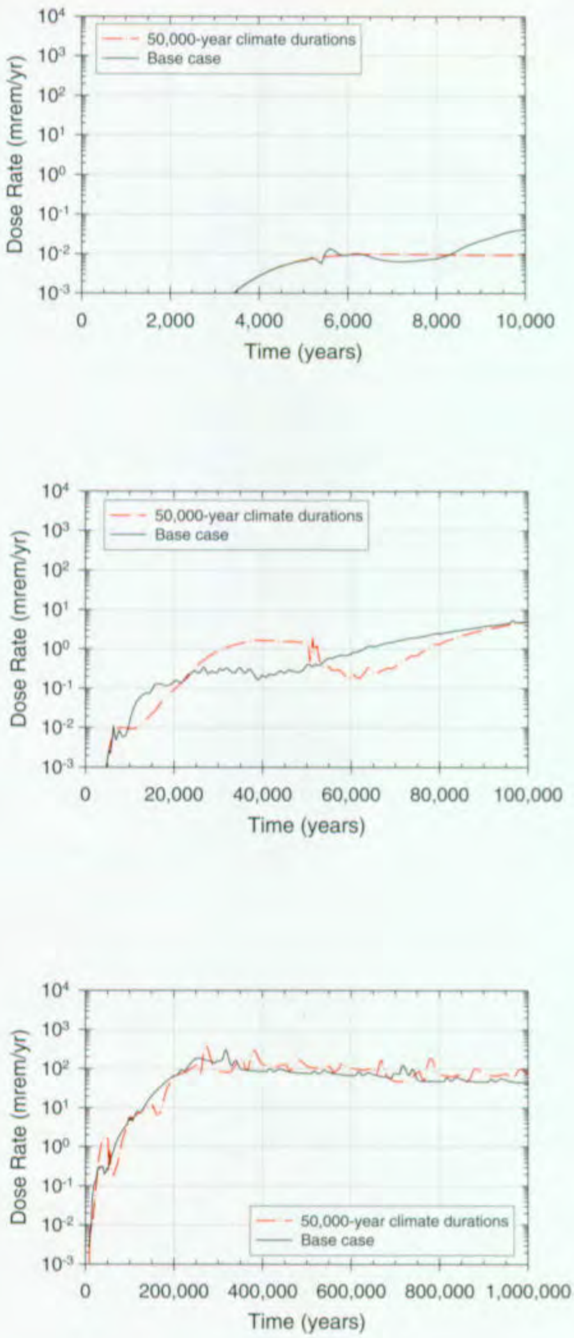

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Figure 5-1. Comparison of the Expected Value Base Case with a Sensitivity Case in Which all Climate Durations are 50,000 Years
The differences in total dose rate are relatively insignificant when climate durations are adjusted by thousands of years.

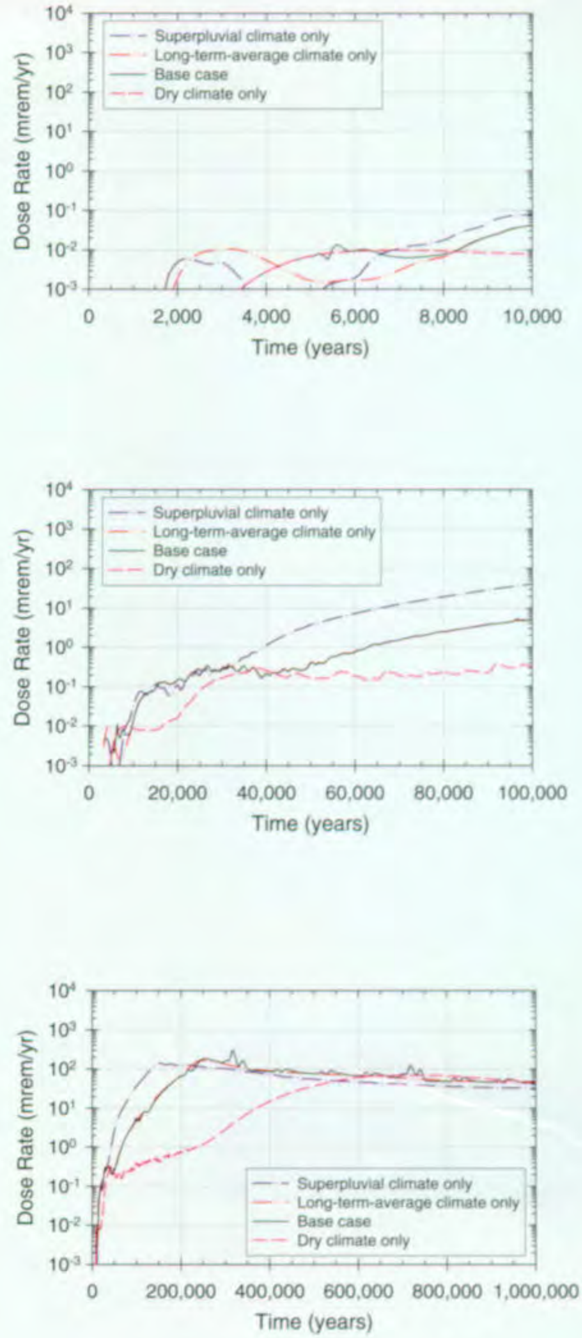

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Figure 5-2. Comparison of the Total Dose Rate for Expected-Value Base Case with Three Sensitivity Cases in Which There are no Climate Changes
The indicated climate state is in effect for the full time period.

magnitude of the dose rate associated with the times of climate changes in the base case.

Figure 5-2 also shows something unexpected. For a 1-million-year period, peak dose rate varies little because of the climate type. The time of the peak dose rate changes in response to the wetter repository conditions associated with wetter climates. However, the additional water that comes with wetter climates tends to compensate for the additional releases of radionuclides by creating additional dilution. The time of the peak dose rate for the dry climate is about 700,000 years; for the long-term average, it is 250,000 years; for the superpluvial, it is 150,000 years. Therefore, the time of peak dose rate is affected by climate, but the magnitude of the peak dose rate is not particularly sensitive to the magnitude of the climate. Indeed, radionuclide decay might be responsible, at least in part, for making the dry-climate peak lower than the other two.

The third sensitivity study investigates the assumption in the base case that climate magnitudes are correlated over time. In the base case, when a high infiltration (i.e., base infiltration times three) is selected for a given realization, all the future climates have infiltrations that are times three, including commensurately high seepage rates. For the third sensitivity study, climate magnitudes were uncorrelated and 100 realizations were simulated. The results are presented as a distribution of peak dose rate in Figure 5-3. Variable climate magnitudes produce higher peak dose rates than more uniform magnitudes. It might seem reasonable that, with variable climate magnitudes, there is a greater chance of getting more extreme climates in any given realization. However, the sensitivity studies show that climate magnitude is relatively unimportant, and the transitions between climates are important for peak dose rate. Variable climate magnitudes create instances when the change from one climate to the next is even greater than in the base case, and the contrast creates the greater dose rate.

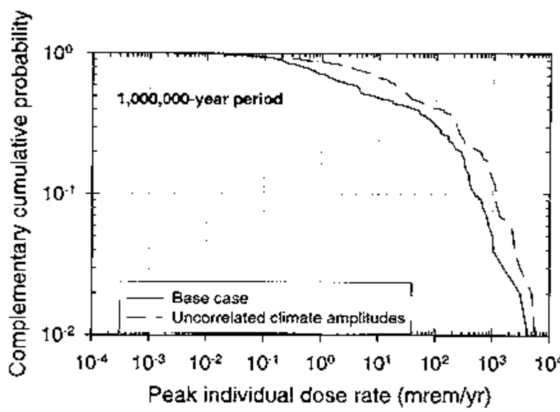

Figure 5-3. Comparison of the Base Case Peak-Dose-Rate Distribution with a Sensitivity Case in Which Infiltration and Climate are Uncorrelated
For example, an "infiltration divided by 3" dry climate could be followed by an "infiltration multiplied by 3" long-term average climate.

5.1.2 Sensitivity to Infiltration

One sensitivity analysis was performed to examine further the effect of the uncertainty in infiltration. In the unsaturated zone flow model expert elicitation (CRWMS M&O 1997n), some experts favored relatively high net infiltration values for Yucca Mountain, so the effect of higher infiltration is particularly of interest.

Performing additional simulations with the infiltration model and the mountain-scale unsaturated zone flow model is very difficult, but changing the probabilities associated with the existing simulations (within a probabilistic TSPA simulation) is relatively easy. Therefore, the effect of infiltration uncertainty was investigated further by changing the probabilities of the three infiltration cases ("base infiltration," "base infiltration divided by 3," and "base infiltration multiplied by 3"). In the base case, the probabilities are skewed to lower infiltrations, so that "infiltration divided by 3" has a 30 percent probability while "infiltration multiplied by 3" has a 10 percent probability. For this sensitivity analysis, the probabilities are reversed so that they are skewed to higher infiltrations: "infiltration divided by 3" is given a 10 percent probability and "infiltration multiplied by 3" is given a 30 percent probability. This change increases the mean infiltration by approximately 50 percent. In the base case, the mean net infiltration averaged over the repository area is a little under 8 mm/year; in this sensitivity analysis the mean net infiltration is about 11 mm/year.

The distribution of peak dose rates over a 100,000-year period for this sensitivity case is compared with the base case distribution in Figure 5-4. There is surprisingly little difference in the two curves, with the base case being higher part of the time and the higher-infiltration case being higher part of the time. The means of the two distributions are nearly the same, both of them about 30 mrem/year. Other factors, such as seepage and waste package corrosion uncertainties, dominate both of these distributions.

5.1.3 Sensitivity to Mountain-Scale Unsaturated Zone Flow

As discussed in Section 3.1.2.3, the base case includes five calibrated, mountain-scale flow fields to account for uncertainty in net infiltration and hydrologic properties, plus the same cases calculated for projected future climates. Figure 5-5 shows a comparison of dose rates calculated for the five base case flow fields, with all other parameters

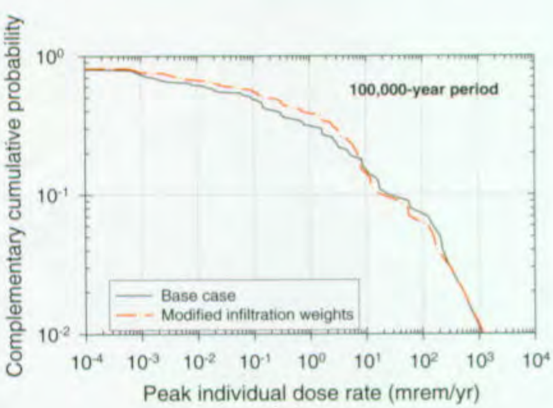

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Figure 5-4. Comparison of the Base Case Dose Rate Results with a Sensitivity Case Involving Infiltration
Comparison of the base case peak-dose rate distribution with a sensitivity case in which infiltration probabilities are changed so that the mean infiltration is 50 percent higher.

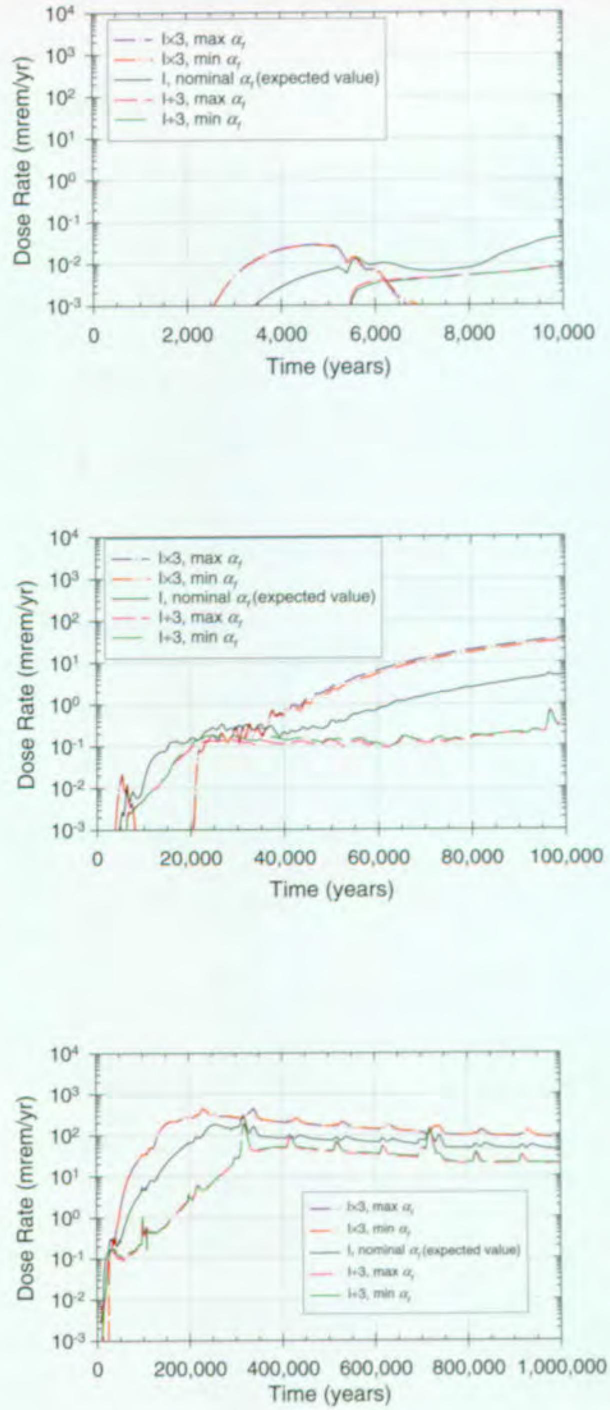

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Figure 5-5. Total Dose Rate History Curves for the Five Base Case Flow Fields, with all Other Parameters Set to Their Expected Values
The middle curve is the expected value base case.

kept the same at their expected values. The infiltration varies from one-third of the base infiltration ("I ÷ 3") to three times the base infiltration ("I × 3"). The fracture air-entry parameter α_f (proportional to fracture aperture) varies over a range that is different for each hydrogeologic unit, but typically the range from minimum to maximum is one to one and one-half orders of magnitude. See CRWMS M&O (1998i, Sec. 2.4.3.2.2) for more detail. The curve marked "I, nominal α_f" is the expected-value base case.

Figure 5-5 shows a fairly large effect on dose rate from the variation in infiltration but very little effect from the variation in fracture air-entry parameter. These observations are consistent with those made for the effects on unsaturated zone groundwater travel time (Figure 3-11). However, most of the effect on dose rate from varying the infiltration is because of the corresponding change in the amount of seepage and not because of changes in transport times. One surprising aspect of the high-infiltration dose-rate histories is that, after an initial peak, the dose rate drops to a very low rate from approximately 7,000 to 20,000 years and is below the base-infiltration dose rate until about 35,000 years. This sharp drop occurs because temperatures are lower for higher net infiltrations (see Section 3.2.3), leading to lower corrosion rates and later onset of waste package failures because of corrosion (see Section 3.4). The initial peak in the high-infiltration dose rate at about 5,000 years is caused by releases from a juvenile-failure waste package; corrosion failures do not start contributing until after 20,000 years. The mean and low infiltration cases also have early releases from a juvenile-failure waste package, but releases from corrosion-failed waste packages follow soon thereafter (see Section 3.4).

Figure 5-6 shows a comparison of dose-rate histories for the expected-value base case and for a sensitivity case; the calculations are the same except that the base-infiltration flow field from the dual-permeability/Weeps model was used. Note that, for this set of calculations, the base case thermal-hydrology and waste package degradation results were used for the dual-permeability/Weeps runs in order to show the effect of the difference in

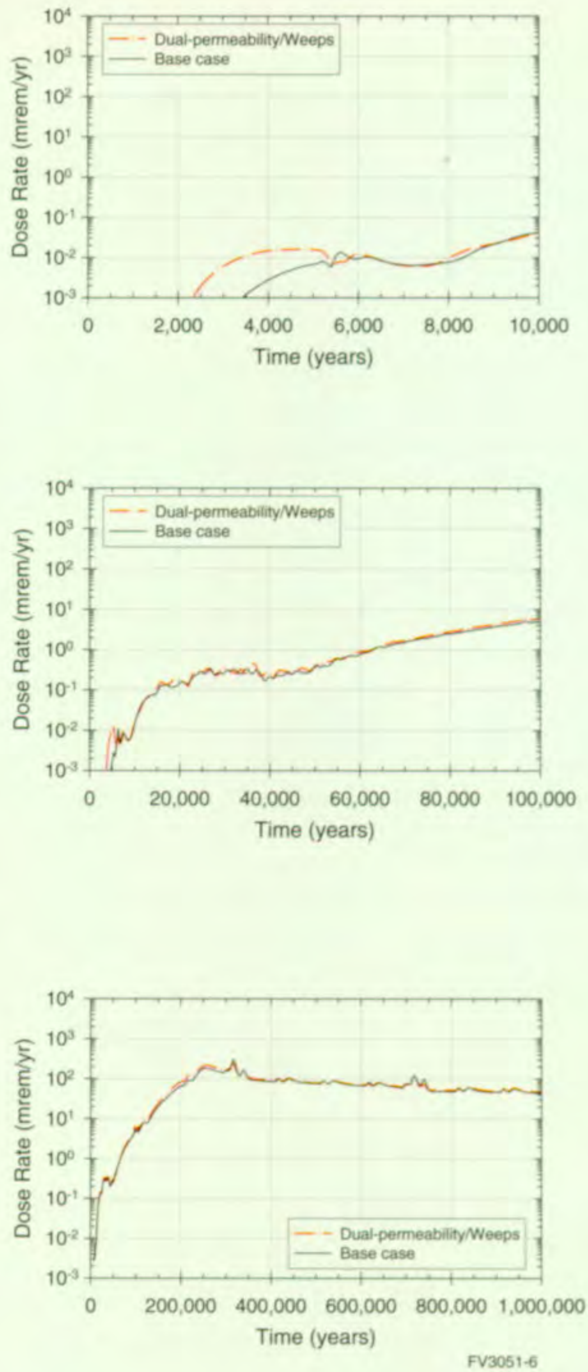

Figure 5-6. Comparison of the Total Dose Rate for Expected-Value Base Case with a Sensitivity Case that Used the Dual-Permeability/Weeps Flow Model Rather than the Base Case Flow Model
The two are nearly identical except that the dual-permeability/Weeps results show an earlier initial breakthrough.

mountain-scale flow only. Dual-permeability/Weeps results including more consistent thermal-hydrology and waste package degradation are given in Section 5.2. The dose-rate histories for the two cases in Figure 5-6 are nearly the same except for an earlier breakthrough and slightly higher early peak for the dual-permeability/Weeps case, both caused by the greater amount of fast fracture flow in the dual-permeability/Weeps model (see Figure 3-12).

A probabilistic comparison was also made between the base case flow model and the dual-permeability/Weeps flow model. Figure 5-7 shows the distributions of peak dose rate over 100,000 years. The two distributions are very similar except in the low-probability tail. About 3 to 4 percent of the time, the peak dose rates from the dual-permeability/Weeps model are significantly higher than the base case (about a factor of three higher). This increase in the lower part of the curve can be traced to transport of colloidal plutonium. Because of the greater prevalence of fast fracture flow in the dual-permeability/Weeps model, more of the colloids are able to travel through the system quickly to the biosphere for those realizations that have very mobile plutonium colloids. This happens a small

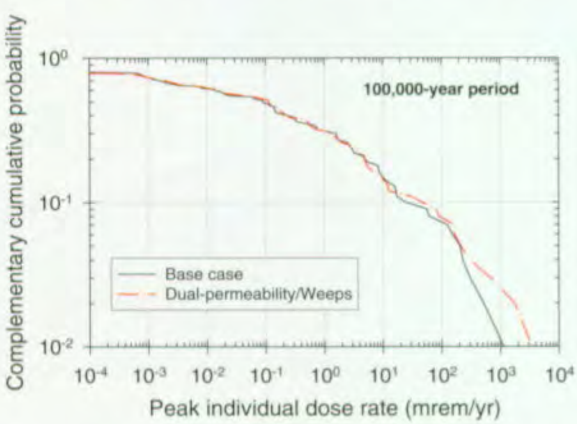

Figure 5-7. Comparison of the Base Case Peak-Dose-Rate Distribution with a Sensitivity Case in Which Dual-Permeability/Weeps Flow Fields are Used Instead of Base Case Flow Fields
About 3 to 4 percent of the time, the peak dose rates from the dual-permeability/weeps model are significantly higher than the base case because of faster transport of colloidal plutonium.

percentage of the time with the base case assumptions (see Section 4.3.1.1).

5.1.4 Sensitivity to Seepage into Drifts

The sensitivity of individual dose rate at 20 km (12 miles) from the repository to seepage into drifts is illustrated in Figure 5-8, which shows dose-rate time histories for three discrete cases: the expected-value base case plus runs in which seepage fraction (fraction of waste packages contacted by seeps) and seep flow rate are assigned values at the fifth and ninety-fifth percentiles of their probability distributions (see Figure 3-13), with all other parameters being held fixed at their expected values. The effect is strongest on the low end, with the fifth-percentile case having no waste package failures except for a single juvenile failure until after 700,000 years. The lack of waste package failures is a direct result of the very low seepage fraction—so low that no waste packages are wetted by seepage in the fifth-percentile case. Because the juvenile failure dominates radionuclide releases for the first 10,000 years, and juvenile-failure waste packages are always assumed to be contacted by seeps, the dose results are fairly similar for all three cases for the 10,000-year period (Figure 5-8, top), but for the 100,000-year and 1-million-year periods the dose results are quite different for the fifth-percentile case because of the lack of corrosion failures.

Because of the importance of seepage to the final results and the large uncertainty in the seepage model, several probabilistic sensitivity analyses were performed for seepage. Similar to the situation with infiltration and mountain-scale unsaturated zone flow, the process-model computations on which the TSPA-VA seepage model is based are very time consuming. However, the probabilities assigned to the various process-model cases can easily be modified to explore the impact on the final dose results.

The results of two kinds of modifications to the seepage-model inputs are presented here. One is to narrow the fracture-property distributions (i.e., assume that the fracture properties have less uncertainty than in the base case) and the other is to

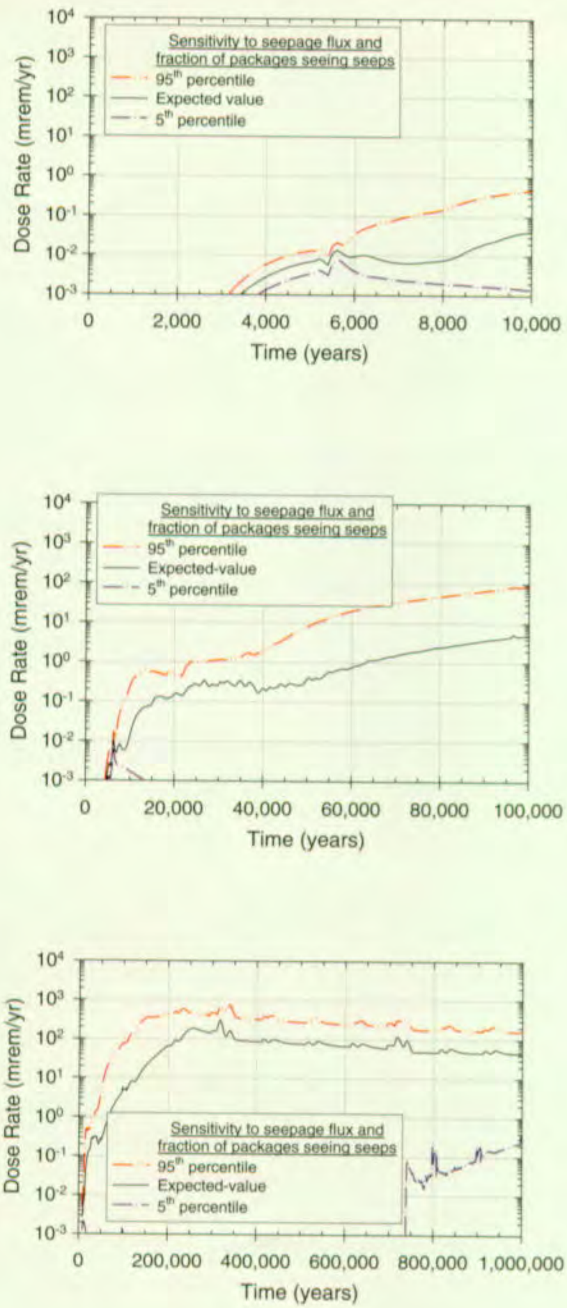

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Figure 5-8. Total Dose Rate History Curves with Seepage Influence
Dose-rate history curves for the expected-value base case and cases that have seepage fraction and seep flow rate at the 5th and 95th percentiles of their base case distributions.

skew the distributions toward higher or lower fracture apertures.

As discussed in Section 3.1.2.4, three values of mean fracture permeability (k_f) and three values of fracture air-entry parameter (α_f) were used in the drift-scale flow simulations that were performed to calculate the amount of seepage into emplacement drifts. The base case probabilities for both parameters were assigned as (0.25, 0.5, 0.25); that is, the lowest value was weighted 25 percent, the middle value 50 percent, and the highest value 25 percent. The middle value is the preferred value, based on the available data, and the low and high values were chosen to span a reasonable range based on the variability in the data. For the first sensitivity analysis, the probabilities of the outlying values were cut in half, giving probabilities of (0.125, 0.75, 0.125). These probabilities emphasize the best-estimate parameter values considerably more. They are also reasonable, given what is known about fracture properties. Taking fracture permeability as an example, the base case probabilities imply a log standard deviation (i.e., the standard deviation of the logarithm of permeability) of 0.7, while the narrower distribution implies a log standard deviation of 0.5. These standard deviations are near the low and high ends of the observed range for air permeability measurements in the repository host rock (see Table 7.11 of Bodvarsson et al. 1997).

This modified set of probabilities was applied to the process-model results to generate new seepage distributions, and then calculations were rerun with the modified seepage distributions. The results are shown in terms of dose-rate history for the mean parameter values in Figure 5-9 and in terms of distribution of peak dose rates in Figure 5-10. This first sensitivity case is shown as the curves labeled "Narrower variance on fractures" in the figures. The results are not very different from the base case results, although the peak dose rates tend to be slightly smaller.

The other two curves shown in Figures 5-9 and 5-10 are for sensitivity cases with modified probabilities that represent systematically smaller or larger fractures. These cases could approximate

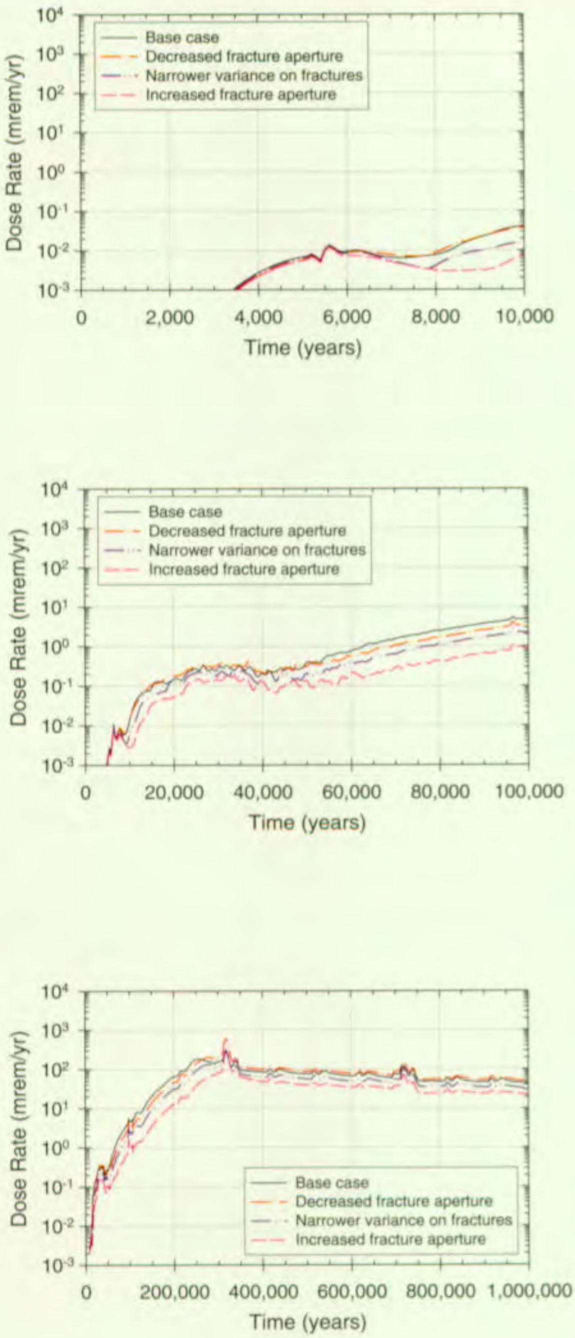

Figure 5-9. Dose Rate History Curves for Three Sensitivity Cases
Total dose-rate history curves for the expected-value base case and for three sensitivity cases in which the probabilities of the fracture-property inputs to the seepage model are modified.

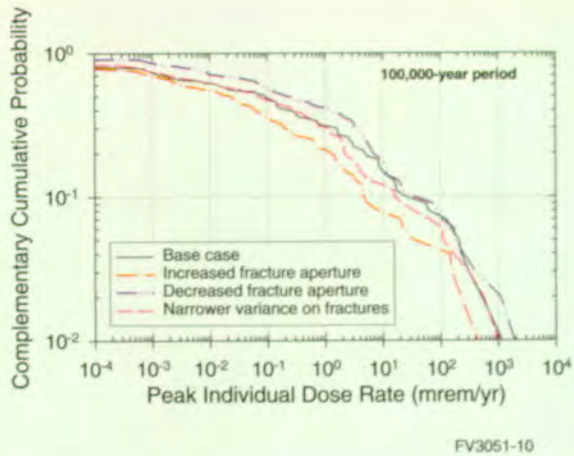

Figure 5-10. Comparison of Base Case Peak Dose Rate with Sensitivity Cases Involving Fracture Aperture
Comparison of the base case peak-dose-rate distribution with three sensitivity cases in which the probabilities of the fracture property inputs to the seepage model are modified.

the possibility of smaller or larger fracture apertures caused by thermal-hydrologic-chemical or thermal-hydrologic-mechanical processes. Smaller fracture apertures imply both lower k_f and lower α_f, because α_f is proportional to the effective fracture aperture and k_f is proportional to the cube of fracture aperture. Therefore, to simulate smaller fracture apertures, the probabilities for the parameter values were shifted to smaller values, giving (0.67, 0.33, 0) as the low, middle, and high parameter values. Similarly, to simulate larger fracture apertures the probabilities were changed to (0, 0.33, and 0.67). Time-history curves for the mean parameter values are shown in Figure 5-9 and the distributions of peak dose rate are shown in Figure 5-10. For smaller fracture apertures, the peak dose rates are somewhat higher, and for larger fracture apertures, the peak dose rates are somewhat lower. Basically, the probability shift to smaller apertures causes more seepage, and the shift to larger apertures causes less seepage. The differences in seepage are then reflected in the final dose results. None of these seepage sensitivity cases causes greatly different dose rates; the one that has the most effect is the increased-fracture-aperture case, which has peak dose rates lower than the base case by a factor of about 3 to 10 for the

100,000-year period, and somewhat less for the other two time periods.

5.2 THERMAL HYDROLOGY

The most important thermal hydrology issues are related to hydrologic parameters and fracture-matrix interaction and to repository design, including backfill, thermal load, and waste package spacing. Some variations in repository design were considered in this TSPA, including emplacement of backfill 100 years after emplacement of waste. Analyses of alternative designs are discussed in Section 4.5. Issues related to hydrologic parameters and fracture-matrix interaction are discussed in Section 5.1 on unsaturated zone flow. In most cases, the thermal-hydrologic implications of hydrologic issues such as climate change, infiltration uncertainty, hydrologic-property uncertainty, and alternative flow models were included. These issues are incorporated in the sensitivity analyses in Section 5.1, particularly in Section 5.1.3, where lower temperatures related to high infiltration rates are shown to have a strong effect on doses for approximately the first 20,000 years. One exception is the sensitivity to alternative conceptual models of mountain-scale unsaturated zone flow, exemplified by the dual-permeability/Weeps model. The results shown in Figures 5-6 and 5-7 were obtained using base case thermal hydrology rather than incorporating thermal-hydrology results calculated using the dual-permeability/Weeps model. Figure 5-11 shows results of a more consistent set of calculations, in which the dual-permeability/Weeps model was used for the thermal-hydrology calculations as well as for the mountain-scale flow calculations. In addition to the earlier breakthrough noted in Figure 5-6, the dual-permeability/Weeps dose rate in Figure 5-11 remains a little bit higher than the base case dose rate for approximately the first 250,000 years. The higher dose rate results from a slightly higher rate of waste package failures at early times, caused by small differences in temperatures and relative humidities.

Figure 5-11 also shows two sets of curves for the "thermal-hydrologic" property set that was mentioned in Section 3.2.3. The curves labeled

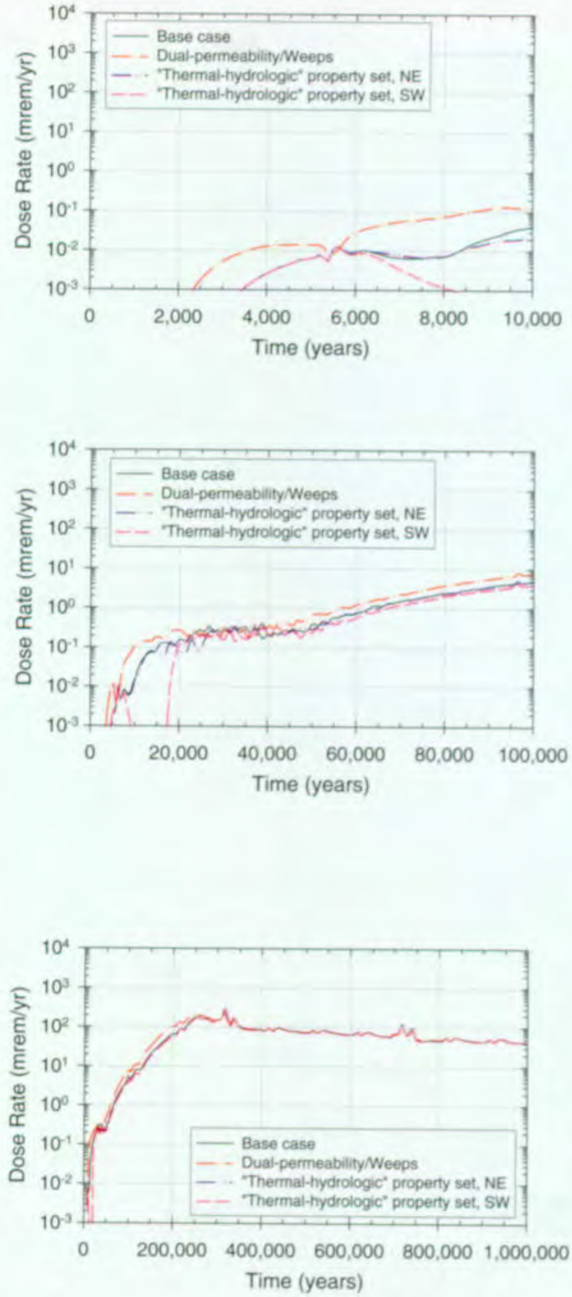

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Figure 5-11. Total Dose Rate History Curves for the Expected-Value Base Case and for Three Sensitivity Cases in Which Different Hydrologic Properties Were Used
The greatest difference is for the case in which the "thermal-hydrologic" property set was used and temperature and relative humidity for Region SW (see Figure 3-20) were applied to the entire repository.

"NE" use the thermal-hydrologic results for Region NE for the entire repository, which was the approximation used for the base case (see Section 4.1.4), while the curves labeled "SW" use the thermal-hydrologic results for Region SW for the entire repository. For the base case, the results for Regions NE and SW are nearly the same, but for the thermal-hydrologic property set results for Region SW are significantly different for about the first 20,000 years. The difference is a result of significantly different matrix hydrologic properties for Region SW in the thermal-hydrologic property set (CRWMS M&O 1998i, Section 3.6.2). It is important to note that the large difference shown in Figure 5-11 exaggerates the actual effect, because of using the Region SW results for the entire repository. Region SW is actually rather small (see Figure 3-20), and the results for the rest of the repository are more like Region NE than Region SW.

Other potentially important issues—thermal-hydrologic-chemical and thermal-hydrologic-mechanical processes and effects—have largely been neglected. Some subsystem-level analyses of these effects are discussed in Section 3.2.1 of this volume, and in Section 3.6.8 of CRWMS M&O (1998i). At the system level, the effects of decreasing or increasing fracture apertures because of thermal-hydrologic-chemical or thermal-hydrologic-mechanical processes were evaluated approximately by modifying the seepage-model input parameters (see Section 5.1.4). The change in calculated peak doses was relatively modest, but these aperture changes do not necessarily represent the full range of possibilities.

5.3 NEAR-FIELD GEOCHEMICAL ENVIRONMENT

The models and results for the near-field geochemical environment component are discussed in Section 3.3. Analyses for the TSPA-VA base case used a water composition that represents the incoming, thermally perturbed fluid that reacted with corrosion products (iron oxides). Changes to the composition of this water as it reacts with freshly exposed spent nuclear fuel and subsequent alteration products were also evaluated in Section 3.3 and those analyses were used as part of the conceptual representation of secondary-phase evolution for the waste form (see Sections 3.5 and 5.5). Because concrete is a major component of the repository design, near-field geochemical analyses were also performed to simulate the reaction of concrete and water. These analyses and results (Section 3.3), although not part of the base case, are used for some of the sensitivity studies in this section. These sensitivity studies evaluate the effect of concrete on both the total system performance and the engineered barrier system performance. Near-field geochemical analyses were also used to develop the conceptual scenarios for sensitivity studies of other component models. The conceptual linkage between the near-field geochemical environment and the sensitivity studies for other TSPA components are discussed below, but the results of those analyses are provided in other sections as indicated below.

5.3.1 Sensitivity of Water Composition to Spent Fuel Alteration

For periods in which water composition is dominated by spent nuclear fuel reactions, different source-term constraints may be necessary than for periods of unaltered water composition. Spent fuel dissolution rates, radionuclide solubility limits, and radionuclide transport (colloidal and aqueous) should particularly reflect the altered solution composition. The impacts on performance from reaction with spent nuclear fuel depend on the magnitude of the water composition changes and the period over which the changes exist.

The near-field geochemical analyses showing the effects of spent-fuel reaction on water composition are presented in Section 3.3.3. Those results indicate that the water composition will change the most where most of the fuel is available for reaction, with lesser effects if only a portion of the fuel is exposed to the water. In addition, maintaining such water composition changes depends on active alteration of primary spent nuclear fuel to secondary uranium phases. Once the fuel is completely altered, the secondary-phase evolution causes only minor changes to water composition. The modeled period for completely altering the spent nuclear fuel once it is exposed is about 1,000 years for the analyses discussed in Section 3.3.3, but the calculated evolution of secondary phases continues over hundreds of thousands of years. These time differences indicate that reaction of spent nuclear fuel with water will produce either major changes to the fluid composition for relatively short times or minor effects over longer times. For the latter case, the current representations should be sufficient to bound the behavior of radionuclide release from the waste form. For the former case, the analyses could be improved by considering a short time period when water composition is dominated by releases from spent nuclear fuel followed by evolution of secondary uranium phases over geologic time.

Although changes to both solubility limits and colloid stabilities have not yet been evaluated for the geochemical conditions during the period of active alteration, changing spent nuclear fuel dissolution rates are explicitly coupled to changing water chemistry in the completed near-field geochemical analyses. Because of the potential for incorporating other actinides (for example, neptunium) into the secondary uranium phases, the primary focus of these sensitivity studies was the long period of secondary-phase evolution. Also, these secondary phases dissolve or alter at a much lower rate than the primary spent nuclear fuel (Section 3.5.3). The model incorporating neptunium into secondary uranium minerals and the resulting rates of release of uranium and neptunium is presented in Section 3.5. The results of sensitivity studies using secondary-phase constraints on releases of both uranium and neptunium over long time frames are shown in Section 5.5.

5.3.2 Sensitivity to Concrete-Modified (Alkaline) Water Compositions

This section provides a discussion of the effect of concrete modified water on performance. The components affected by the concrete-modified water are described followed by performance results for such a system.

5.3.2.1 Components Affected by Concrete-Modified Water

The near-field geochemical analyses for the composition of water reacting with concrete support components in the emplacement drifts indicate that water will have a pH near 11 for at least 10,000 years but for much less than 100,000 years. This pH change represents a substantial change to ambient water composition and may affect the ability of the engineered barrier system to contain radionuclides within the drift. This would in turn change the source term for total system performance analyses. In addition, higher pH fluids migrating into the geosphere can react with the host rock, potentially altering minerals along flow paths and providing an aqueous medium that enhances actinide transport in the unsaturated zone. Both of these aspects of the analysis were assessed with sensitivity studies. As described in the following paragraphs, impacts on the source term and the engineered barrier system were explicitly addressed based on the changed fluid composition. The impacts on transport through the geosphere were evaluated through simulations using conservative assumptions.

Source Term and Engineered Barrier System. The effects of concrete-modified water on a number of the components related to the source term have been explicitly evaluated. The waste package corrosion models (Section 3.4) include a submodel for aqueous corrosion of the outer barrier, specifically designed to address pH conditions above 10. This submodel was used to generate a set of waste package degradation histories for concrete-modified water. These waste package models and results are presented in Sections 3.4 and 5.4. The waste form dissolution models directly incorporate rates that depend on pH and total dissolved carbonate (Section 3.5). Changes to the ionic strength for concrete-modified water were reflected in changes to colloid amounts (Sections 3.3 and 3.5). These results are provided directly to the TSPA integration model, and impacts on performance were analyzed in a sensitivity analysis based on the expected value compositions for concrete-modified water. The effects on the source term are presented in the following paragraphs of this section. Additional discussion of these results that are related to changes in waste package degradation is included in Section 5.4.

Transport Through the Geosphere. The migration of alkaline fluids into the unsaturated zone may have a number of effects, depending on the flow pathway and diffusion rates. Such fluids can alter the siliceous host rock, both along fracture pathways and in the matrix. The mineralogical changes, the distance over which they may occur, and the distribution of such alterations are still very uncertain but could produce changes in the amount of fracture-matrix interaction and the sorption of radionuclides within the unsaturated zone host rock (CRWMS M&O 1998i, Chapters 4 and 7). Because of the complex and uncertain nature of this alteration, current process models do not include any explicit representation of these potential changes to minerals along radionuclide migration pathways. In addition, the migration of a high-pH (alkaline) plume through the unsaturated zone would create aqueous pathways with little or no actinide sorption capability (CRWMS M&O 1998i, Chapters 4 and 7). Also, this potential effect has not been incorporated explicitly into process models. Because of the likelihood that alkaline fluids will be generated by water in contact with concrete in the repository (Section 3.3.3), this effect was included in sensitivity analyses. To bound the consequences of this effect, the unsaturated zone sorption coefficients for actinides (uranium, neptunium, plutonium, and protactinium) were set to zero (Section 5.6). The changes to dose rates specifically caused by impacts to transport in the unsaturated zone, as well as the changes caused by combining the impacts to the source term with the transport impacts, are presented in Section 5.6.

5.3.2.2 Effect on Dose Rate and Engineered Barrier Release Rate

In this section, sensitivity analyses are presented based on changes to the source term only (waste package degradation and waste form degradation) caused by concrete-modified water.

Release Rate from the Engineered Barrier System. The expected value release rates from the engineered barrier system for the total activity of nine main radionuclides (carbon-14, selenium-79, technetium-99, iodine-129, protactinium-231, uranium-234, neptunium-237, plutonium-239, and plutonium-242) through 10,000 years are shown in Figure 5-12. The rates for both the concrete-modified water case and the base case are shown. In Figure 5-12, the total activity release rate for the concrete-modified water case is higher than that for the base case for all times to 10,000 years. Early in this period, the difference is about two to three orders of magnitude but is only about 1.5 orders of magnitude at 10,000 years. For the concrete-modified water case, the peak total release rate from the engineered barrier system is increased by slightly more than three orders of magnitude over that for the base case. These two peaks both occur at about 5,200 years, reflecting the climate change that is imposed at 5,000 years. These higher total release rates from the engineered barrier system primarily reflect a larger amount of exposed inventory at earlier times caused by higher rates of waste package degradation (see Section 5.4).

The differences between these two cases become less extreme between 10,000 years and 100,000 years, as shown in Figure 5-12. The release rate in the engineered barrier system for the concrete-modified water case is a factor of two to six times higher than for the base case after about 15,000 years. The individual radionuclide contributions to the total activity release rate for the concrete-modified water case over the three time frames are shown in Figure 5-13. Technetium-99 is the major component of this total activity release rate for the entire time. At times less than about 15,000 years, carbon-14 and iodine-129 were the next highest contributors. At times greater than 30,000 years, plutonium-239 and neptunium-237 were the second and third largest contributors, respectively, to the total activity release rate. The activity release rates for these radionuclides increase in a relatively uniform manner over the base case results because of an increase in the inventory available for release compared to the base case.

Dose Rate at the 20-km (12-mile) Boundary (Accessible Environment). The increases in the source term directly impact expected-value dose

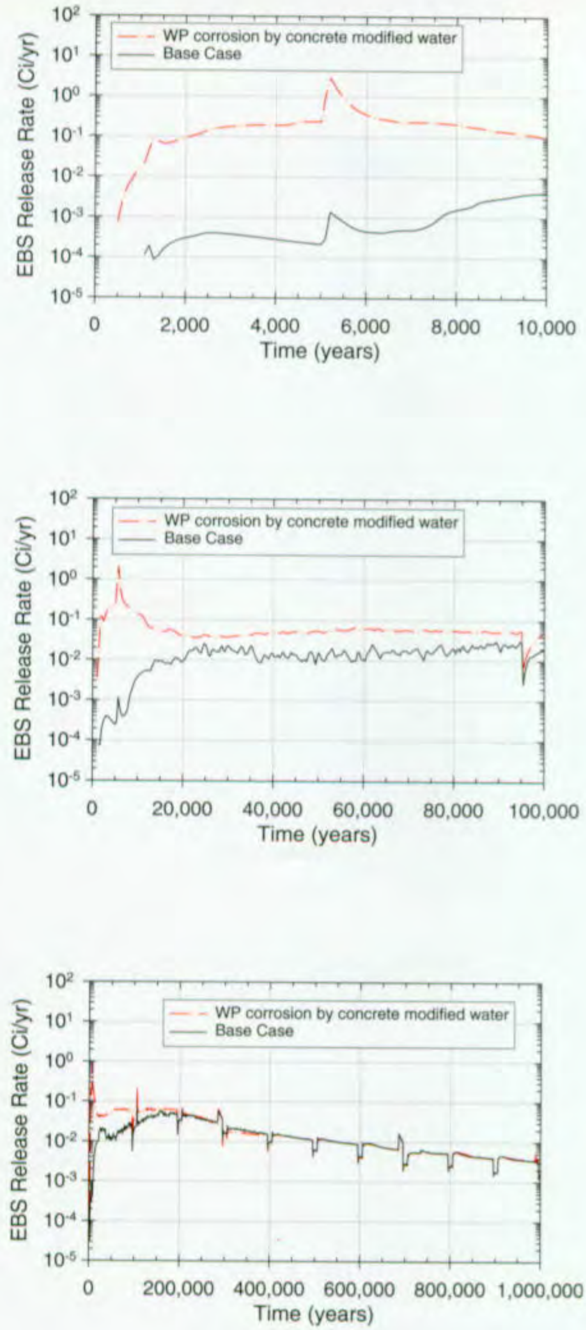

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Figure 5-12. Comparison of Expected-Value Total Release-Rate Histories from the Engineered Barrier System for Concrete-Modified Water and the Base Case
The release rates include the total activity of nine radionuclides.

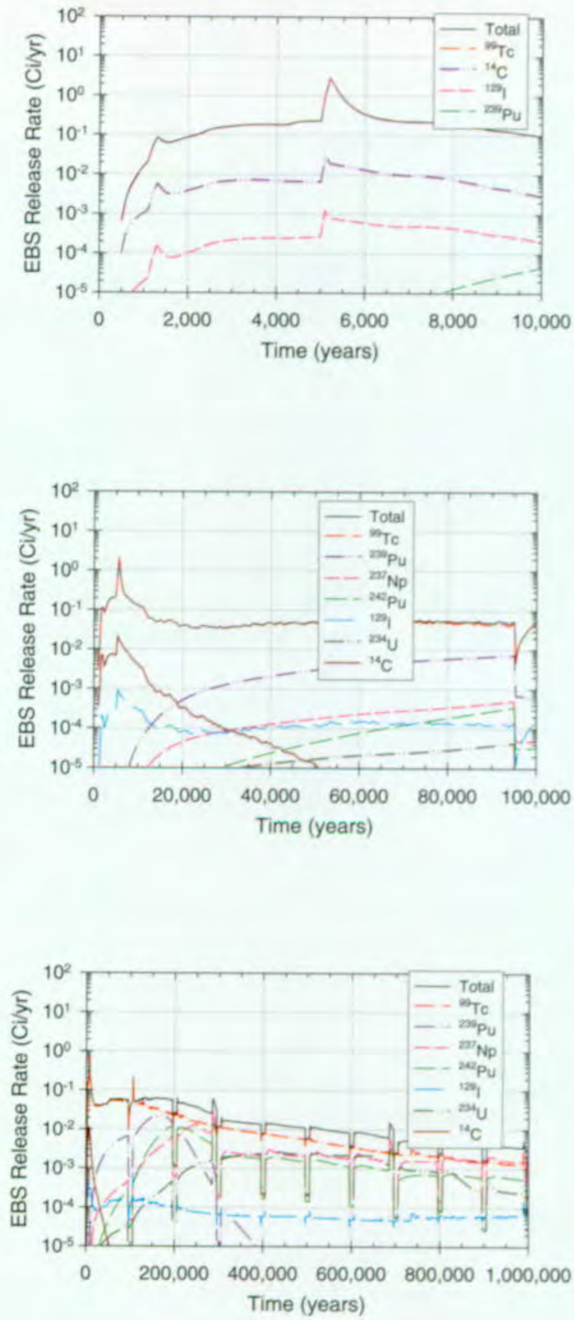

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Figure 5-13. Expected-Value Time Histories for Individual and Total Radionuclide Release Rates from the Engineered Barrier System for the Concrete-Modified Water Case rates at the 20-km (12-mile) boundary, which are shown in Figure 5-14 for both the concrete-modified water case and the base case over 10,000 years, 100,000 years, and 1 million years. The orders-of-magnitude increases in activity release rate for the engineered barrier system are reproduced in the orders-of-magnitude increases in expected values of dose rate over 10,000 years. Note that for the concrete-modified water case, the peak value at about 6,100 years is slightly more than three orders of magnitude higher than the values for the base case at that time. One qualitative change that can be seen for the concrete-modified water case in Figure 5-14 is that the peak dose rate at about 6,100 years is the peak dose rate over the first 100,000 years after repository closure; this is not true for the base case results. Figure 5-14 also shows that the expected value dose rate histories for these two cases later converge, similar to the source term behavior.

In the concrete-modified water case for the 10,000-year time frame, technetium-99 is the major dose contributor as well as the major contributor to activity release rate from the engineered barrier system. At times approaching 100,000 years, plutonium-239 and neptunium-237 are the biggest contributors to dose rate, even though technetium-99 was still the largest contributor to the activity release rate for the engineered barrier system. Further discussion of the impacts on the engineered barrier system and system performance as evaluated in the sensitivity analyses for concrete-modified water is given in Sections 5.4, 5.5, and 5.6.

5.4 WASTE PACKAGE DEGRADATION

The key uncertainties in the waste package degradation analyses were identified in Sections 3.4 and 4. The results in Section 4 indicate that the general corrosion rates for Alloy 22 under dripping conditions are a key determinant of overall performance. These corrosion rates are highly uncertain. The corrosion rates provided by the experts have a range of more than five orders of magnitude. This is mostly caused by a lack of information on local chemical and electrochemical conditions on Alloy 22 and limited experience with the alloy.

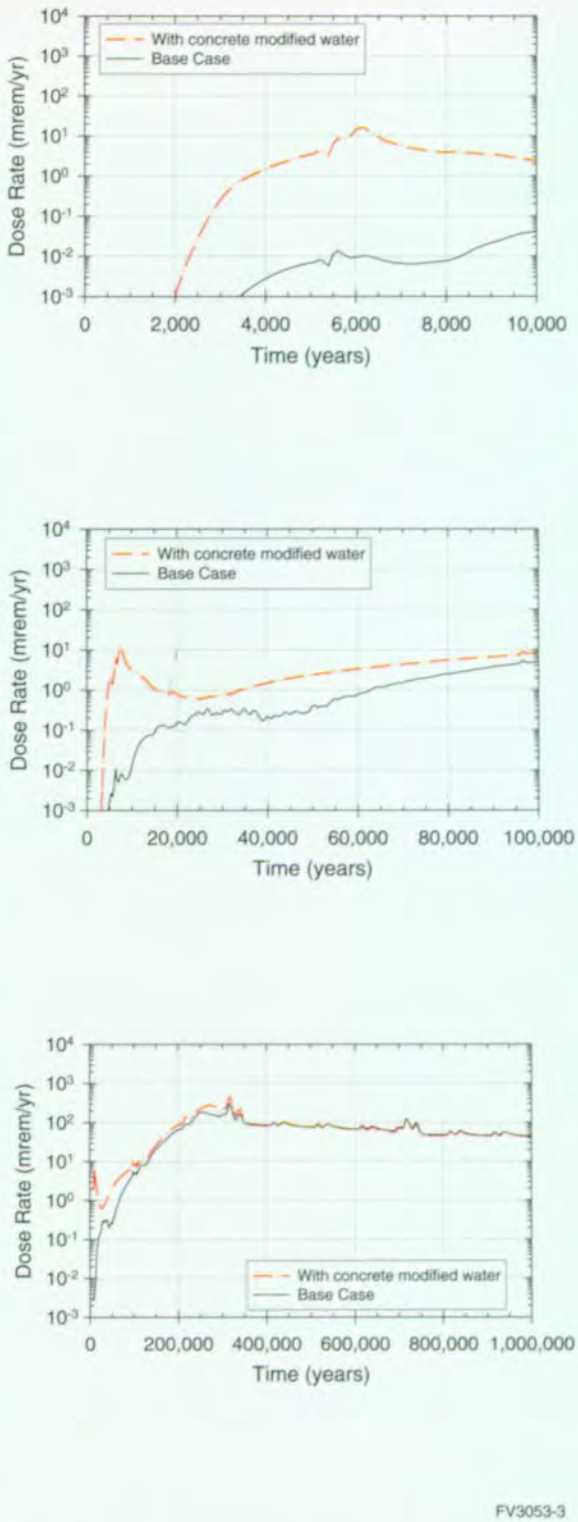

Figure 5-14. Comparison of Expected-Value Total Dose-Rate Histories for the Concrete-Modified Water Case and the Base Case

Also, the effect of the emplacement environment on waste package degradation has been identified as very important, especially in terms of overall performance of the potential repository. The number of waste packages that will be exposed to dripping water, the percentage of the waste package surface that gets wet in the dripping zones, and the chemistry of the drift seepage are also uncertain factors in the analyses. The effects of these uncertainties on repository performance are evaluated in the following sensitivity cases. In addition, the importance of juvenile failures on system performance is evaluated. A final sensitivity analysis is presented concerning the effect on total dose of the corrosion patch size, which is the area over which the corrosion model parameters are relatively constant.

5.4.1 Sensitivity to Uncertainty/Variability Assumptions in Alloy 22 General Corrosion Rates

There are limited data for the general corrosion rate for Alloy 22. However, the corrosion rate over long periods is highly uncertain due to limited experience with the material and lack of information on local exposure conditions on the inner barrier. Panelists for the waste package degradation expert elicitation (CRWMS M&O 1998b) estimated the Alloy 22 general corrosion rate and the allocation of the total variance to its variability and uncertainty. The effect on dose of the corrosion rate variability among waste packages and patches and the corrosion rate uncertainty on waste package failure was evaluated by splitting the total variance into three different variability and uncertainty combinations: 75 percent variability and 25 percent uncertainty, 50 percent variability and 50 percent uncertainty, and 25 percent variability and 75 percent uncertainty. For each variability and uncertainty split, the median general corrosion rate was sampled from the 5th, 50th, and 95th percentile of the uncertainty variance respectively. This sampling scheme creates a matrix of nine pairs of values for the median general corrosion rate and the degree of variability in the corrosion rates when spread over corrosion patches on a given package and from package to package.

The expected-value base case described in Section 4.2 allocates a split of 50 percent variability and 50 percent uncertainty of the total variance, with the median general corrosion rate at the 50th percentile of the uncertainty variance. The results for the cases with different variability-uncertainty splits and the median general corrosion rate at the 50th percentile of the uncertainty variance show a difference in the performance of the waste package, with a large spread in the failure times and rates (CRWMS M&O 1998i). The results for the base case variability-uncertainty split (50/50) with the median corrosion rate sampled at the 5th, 50th, and 95th percentile from the uncertainty variance show a considerable difference in waste package failure rate, as indicated in Figure 5-15. The failure curves in this figure are shown only for the waste packages that are dripped on. The impact of this range of Alloy 22 general corrosion rate on the uncertainty range of dose rate is demonstrated in Figure 5-16 for the TSPA-VA base case. These dose rate histories further confirm the results shown in Figure 4-34 that the Alloy 22 corrosion rate has more impact on dose than most other TSPA-VA model parameters, especially over the 10,000-year and 100,000-year periods.

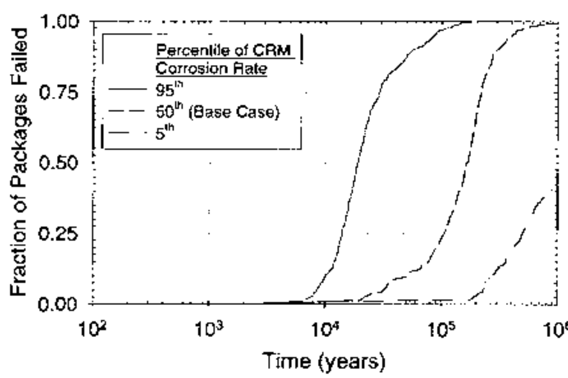

Figure 5-15. Effect of Uncertainty in Alloy 22 Corrosion Rate on Package Failure Time
The effect on waste package degradation is shown for the 5th, 50th and 95th percentile corrosion rates for Alloy 22.

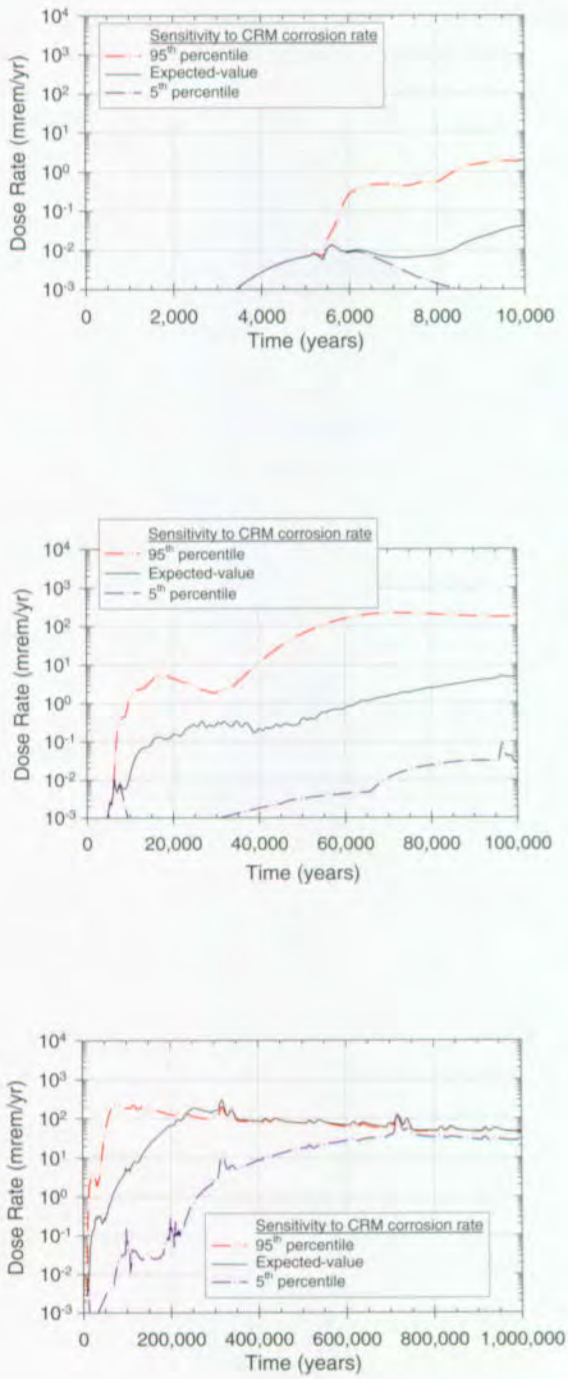

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Figure 5-16. Effect of Uncertainty in Alloy 22 Corrosion Rate on Total Dose Rate
Results are shown for the 5th, 50th, and 95th percentile corrosion rates for Alloy 22 for the base case uncertainty distribution.

5.4.2 Sensitivity to Environment (Drip Versus No Drip Conditions)

Waste package degradation is sensitive to the impacts of water dripping onto the waste package, as noted in Section 3.4. The general corrosion rate of Alloy 22 is extremely low under moist conditions (a thin water film on the surface, but no flowing water), but the material may undergo a more rapid, although still relatively slow, general corrosion under dripping conditions. Figure 5-17 shows the difference in waste package degradation for the two conditions in the base case. The first penetration of the waste packages in the dripping zone occurs several hundred thousand years before penetration of the waste packages that are not dripped on. Only a small fraction of the waste packages in the nondripping portion of the repository are penetrated over 1 million years, while the simulation indicates nearly 100 percent of the waste packages in the dripping zone develop at least one penetration.

5.4.3 Sensitivity to Percent of Waste Package Surface Wetted Under Dripping Conditions

Uncertainty about seepage into the drift, and subsequently onto waste package surfaces, is present at a variety of different spatial-temporal scales. A

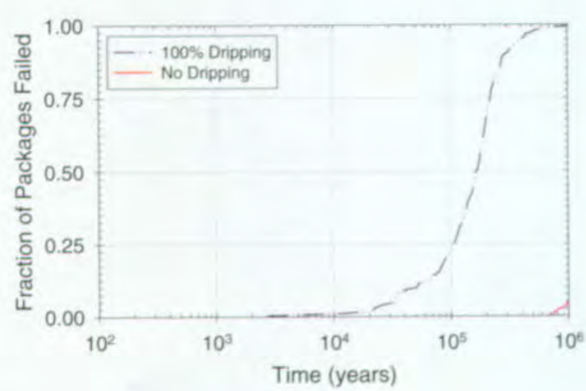

Figure 5-17. Effect of Dripping Conditions on Waste Package Failure for the Base Case
Waste packages in dripping conditions fail more rapidly than those in nondripping zones.

variety of different alternative models might be imagined for the percentage of waste package surface area that is wet as a function of time and location. To determine the sensitivity of performance to the wetted surface area, three different models have been examined. These three cases represent a combination of two main factors: percent of waste package area corroded and percent of drift seepage that enters the package through the corroded area and subsequently interacts with all the exposed (i.e., non-clad) fuel:

1. *TSPA-VA base case.* In this case 100 percent of the surface area of a dripped-on package is assumed to be dripped upon and to corrode under dripping conditions for all time. The flux of dripping water that is allowed to flow through the waste package is set equal to the flux of water seeping into a drift multiplied by the combined patch-pit area divided by the total waste package surface area. This seeping water flowing through the package is assumed to interact with all the exposed waste in the package (i.e., all waste that is either unclad or whose cladding has failed). The conceptual scenario most suited for this model is one where a spatially focused seep moves back and forth across the package on a rather short time scale such that it contacts the entire package area (at least the top half of the package) within this short time span, say, within 100 years, which is the numerical timestep size for the total system model. As the seep moves, sometimes it will encounter a fully corroded patch or pit and is able to flow through the package and sometimes it will encounter a still-intact area of the package surface and be unable to enter. (This model is mathematically equivalent to assuming that the seep is spread uniformly across the entire package area at all times, such that the flux of water able to enter the package is proportional to the fully corroded area.)

2. *Ten-percent-wet model.* In this case it is assumed that only 10 percent of the waste package surface area ever corrodes under dripping conditions throughout time. Further, the fraction of drift-seepage water that can enter the package and interact with all the non-clad waste is equal to the combined patch-pit area divided by one-tenth of the total package surface area. The conceptual model corresponding to this case is that a seep does not move with time and has a high enough flux that it splashes over 10 percent of the package area. Then the fraction of drift seepage able to enter the package is limited by the corroded area. Equivalently, it could be a narrow focussed seep that only moves back and forth across 10 percent of the surface area, sometimes encountering a fully corroded patch and sometimes not. (The scenario where the flux entering the package is not scaled by the corroded area but is equal to the entire drift seepage is discussed in Section 5.5.)

3. *One-percent-wet-model.* In this case it is assumed that only 1 percent of the waste package surface area ever corrodes under dripping conditions throughout time. This would be the conceptual scenario of an even more spatially concentrated seep that does not move with time. Furthermore, although the fraction of drift seepage water entering the package is again scaled by the combined patch-pit area (specifically, equal to this patch-pit area divided by one-hundredth of the total waste package surface area), the scaling for this scenario results in almost all the drift seepage entering the package. Specifically, one patch has an area approximately equal to 1/1,000 of the total package area. Dividing this by the 1/100 of the package area exposed to dripping yields a factor of 0.1, which would mean that one-tenth of the drift seepage enters the package and reacts with all the non-clad fuel. However, for the expected-value base case realization, the seepage collection factor mentioned in Section 4.3.2.2 is equal to 5.5, which when multiplied by the 0.1 yields 0.55. This means that about one-half of the drift seepage flows through the waste package

in this case. (Note that the collection factor is applied in all three of the above cases.)

Figure 5-18 shows the waste package failure curves for the above three scenarios. Generally, fewer waste packages fail, and waste packages fail later when a smaller percentage of the waste package surface is assumed to get wet. Figure 5-19 shows the effect on dose rate for these three dripping scenarios. The effect is generally small at all three time scales and seems to be an unimportant uncertainty as far as total system performance is concerned.

5.4.4 Sensitivity to High-pH (Concrete-Modified) Seepage Water

The base case does not include the effect of seepage water interacting with the concrete drift lining, because the assumption in the base case is that the drift lining collapses very early in the simulation. The simulations of geochemical conditions reported in Section 3.3 indicate that seepage water may have a high pH (>10) for some time, because the seepage water interacts with the concrete drift lining. The sensitivity of waste

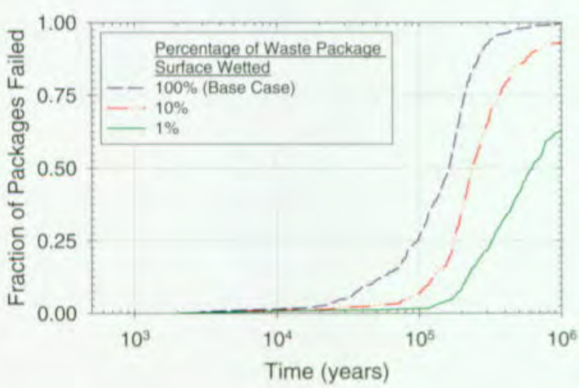

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Figure 5-18. Waste Package Degradation Curves for Different Percentages (Alternative Models) of Wetted Surface Area
a) 100 percent of surface area corrodes under dripping conditions (base case), b) 10 percent of surface area corrodes under dripping conditions, and (c) 1 percent of surface area corrodes under dripping conditions. (WP—waste package)

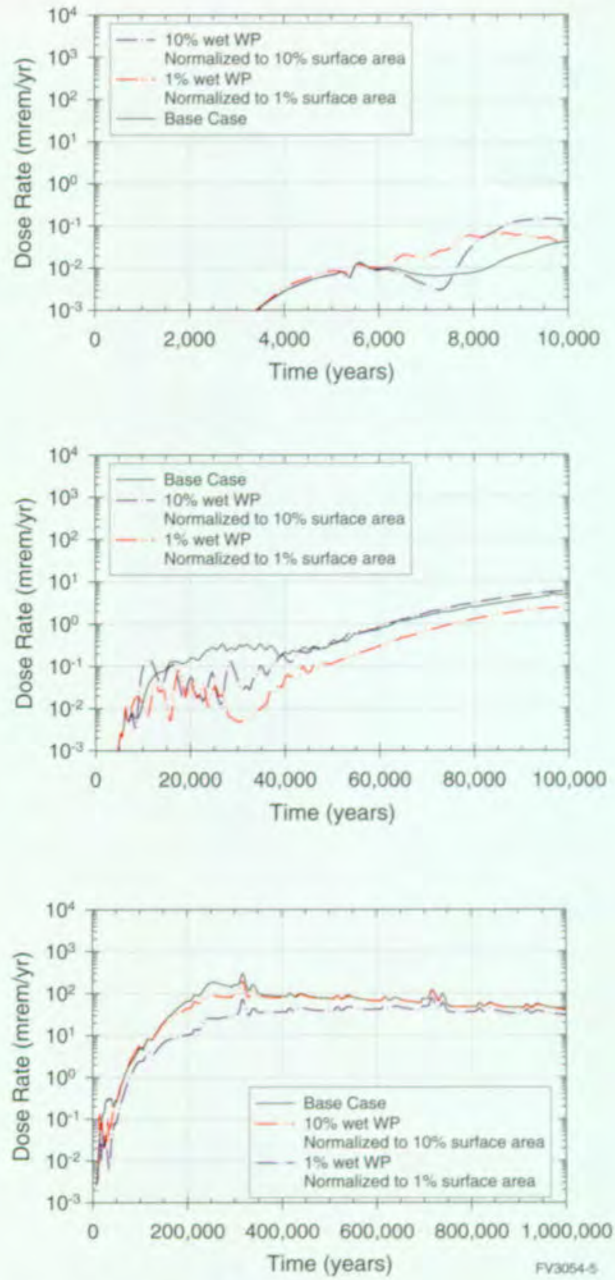

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Figure 5-19. Total Dose Rate Histories for Different Percentages (Alternative Models) of Wetted Waste Package Surface Area
a) 100 percent of surface area corrodes under dripping conditions (base case), (b) 10 percent of surface area corrodes under dripping conditions, and (c) 1 percent of surface area corrodes under dripping conditions. For all of these models the fraction of drift seepage entering the package is proportional to the fully corroded surface area divided by the fraction of the area that is wetted (i.e., either 1.0, 0.1, or 0.01) times the seepage collection factor.

package degradation to elevated pH of incoming water is shown in Figure 5-20. As discussed in Section 3.4, while the seepage water pH > 10, the carbon-steel outer barrier undergoes high-aspect ratio pitting corrosion, which fails the outer barrier shortly after corrosion starts. The early failure of the outer barrier leads to early, rapid failure of the waste package because of higher probability of localized corrosion of the inner barrier and associated early release of radionuclides. At later times, the dose from the two cases is not significantly different, because advective release rates through patch openings from general corrosion would be about the same. The one-year data for carbon steel in a concentrated J-13 water condition underway at Lawrence Livermore National Laboratory in Livermore, California, do not show such high-aspect ratio pitting corrosion (McCright 1998). The testing solution is maintained at pH 9.7. Mixed anions in the concentrated testing solution may prohibit pitting corrosion of carbon steel even in such elevated pH conditions. Thus, the current pitting model from the expert elicitation may be unrealistically conservative.

The effect of the high-pH alternative model on dose rate at 20 km (12 miles) has already been demonstrated in Figure 5-14 where it was found to have a very large effect within the first 10,000 years. But, again this is based on corrosion model for carbon steel that is probably overly conservative.

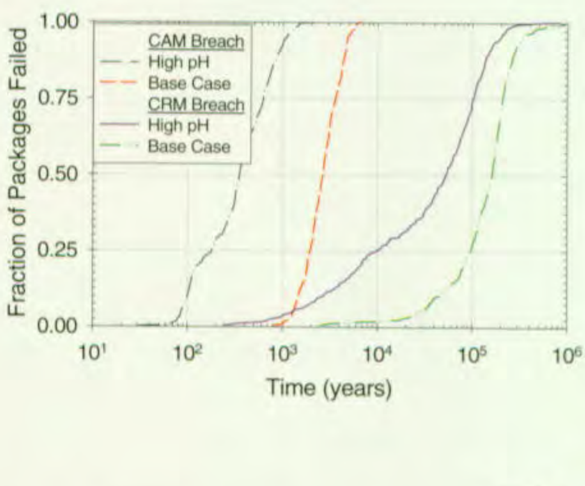

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Figure 5-20. Effect of High-pH (Concrete-Modified) Seepage Water on Waste Package Degradation
The high-pH alternative model has a major effect on waste package degradation at early times.

5.4.5 Sensitivity to Microbiologically Influenced Corrosion

Microbiologically influenced corrosion was not included in the base case in part because the potential for this corrosion of the waste packages was discounted in the recent expert elicitation (CRWMS M&O 1998b). However, the process has uncertainty and could affect long term performance. In order to assess this uncertainty on repository performance an alternative model for microbiologically influenced corrosion effects on corrosion allowance material corrosion was implemented.

In this sensitivity analysis, sustained and sufficient microbe activities are assumed to exist all the time in the postclosure repository at a level high enough to affect waste package corrosion. The Waste Package Degradation Expert Elicitation Expert Panel generally agreed that Alloy 22 would not be affected by the microbiologically influenced corrosion, thus it was assumed only the corrosion allowance material is affected by the microbiologically influenced corrosion. In addition, since drips are required for microbe growth (as indicated by the expert panel), microbiologically influenced corrosion is assumed to be operative only for the dripping case.

The sensitivity analysis was conducted using a simple microbiologically influenced corrosion-enhancement factor (or multiplication factor) to the corrosion allowance material general corrosion rate. The enhancement factor was derived from the corrosion current differences between inoculated and "sterile" samples that were measured at ambient temperature (Horn et al. 1998). This study reported an increase of the corrosion current of five- to six-fold for the inoculated samples over the sterile samples. For this analysis the enhancement factor to the corrosion allowance material general corrosion rate was assumed to have uniform distribution between 1 and 5. In the simulation, the factor was sampled randomly from the distribution and applied as a multiplier to the corrosion allowance material aqueous general corrosion rate. The simulations were conducted with the base case parameter values: NE region, spent nuclear fuel waste packages, always dripping, and 100 percent of the surface wetted by drips.

The effect of microbiologically influenced corrosion on waste package failure time is shown in Figure 5-21 and shows that early failures are significantly increased but the average failure time is not changed that much. The early exposure of the corrosion resistant material from the early corrosion allowance material breaches results in an earlier waste package breach initiation and a substantially greater number of breached waste packages. Compared to the base case results, much more extensive pit perforation is predicted for the microbiologically influenced corrosion case; however, there is not much difference in the patch perforations by corrosion resistant material general corrosion. The effect of microbiologically influenced corrosion on peak dose rate is very similar to the effect of concrete-modified water (Figure 5-14) and is shown in Figure 5-22 for the three periods. The main effect on peak dose rate is the large (over two orders of magnitude) increase in the first 10,000 years due to more aggressive degradation of the carbons steel outer barrier.

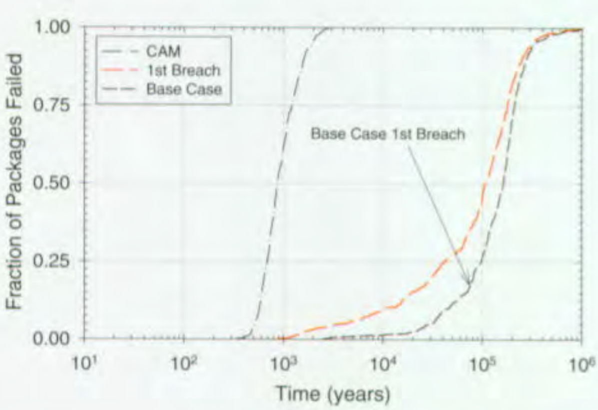

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Figure 5-21. Effect of Microbiologically Influenced Corrosion on Waste Package Degradation
The microbiologically influenced corrosion alternative model has a major effect on waste package degradation at early times. (CAM—corrosion allowance material)

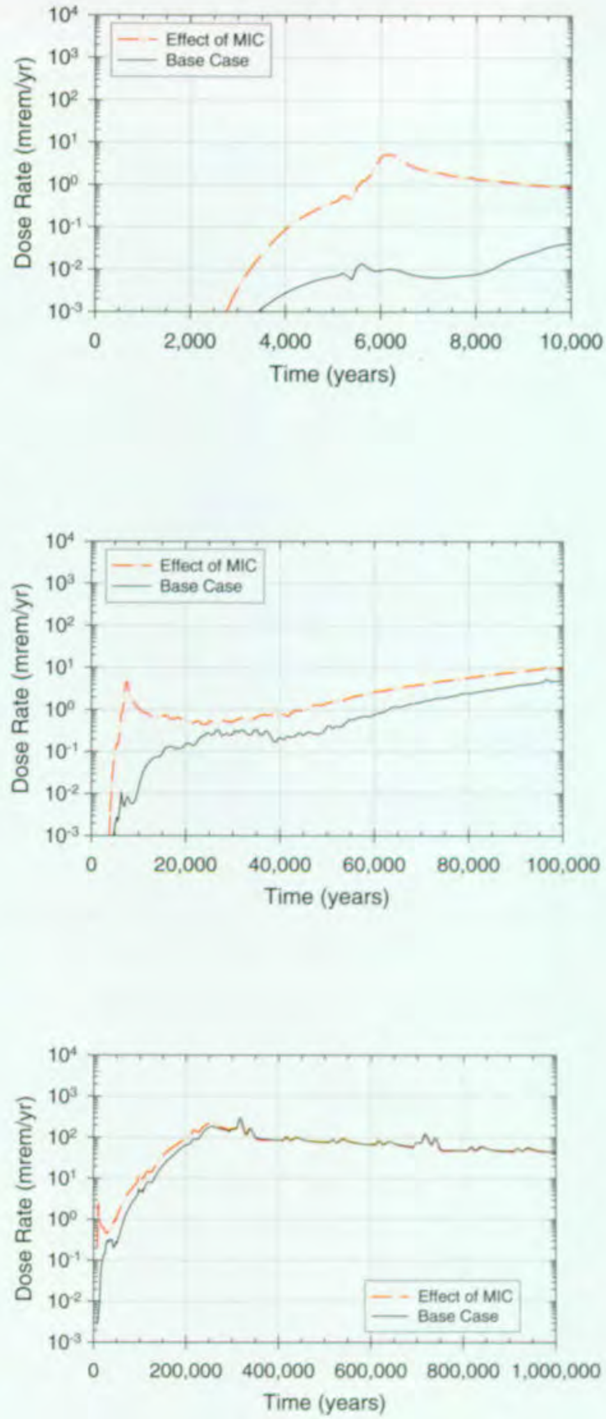

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Figure 5-22. Total Dose Rate Histories for Microbiologically Influenced Corrosion Waste Package Degradation Versus the Base Case
The effect is similar to the high-pH alternative model. (MIC—microbiologically influenced corrosion)

5.4.6 Sensitivity to Juvenile Failures

Another uncertainty in the waste package degradation model is the number of early, or juvenile, failures that may occur in the repository. The effect of such failures is also uncertain. Failure could be a merely a slight crack in the waste package that provides a little opening for radionuclide transport out of the waste package or a more general failure such as that caused by corrosion patch failure. The expected-value base case assumes that a single waste package fails at 1,000 years in a dripping zone. The failure is assumed to be one general corrosion patch (about 300 cm^2) that allows advective release to begin at the failure time. This juvenile-failure model implemented in the base case has an effect on the early-time dose, as expected but does not affect the long-term peak dose. This is demonstrated in Figure 5-23 which compares the base case model to a case (alternative model) where there are no juvenile failures. The effect of juvenile failures is to cause earlier appearance of radionuclides at the 20-km (12-mile) boundary, by about two to three thousand years. The 95th percentile, juvenile-failure scenario in the TSPA-VA base case, which assumes that eight waste packages fail at 1,000 years, is also shown in Figure 5-23. It indicates that the early-time peak dose rate increases by about eight times but there is no affect on the long-term dose.

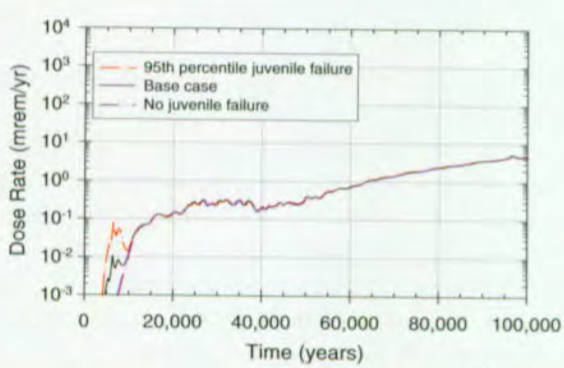

Figure 5-23. Sensitivity of Dose to Juvenile Failure of Waste Packages
Early-time dose is linearly increased by additional juvenile waste package failures.

5.4.7 Sensitivity to Corrosion Patch Size

In the waste package degradation model, the size of the corrosion patches used in the stochastic waste package degradation to model general corrosion of Alloy 22 is also uncertain. The individual patches are intended to represent a minimum area on the waste package, that has a uniform, local exposure condition and thereby uniform general corrosion rate within a patch. The base case corrosion patch size is assumed to be 310 cm^2 (48 $in.^2$), close to the surface area of the standard atmospheric corrosion test coupon, or sample. The total number of corrosion patches on a single waste package for the base case is assumed to be 964.

If the patch size is increased by a factor of 10, there are early failures because there are more pits per patch, which leads to early pit failure. But the average waste package failure time for dripping conditions is later than the base case, and the number of waste packages failing in 1 million years is reduced by about 10 percent (Figure 5-24). If the corrosion patch size is decreased by a factor of 10, giving 9,641 corrosion patches per waste package, average waste package failure occurs earlier for dripping conditions than it does in the base case.

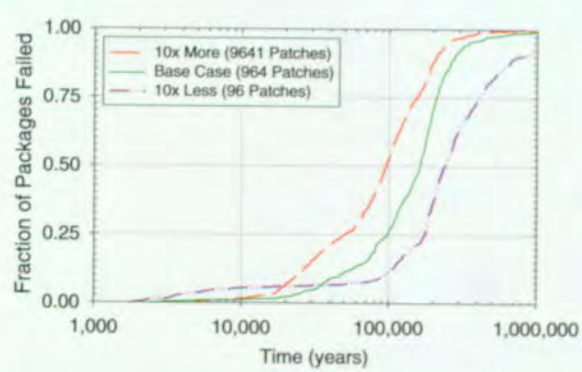

Figure 5-24. Sensitivity of Waste Package Degradation to Corrosion Patch Size

When the alternative-model corrosion-patch-size curves are implemented into the total system analysis, the dose rate results differ in the first part of the time period but not at late time (see Figure 5-25). Some of the larger patches fail earlier, and they have larger area for seepage into and release from the waste package. The smaller patch case fails earlier in general than the other two cases, but there is less seepage into the waste package and smaller openings for radionuclide release from the waste package, so the dose rates are similar to the base case.

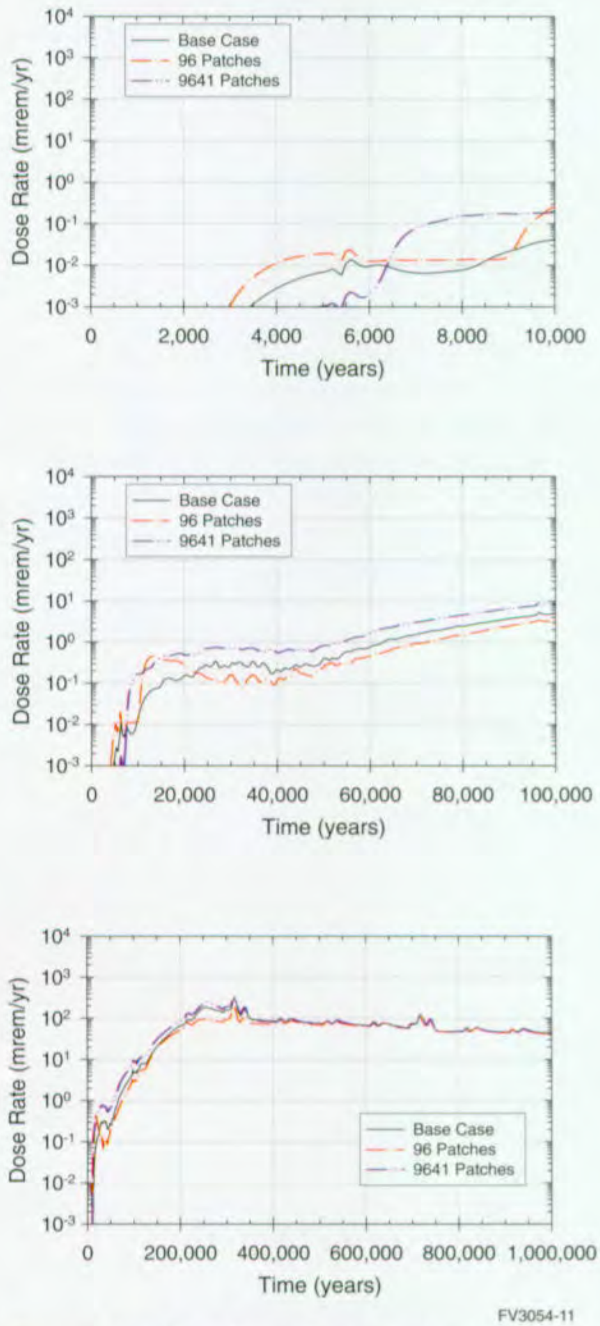

Figure 5-25. Total Dose Rate Histories for Different Sizes of Corrosion Patch Failure
There is little effect on dose rate for these alternative models.

5.5 WASTE FORM ALTERATION, RADIONUCLIDE MOBILIZATION, AND TRANSPORT THROUGH THE ENGINEERED BARRIER SYSTEM

The key uncertainties in waste form degradation and mobilization and transport of radionuclides through the engineered barrier system are discussed in this section. These uncertainties were identified in Section 3.5 and Section 4.3. Evaluations of the importance of seepage through the waste package, integrity of the spent nuclear fuel cladding, secondary phase retention of neptunium, neptunium solubility, the formation and transport of radionuclide-bearing colloids, and radionuclide retardation in the invert are included in this section.

Additional sensitivity analyses comparing the defense spent nuclear fuel with an equivalent amount of commercial spent nuclear fuel, and the plutonium waste forms with an equivalent amount of commercial spent nuclear fuel and high-level radioactive waste are also presented in this section.

5.5.1 Sensitivity to Seepage into the Waste Package

The flux into the waste package is an important parameter affecting release from the waste package, and little is known experimentally about how much of the seeping water will flow into the waste package. Some of the water seeping into a drift and dripping onto a package is expected to flow around the package without entering it. The releases from the waste package are directly proportional to the amount of seepage that flows through the waste package. As discussed in Section 5.4.3, a base case assumption is that the seepage into the waste package is directly proportional to the size of the corroded openings in the waste package (divided by the total package surface area) and multiplied by a "seepage collection factor" that varies uniformly between

1 and 10, with a mean of 5.5. A sensitivity evaluation of the importance of this parameter was conducted using the 5th and 95th percentile values for the seepage collection factor. The results are shown in Figure 5-26. The effect is small at all times but is most noticeable between 50,000 and 250,000 years when the dose is dominated by solubility-limited releases of neptunium-237. During this time a change in seepage produces a change in the dose that is linear with the increase or decrease in the seepage fraction into the waste package. However, at later times the effect is masked by the cladding model, which is not exposing enough additional waste to reach the solubility limit of the highest dose contributor, neptunium.

As a further investigation into the effect of seepage through the package on total dose rate, an alternative model was examined which is similar to Case 2 in Section 5.4.3. In this case, 10 percent of the package area is corroded under dripping conditions for all time. However, in contrast to the aforementioned case (which has seepage into the package scaled by the corroded patch-pit area), in the alternative model all of the drift seepage is assumed to flow through the package. These two seepage models are compared to the base case seepage model in Figure 5-27. As with the seepage collection factor, the effect is relatively minor over all periods.[1]

5.5.2 Sensitivity to Cladding Degradation

Commercial spent fuel cladding degradation through time is another uncertain process. The comparison of the total dose rate arising from the expected-value cladding degradation is compared to the total dose rate for the 5th and 95th percentiles of the cladding-failure distribution in Figure 5-28 for the TSPA-VA base case. The base case cladding degradation distribution is defined with an expected value of 31 percent, a maximum of 47 percent, and a minimum of 1 percent for the amount of exposed surface area of the commercial spent nuclear fuel at 1 million years. The base case

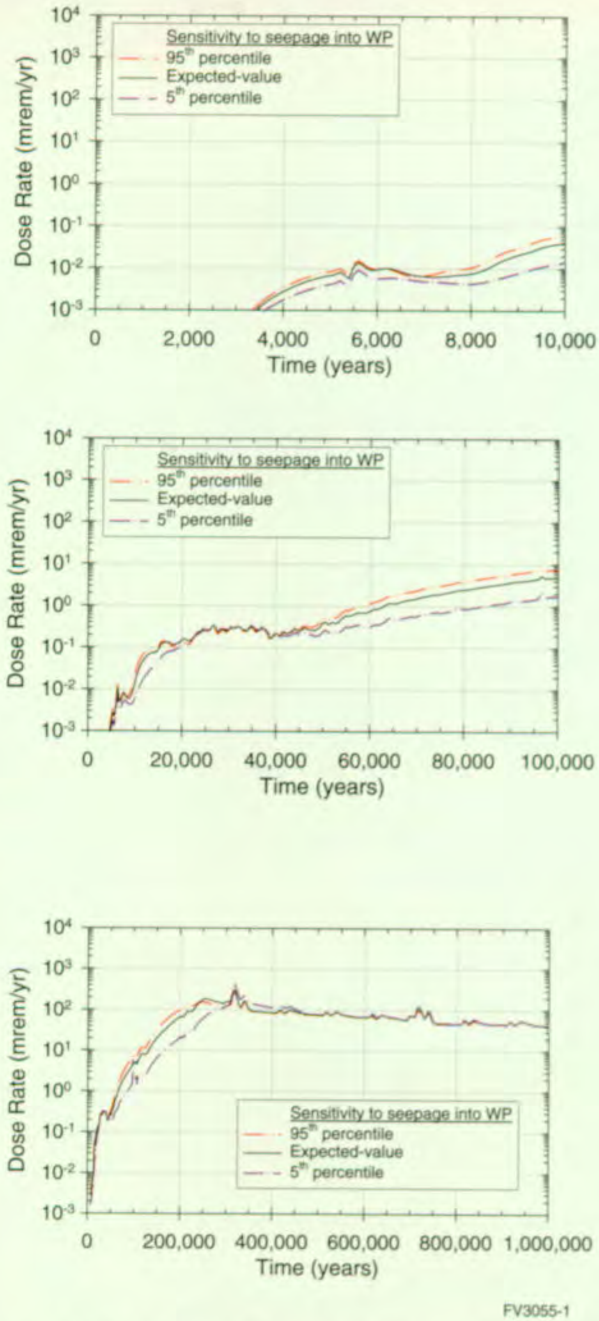

Figure 5-26. Effect of Seepage Collection Factor on Total Dose Rate
Comparison of base case expected-value seepage collection factor with the base case 5th and 95th percentile values. This factor partially determines the amount of liquid flowing through the corroded areas of the waste packages. (WP—waste package)

[1] For all three of these cases the water flowing through the package is assumed to react with all of the exposed fuel (that is, all fuel for which the cladding has degrading or which had no cladding at emplacement).

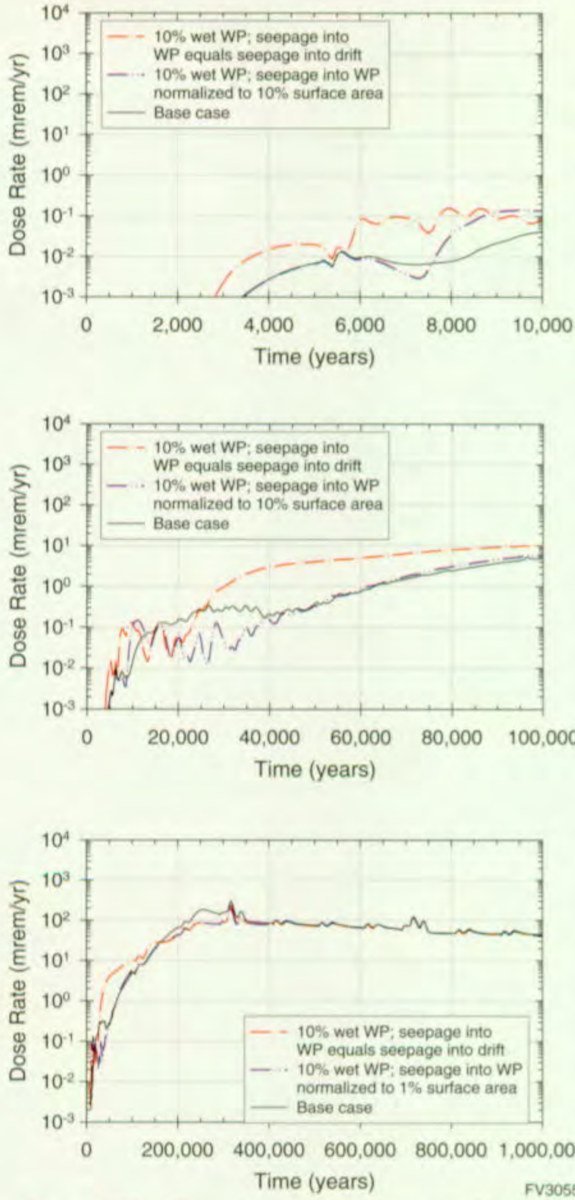

Figure 5-27. Effect of Alternative Seepage Model on Total Dose Rate

a) Base case model assumes that the entire waste package surface area corrodes under dripping conditions but the seepage flux through the waste package is proportional to the corroded area divided by the total surface area; b) "10 percent wet waste package model combined with seepage into package normalized by 10 percent of surface area" means that 10 percent of the waste package surface corrodes under dripping conditions and that the seepage flux through the waste package is proportional to the corroded area divided by one-tenth of the total surface area; (c) "10 percent wet waste package model combined with drift seepage equals seepage into package" means that 10 percent of the waste package surface corrodes under dripping conditions and that the seepage flux through the waste package is equal to the drift seepage flux.

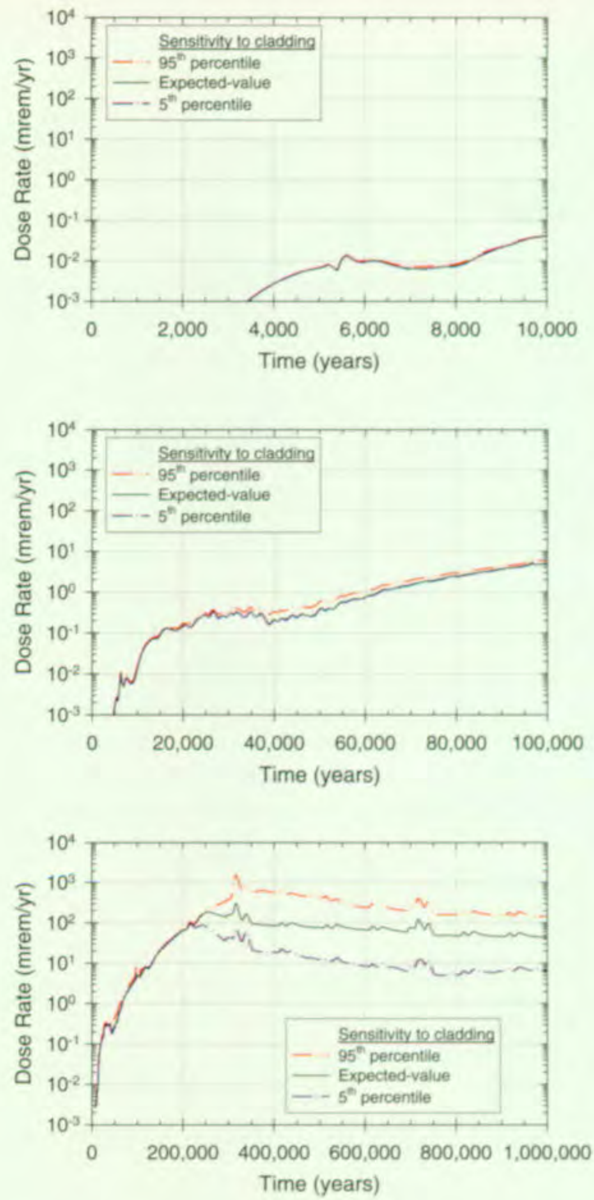

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Figure 5-28. Effect of Cladding Failure Fraction on Total Dose Rate

Comparison of base-case expected-value cladding failure fraction with the base-case 5th and 95th percentile values.

model does not have a significant range for the surface area until 100,000 years after waste package failure, when mechanical and corrosion degradation processes begin to increase. The delay in initiation of these processes is one reason why the spread, or variance, in the dose is not observed until after 250,000 years. The other reason is that releases become cladding-degradation-rate-limited after 250,000 years, as explained in Section 4.2. Prior to that time, between 50,000 years and 250,000 years, the dose rate is dominated by solubility-limited neptunium releases, implying that within the range of the base case cladding model, the variations in cladding degradation would not have an effect on dose rate prior to 250,000 years.

The base case cladding model does not have a very wide uncertainty range, so the parameter does not show up in section 4.3 as a top rank-regression parameter. For example, the extreme of no cladding credit is not part of the base case model uncertainty range. Thus, additional analyses are conducted to evaluate alternative conceptual models of cladding failure. The alternative models range from immediate cladding failure to total failure of the cladding at either 100,000 years or 1 million years. These are compared to the base case model in Figure 5-29. The most significant difference in the models up through 100,000 years is that the no-cladding model has significantly higher releases. This is primarily due to the increased release of technetium-99 at early times from the fully exposed spent fuel rods. After the technetium is exhausted, the releases are dominated by solubility-limited neptunium releases up till about 250,000 years, which is the reason that all models yield approximately the same dose rates. Beyond 250,000 years, the no-cladding model and 100,000-year-failure model yield similar results, as might be expected, with their predicted dose rates being higher than the other two models (the base case and 1-million-year models). At close to 1 million years the fuel inventory from these two models is starting to be exhausted, so their predicted dose rates fall below the 1-million-year-failure model. The other two models, the base case and 1-million-year-total-failure have similarly shaped dose rate curves, but

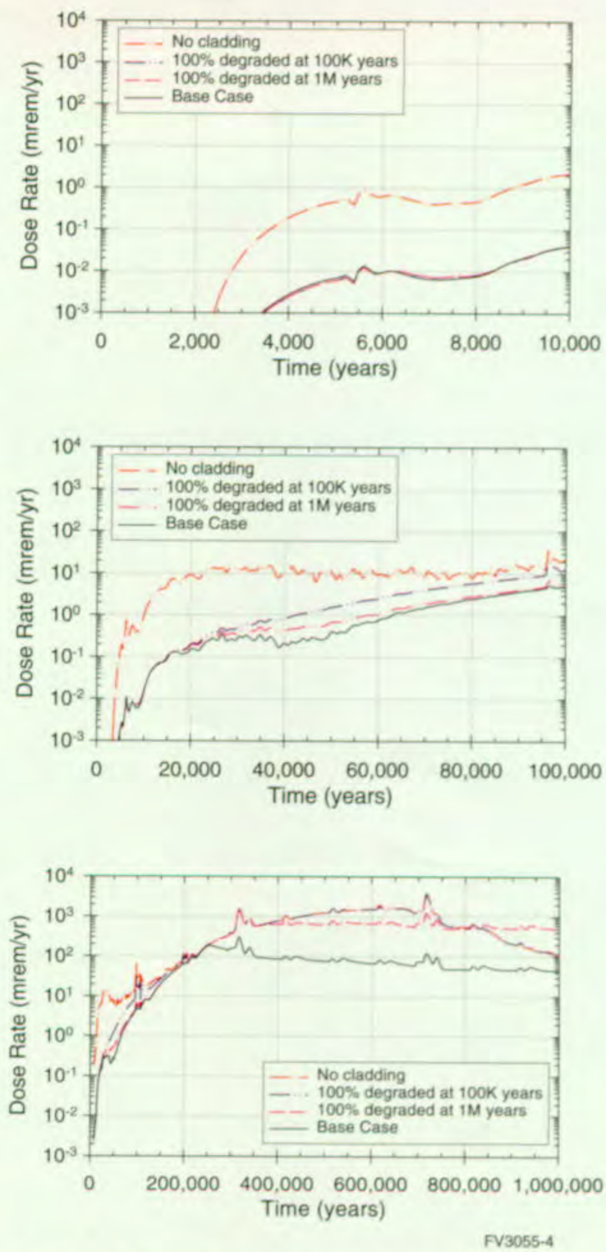

Figure 5-29. Effect of Alternative Models for Cladding Failure on Total Dose Rate
One model assumes no cladding credit at any time. The "100 percent degraded at 100,000 years" model assumes a linear relationship between log (clad failure fraction) and log (time) with 1 percent failure at 10,000 years and 100 percent failure at 100,000 years. The "100 percent degraded at 1 million years" model assumes a linear relationship between log (clad failure fraction) and log (time) with 1 percent failure at 10,000 years and 100 percent failure at 1 million years. These three models are compared to the base case model, which has several different failure modes as shown in Figure 3-54.

separated by a constant factor. This behavior is not unexpected, since the cladding failure curves for the two models are about the same, only differing with respect to the endpoint value at 1 million years, which is 31 percent failure for the base case and 100 percent failure for the 1-million-year-total-failure model.

Because there might be more uncertainty in the cladding degradation than that included in the base case and because the base case cladding model did not show up as an important rank-regression parameter in Figure 4-34, an alternative cladding model was investigated besides those discussed in the previous paragraph. The alternative model assumes a much wider range on the cladding uncertainty by assuming that the percentage of failed cladding in a waste package is log-uniformly distributed between 0.01 and 1.0, with a mean of 0.215. This percentage is assumed to be failed instantaneously when the waste package inner barrier is breached. To evaluate the differences between the base case and modified cladding models, probabilistic simulations were conducted for each of the three simulation periods. The distributions of peak dose rates to an individual located at 20 km (12 miles) from the repository for the base case and the modified cladding case are compared in Figure 5-30. The case with the modified cladding model gives higher peak dose rates than the base case in all three time. This increase in peak dose rates is primarily due to the fact that the fraction of fuel exposed because of cladding failure is available immediately for dissolution after waste package failure rather than gradually increasing with time as in the base case. Note that differences in dose rates tend to be more pronounced on the low probability regions of the curves, except in the 10,000-year period where differences tend to be significant for the entire range of doses. Figure 5-31 is a comparison of the most important model parameters for total system performance, similar to Figure 4-34, for the modified cladding case. Based on the R^2-loss, the modified cladding case indicates that cladding degradation is a key parameter in the 10,000-year and 1-million-year periods. However, it is not a key parameter in the 100,000-year period because of the dominance of solubility-limited neptunium releases, which

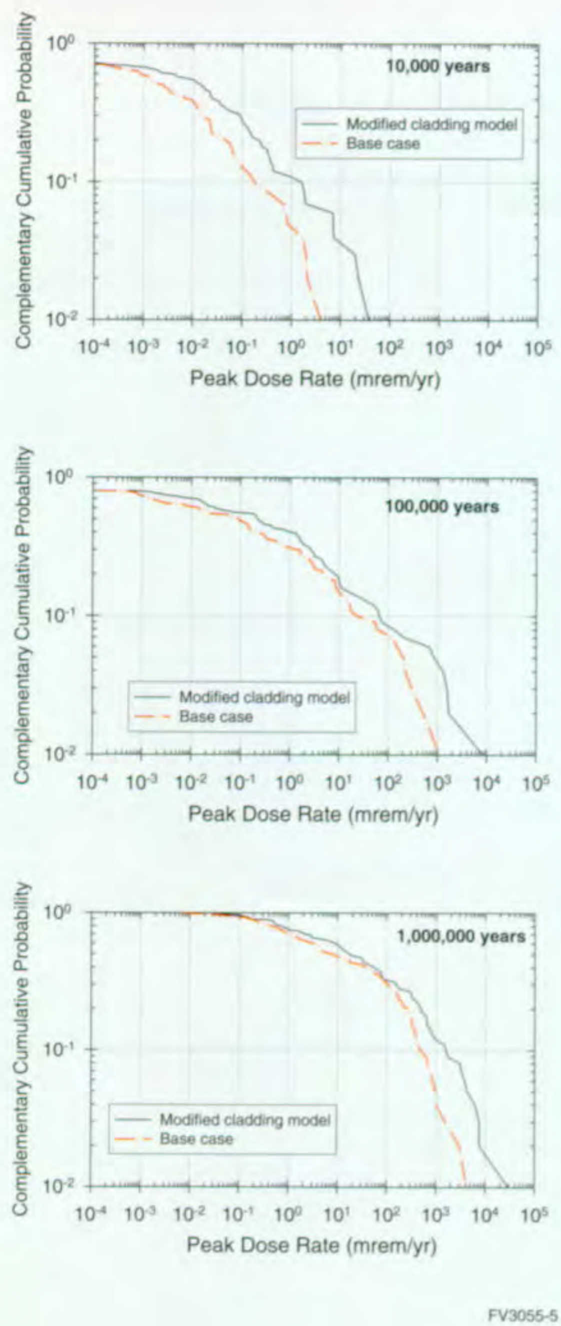

Figure 5-30. Effect of an Alternative Cladding Model on 10,000-Year, 100,000-Year, and 1 Million Year Peak Dose Rate over Multiple Realizations
The alternative model assumes a wider uncertainty band for cladding failure fraction than used in the base case (see Figure 3-54). The fraction is assumed to be log-uniform from 0.01 to 1.0 and it is assumed that the cladding is failed instantaneously when the waste package fails.

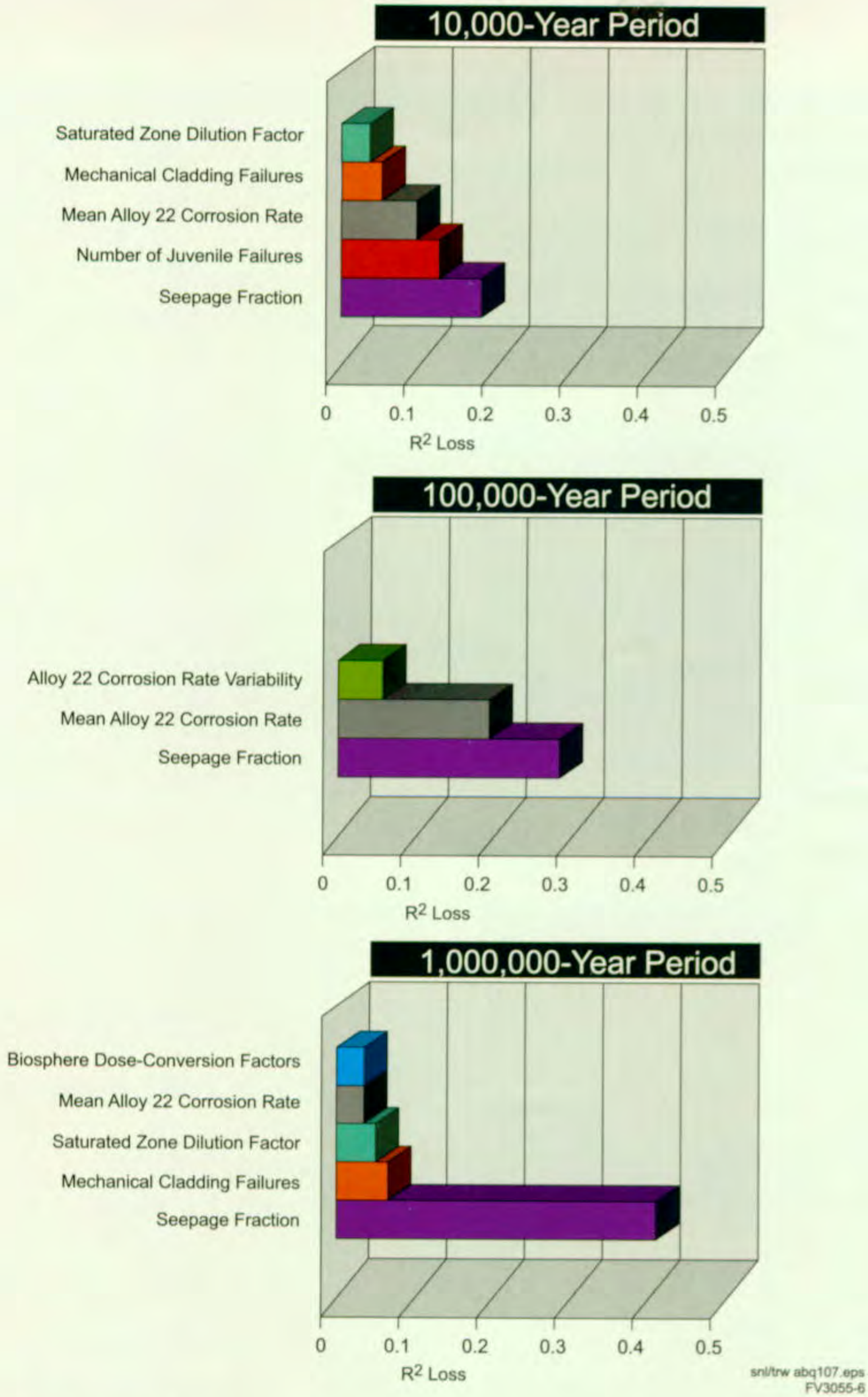

Figure 5-31. Most Important Parameters for the Modified Cladding Model Case
Results are from stepwise regression analysis for three time periods. These charts show the relative importance of various parameters to the calculated uncertainty in dose rate for the three time periods. Importance of an individual parameter is shown by R^2-loss, the reduction of goodness of the regression fit.

prevents the cladding from having much of an effect.

5.5.3 Sensitivity to Dissolution Rate and Secondary-Phase Retention of Neptunium

The dissolution rates of commercial spent nuclear fuel and high-level glass waste are potentially important parameters for repository performance, although they did not appear in the R^2-loss ranking in Figure 4-34. To examine their possible effect on dose rate, the single realizations of their 5th and 95th percentile values are compared to the expected-value base case in Figure 5-32. The uncertainty ranges in these parameters are derived from experimental studies described in Chapter 6 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i). Clearly for the uncertainty ranges considered, there is almost no effect on dose rate.

The uncertainty in dissolution rate in the base case did not consider a potentially important phenomenon: the retention (re-precipitation) of neptunium in secondary phases after initial dissolution. This effect is observed in laboratory experiments (Finn et al. 1997) The effect of such retention is analyzed as an alternative model described in Section 3.5 and in Chapter 6 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i). The retention effect is gained by reducing the solubility of neptunium by a factor of 45 to approximate the results of reactive transport modeling discussed in Section 3.5.2.5. The dose rate resulting from this alternative model is shown for all three time frames in Figure 5-33. At early times, prior to about 50,000 years, the alternative model has no effect because doses are dominated by technetium. The greatest effect is noticed at about 200,000 years when neptunium comprises more than 99 percent of the total dose in the base case (see Figure 4-12). At this time, the total dose from the alternative neptunium solubility model is about a factor of 25 lower than the base case total dose rate. If the total dose rate at this time in the alternative model were

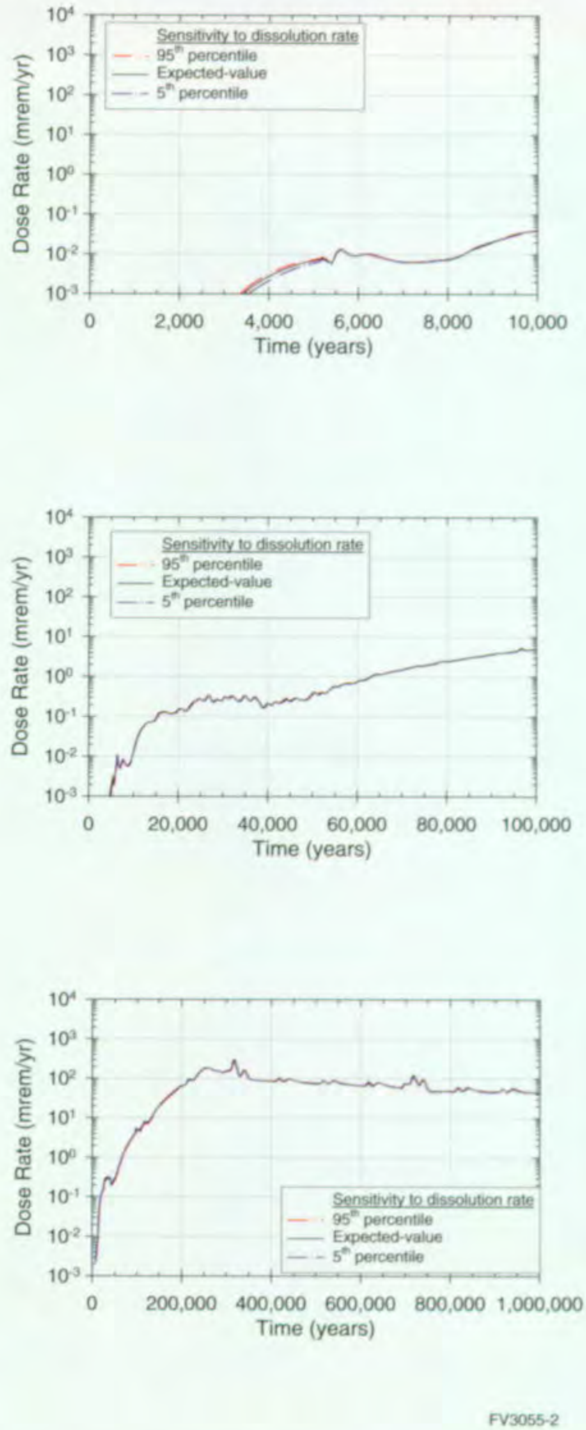

Figure 5-32. Effect of Commercial-Spent-Fuel and High-Level-Glass Dissolution Rates on Total Dose Rate
Comparison of base case expected-value dissolution rates with the base case 5th and 95th percentile values.

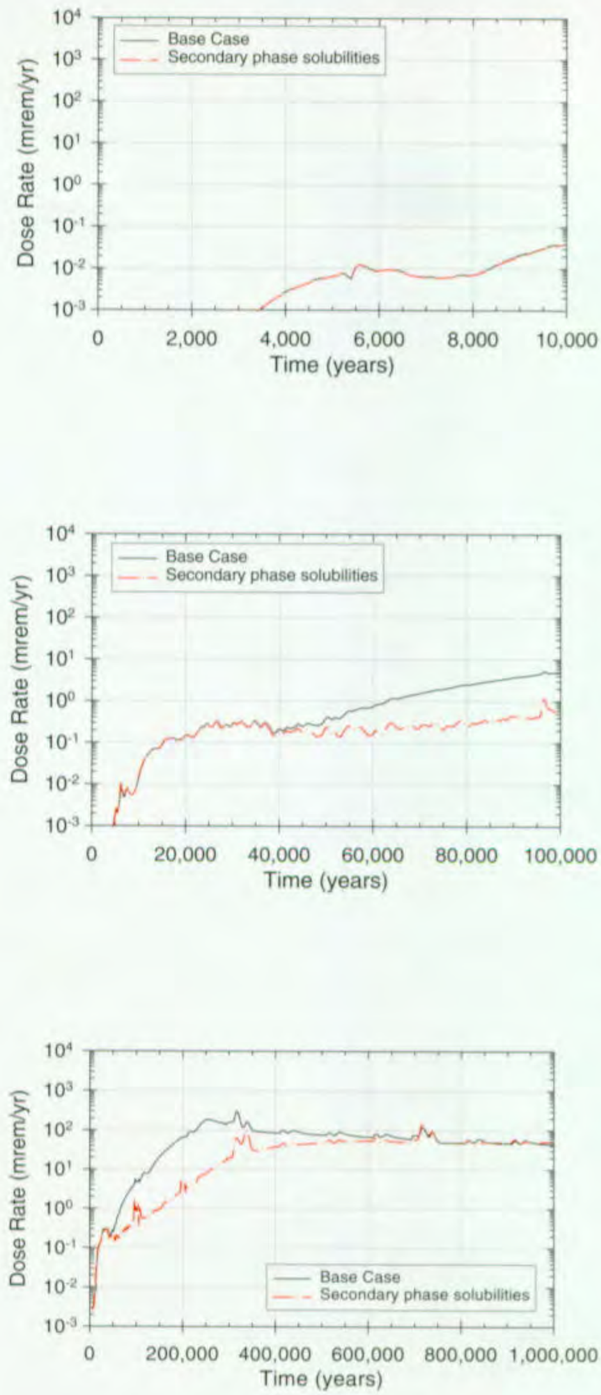

Figure 5-33. Sensitivity of Total Dose Rate to Reprecipitation of Neptunium in Secondary Mineral Phases
These phases have lower solubility than the primary mineral phases. This effect is implemented by dividing the base case neptunium solubility by 45.

composed almost solely of neptunium, as it is in the base case, then the difference between the two models would be exactly a factor of 45 because of the difference in neptunium solubility. However, in the alternative model the neptunium dose is low enough at this time that some of the other radionuclides, such as plutonium-242, uranium-234, and technetium-99 (see Figure 4-12) have a non-negligible contribution to dose. At a later times (> 250,000 years), the effect of the alternative neptunium model is not as significant because neptunium is no longer solubility limited (all waste packages have failed) and the cladding degradation rate limits the surface area available for dissolution and release.

The retention of neptunium as a secondary phase, while a potentially important effect in repository performance, is still being evaluated in laboratory testing and corresponding reactive transport modeling. Further analyses are necessary before including such effects in the base case TSPA.

5.5.4 Sensitivity of Dose to Neptunium Solubility

The solubility of neptunium is uncertain to within at least a three-order of magnitude range as shown by the shaded region in Figure 5-34. This shaded region was the range of neptunium solubility used in the TSPA-VA base case. It does not encompass all experimental measurements, as shown in the figure, but the measurements outside of this range are thought to be unrepresentative of any potential repository conditions, as described briefly in Section 3.5.1.8 and in much more detail in Chapter 6 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i). Neptunium solubility did not appear as one of the most important rank-regression parameters in Figure 4-34, however, it did appear as an important parameter in Figure 4-40 for the 100,000-year time span, when the seepage and corrosion-rate parameters were held at their mean values. In other words it was in the second-tier of key R^2-loss parameters. However, it only appears in the second tier during the 100,000-year time frame because this is the time frame when

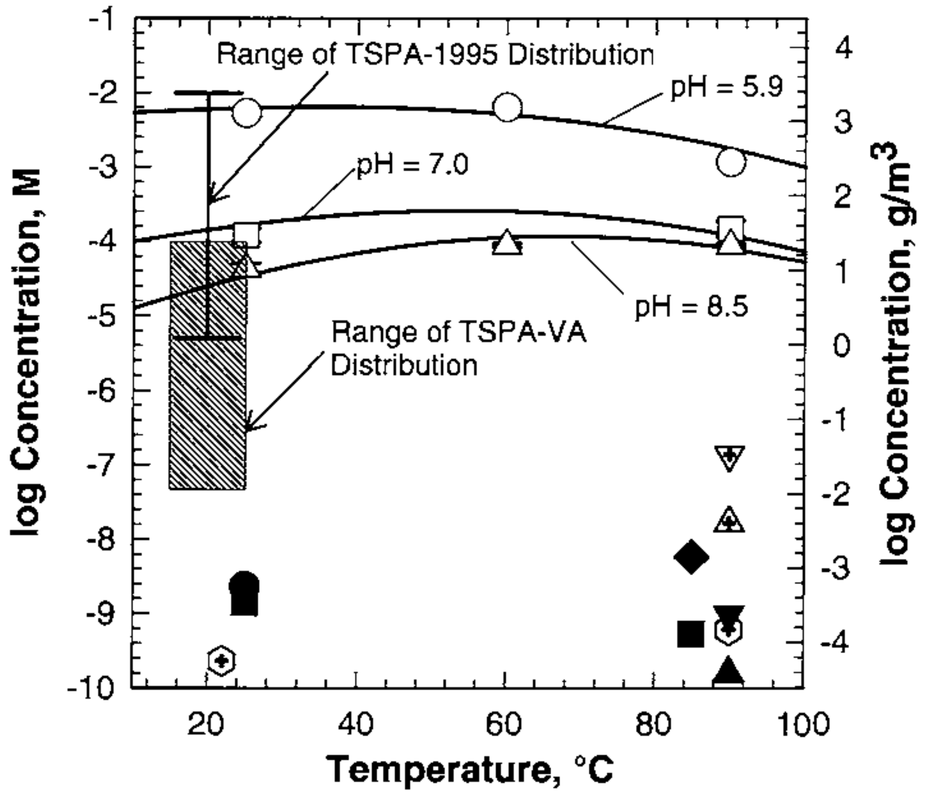

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Figure 5-34. TSPA-VA Base Case Neptunium Solubility Distribution Compared to TSPA-1995 Distribution and to Eleven Different Precipitation/Dissolution Experiments
The open symbols at the top of the diagram are precipitation (oversaturation) experiments. The filled symbols at the bottom are dissolution studies, including batch, flow-through, and drip experiments. See Chapter 6 of CRWMS M&O 1998i for more detail.

solubility-limited neptunium tends to dominate doses. This is born out in the total-dose time-history comparison shown in Figure 5-35, comparing the expected-value neptunium solubility case to the dose rate arising from the 5th and 95th percentile values from the base case neptunium solubility range.[2] As with the secondary-phase model described by Figure 5-33, the greatest effect on dose for the base case model is in the time frame of 50,000 to 250,000 years when the total dose is dominated by solubility-limited neptunium releases. At later times the cladding degradation rate controls total dose rate.

5.5.5 Sensitivity to Formation and Transport of Radionuclide-Bearing Colloids

Uncertainty in colloid formation and transport parameters is quite high in the TSPA-VA base case (see Tables 3-16 and 3-18), particularly in far-field transport, because of a lack of experimental or field-scale evidence to verify the models. Nevertheless, for the models and parameters used in the base case, colloid model parameters do not show up in the key rank-regression parameters in Figures 4-34 or 4-40, even though Figure 4-29 indicates that plutonium dose rate is the most significant contributor to total dose rate for 2 percent of the 100,000-year multiple realizations

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[2] Neptunium solubility was modeled as a log-beta distribution and the 5th, 95th, and expected values were chosen for the log of solubility, not for the solubility itself.

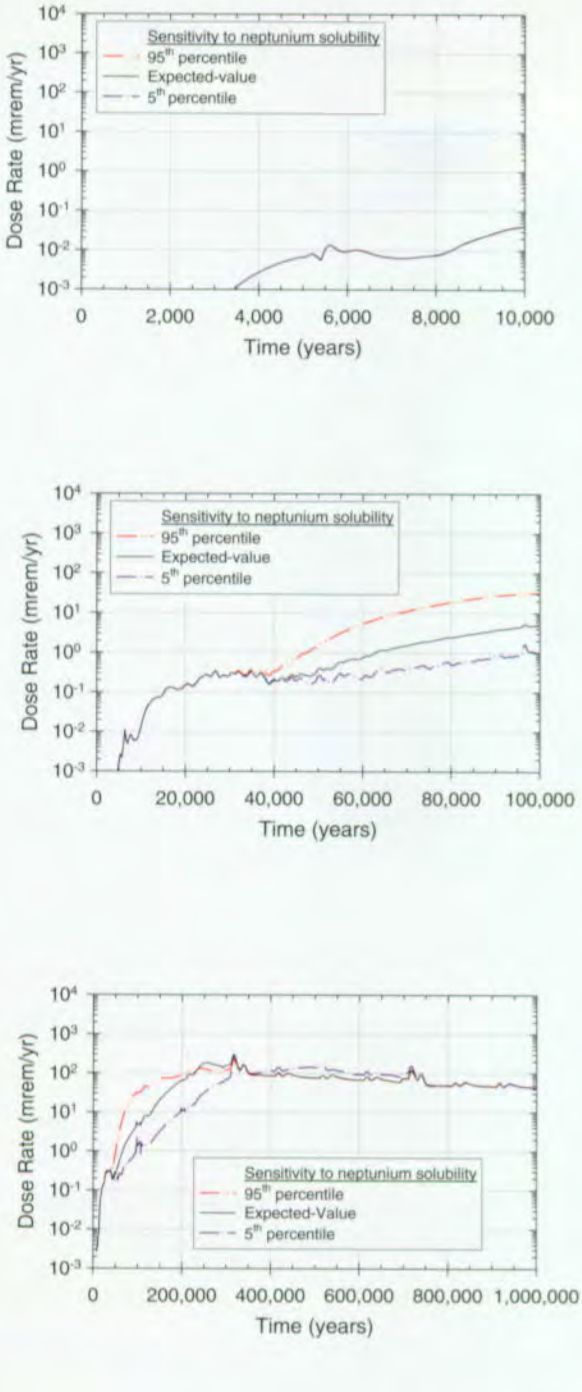


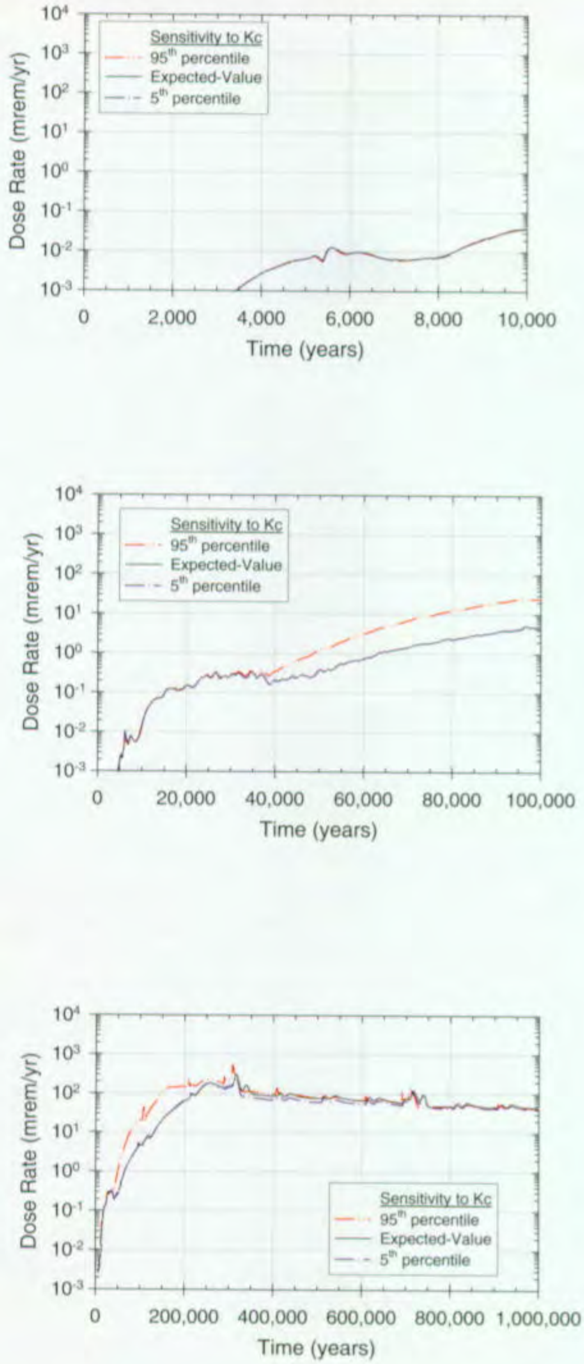
Figure 5-35. Effect of Neptunium Solubility on Total Dose Rate
Comparison of base case expected-value solubility with the base case 5th and 95th percentile values.

and 8 percent of the 1-million-year multiple realizations. However, in examining the sensitivity to the 95th percentile value of the K_c range ($K_c = 5.0$) in the base case, it is evident that there is an effect on dose rate during the 50,000-year to 250,000-year time period, as indicated in Figure 5-36. The K_c parameter is high enough at this percentile to allow a significant fraction of plutonium to be transported as reversibly sorbing on colloids. There is no difference in this figure between the 5th percentile value for K_c and the expected-value because the expected value is already too low to allow a significant fraction of aqueous plutonium to be transported in colloidal form.

5.5.6 Sensitivity to Transport in the Engineered Barrier System

The TSPA-VA base case evaluation includes radionuclide retardation in the concrete invert (i.e., the structure that supports the waste package) for neptunium, plutonium, uranium, and protactinium. There is uncertainty in the appropriate values for these retardation factors (as indicated by the ranges given in Section 4.1.10) and about what will happen to the retardation capability of the invert over a long time as the system degrades. To evaluate the effect on dose rate of the retardation capability of the invert, a sensitivity case was conducted (Figure 5-37) using the 5th and 95th percentiles of these K_d ranges. The results indicate that for early times (< 35,000 years) when the dose is dominated by technetium release (no retardation), there is no change. At later times when the dose is primarily a result of neptunium release, the difference in total dose rate mimics the change in neptunium dose rate, but the increase is less than a factor of two. At much later times (> 250,000 years) cladding degradation rate eliminates any difference between the cases.

An additional sensitivity analysis was carried out to examine retardation in the invert. This is an alternative model wherein all the distribution coefficients (K_ds) in the invert are set to zero (i.e., no retardation in the invert). Figure 5-38 presents the results of these analyses, which are about the same as the 5th percentile case in Figure 5-37. In



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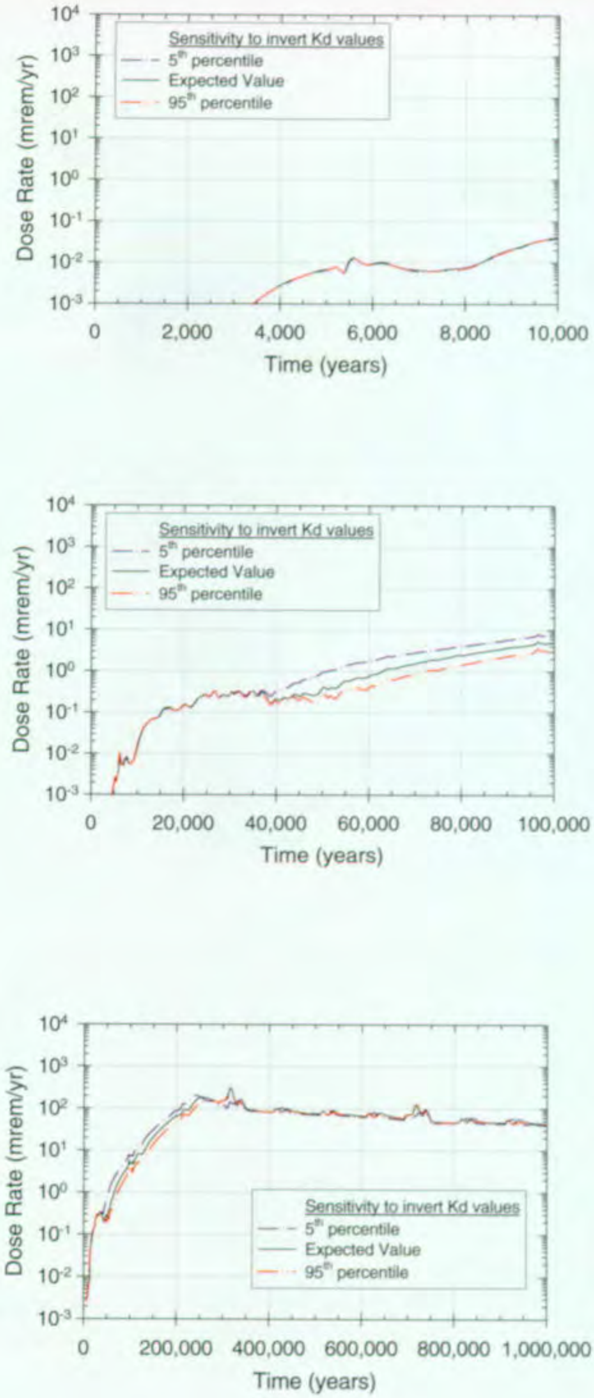

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Figure 5-36. Effect of Colloid Distribution Coefficient on Total Dose Rate
Comparison of base case expected-value K_c with the base case 5th and 95th percentile values.

Figure 5-37. Effect of Radionuclide Distribution Coefficients in the Concrete Invert on Total Dose Rate at 20 km (12 miles)
Comparison of base case expected-value K_ds with the base case 5th and 95th percentile values. The primary effect is from the neptunium K_d.

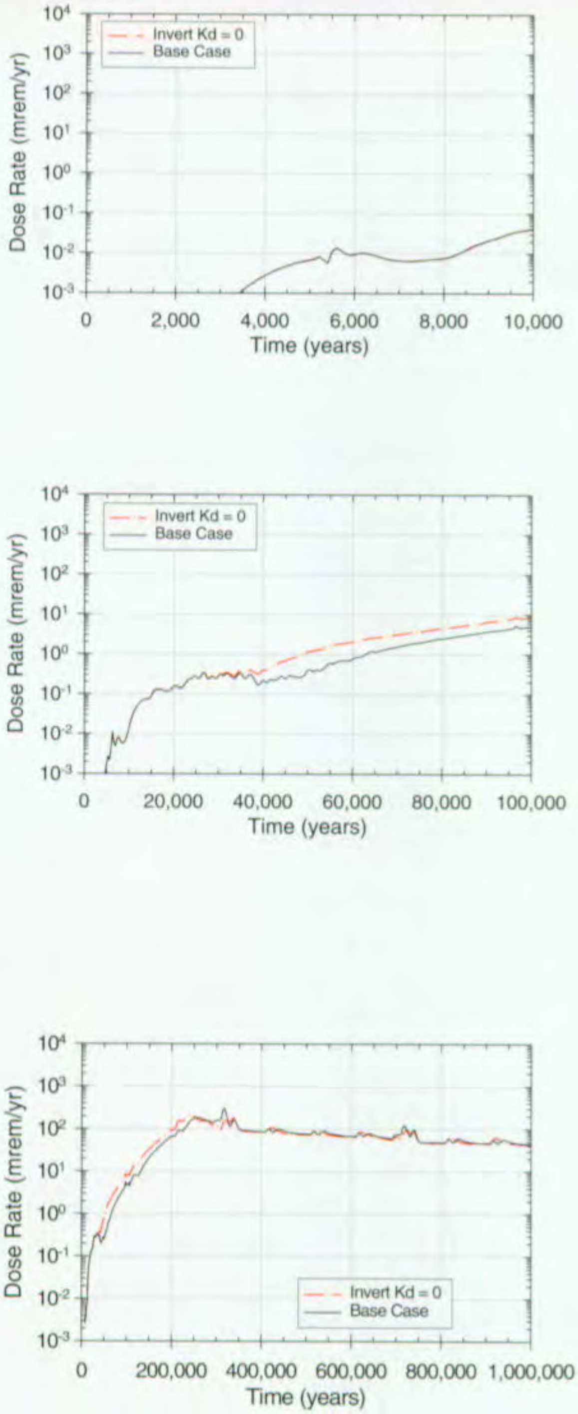

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Figure 5-38. Sensitivity of Total Dose Rate to an Alternative Engineered Barrier System Transport Model that Assumes the Radionuclide K_ds in the Invert are Equal to Zero
In the top figure, the two curves overlay.

summary, the retardation in the invert appears to have little significance to total dose, and this is in spite of the fact that the expected-value K_d in the base case is quite high.

5.5.7 Comparison of the Surrogate U.S. Department of Energy Spent Nuclear Fuel to DOE Spent Nuclear Fuel Total

The base case analyses reported in Sections 3.5, 4.2, and 4.3 used a surrogate inventory for DOE spent nuclear fuel rather than explicitly modeling each of more than 250 types of the fuel. The surrogate was based on the key dose contributors from the 15 categories of DOE spent nuclear fuel that were determined in the original inventory. To determine the key dose contributors, each individual DOE spent nuclear fuel category was analyzed by placing it in the environment of the base case, one category at a time, and calculating the expected dose to humans located 20 km (12 miles) downgradient from the repository. The 2,333 metric tons of heavy metal (MTHM) of surrogate have a radionuclide inventory that is a weighted average (on an MTHM basis) of the radionuclide inventories of Categories 1, 4, 5, 6, 8, and 11. These six categories were found to contribute significantly to the dose from all DOE spent nuclear fuel (Duguid et al. 1997, pp. 4-1 to 4-10, 6-2 to 6-3). Because the majority of the DOE spent nuclear fuel is metallic, the metallic dissolution model was assumed for the surrogate. In addition to these six categories, Categories 7 and 16 were also found, in the current analyses, to contribute significantly to the dose from all DOE spent nuclear fuel (CRWMS M&O 1998i). In the base case, these categories are represented by the surrogate spent nuclear fuel, which would yield conservative results. (For more detail on this analysis, see Chapter 6 of *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* [CRWMS M&O 1998i].)

The dose rate from each DOE spent nuclear fuel category is presented in Figure 5-39 over the first 100,000 years after repository closure. Figure 5-39 also compares the total dose from the surrogate DOE spent nuclear fuel to the total DOE spent

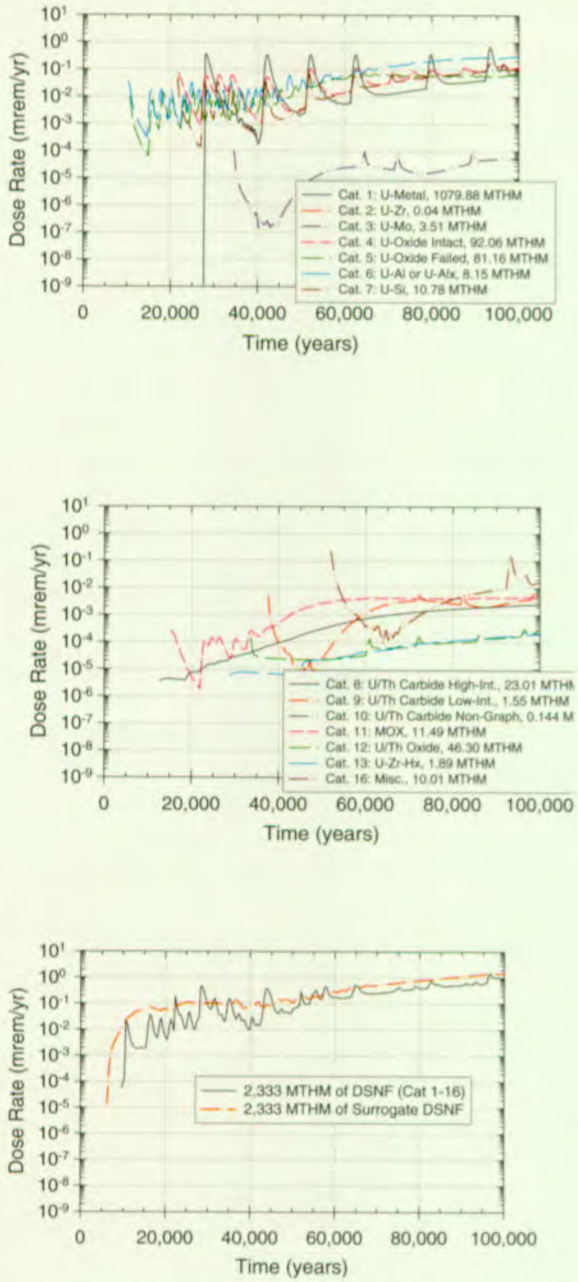

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Figure 5-39. Expected-Value Total Dose-Rate History at 20 km (12 miles) over 100,000 Years for Sixteen Categories of U.S. Department of Energy Spent Nuclear Fuel
Total dose rate for all DOE spent nuclear fuel, found by summing the individual categories, compared to the surrogate DOE spent nuclear fuel (bottom figure). Fuel is assumed to be exposed upon waste package failure (no credit for cladding).
(DSNF—DOE Spent Nuclear Fuel)

nuclear fuel, where the latter is defined as the sum of the dose rates from the various categories 1-16. The spikes on the total DOE spent nuclear fuel curve are caused by individual package failures of DOE spent nuclear fuel, with the highest spikes being from Category 1. They occur because of the small number of packages in some DOE spent nuclear fuel categories (e.g., Category 1 has 107 packages). In contrast, the surrogate spent nuclear fuel curve is smoother because of the larger number of disposed packages (2,546 waste packages).

The contributions to total dose from naval spent nuclear fuel (Category 15) that had not been analyzed previously were found to be insignificant (CRWMS M&O 1998i).

5.5.8 Comparison of Plutonium Waste Form with Commercial Spent Nuclear Fuel Equivalent Waste Form

Another waste form currently planned for disposal in the repository is plutonium waste. The plutonium waste consists of two subtypes: a Zircaloy-clad, mixed uranium-plutonium oxide spent nuclear fuel, and a can-in-canister, ceramic waste, which is a plutonium ceramic in cans that are encapsulated in high-level radioactive waste in a standard high-level radioactive waste canister. The cans of ceramic comprise approximately 12 percent of the canister volume, and four of these canisters are assumed to be disposed of in a high-level radioactive waste package. The mixed uranium-plutonium oxide fuel contains approximately 5 percent plutonium and is assumed to be used as fuel in a standard, pressurized water reactor. The spent nuclear fuel is assumed to be disposed of in 21 assembly, pressurized water reactor waste packages. There are 75 packages of mixed uranium-plutonium oxide spent nuclear fuel and 159 packages of can-in-canister ceramic (CRWMS M&O 1998i).

The effects of disposal of the 50 tons of surplus plutonium was analyzed by simulating the emplacement of the plutonium waste forms, one at a time, in the base case environment and analyzing the expected value dose history 20 km (12 miles) from the repository (CRWMS M&O 1998i).

Figure 5-40 compares the doses from 33 tons of plutonium as mixed uranium-plutonium oxide spent nuclear fuel and 17 tons of plutonium as can-in-canister ceramic with equivalent commercial spent nuclear fuel and high-level radioactive waste.

A comparison of the mixed uranium-plutonium oxide spent nuclear fuel with an equivalent amount of commercial spent nuclear fuel shows that the dose from both is nearly identical, with the mixed uranium-plutonium oxide being negligibly higher. A comparison of the can-in-canister ceramic with an equal number of packages of high-level radioactive waste indicates that the dose from the high-level radioactive waste is negligibly higher than from the can-in-canister ceramic. These results show that the dose attributed to mixed uranium-plutonium oxide fuel and can-in-canister ceramic is essentially the same as that from commercial spent nuclear fuel and high-level radioactive waste, respectively. These results can be interpreted to mean that the effects on dose from disposal of plutonium are not significant, because the spent nuclear fuel equivalent and equivalent number of high-level radioactive waste packages would have been disposed of anyway.

5.6 UNSATURATED ZONE TRANSPORT

This section examines the sensitivities of the total system performance to variations in matrix

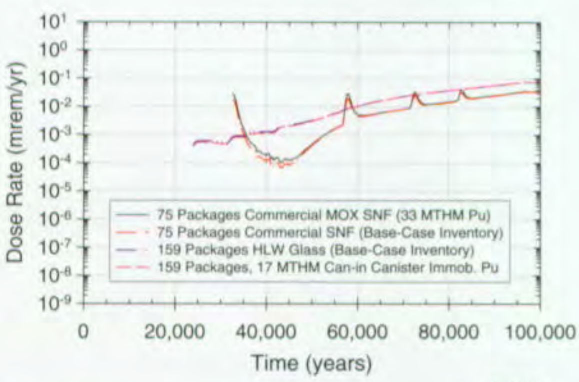

Figure 5-40. Comparison of Total Dose Rate for Plutonium Waste Forms (Mixed Oxide Fuel and "Can-in-Canister")

(CSNF—commercial spent nuclear fuel; HLW—high-level radioactive waste; MOX—mixed oxide fuel)

diffusion, sorption, and a coupled sorption and source term scenario. These sensitivity analyses are intended to complement the unsaturated zone subsystem calculations presented in Section 3.6 and provide information on how changes in the subsystem affect performance of the total system.

5.6.1 Sensitivity to Matrix Diffusion

Analyses of matrix diffusion sensitivity for unsaturated zone flow were carried out for different infiltration rates and conceptual models. The relationship between matrix diffusion and flow is important because the relative rates of these processes influence how matrix diffusion affects unsaturated zone transport. In particular, the relative rates of advective transport in the fractures to diffusion in the matrix (coupled with matrix sorption) determine how strongly matrix diffusion influences transport in the unsaturated zone.

The base case, expected-value dose-rate history is shown in Figure 4-12. This figure shows that early dose (up to about 50,000 years) is dominated by technetium; after that time, the dose is dominated by neptunium. The dominant radionuclide for dose affects the sensitivity analyses discussed below because of the differences in matrix diffusion and sorption between these radionuclides (see Section 3.6). Sorption enhances the effects of matrix diffusion because it sharpens concentration gradients, the driving force for diffusion, between the fractures and matrix. Technetium is nonsorbing and in the aqueous phase is expected to be in the form of the negatively charged pertechnetate anion, TcO_4^- (Triay et al. 1997, p 177). Mineral surfaces also are commonly negative in charge, so pertechnatate tends to be repelled slightly by the rock matrix surfaces (ibid, p.182). Both these characteristics of technetium lead to less matrix diffusion. Neptunium is weakly sorbing on the different rock matrix types in the unsaturated zone. For oxidizing conditions at values of pH less than about 7, the aqueous form of neptunium is expected to be predominantly the positively charged neptunyl cation, NpO_2^+ (ibid, p. 124, Table 47). Therefore, neptunium transport is expected to be more strongly influenced by matrix diffusion. Although values of pH are likely to be larger than 7 (see

Section 3.3), we have used the matrix diffusion coefficient for a positively charged species as a conservative measure.

Figure 5-41 shows the change in dose at the 20-km (12-mile) boundary for the expected value base case with and without matrix diffusion. In general, matrix diffusion is found to have little influence on dose; some slight variations at very early times (< 5,000 years) and then at longer times (about 70,000 to 250,000 years) are seen in the figure. Matrix diffusion is expected to have greater influence at the leading edge of the releases because the concentration gradients at the front are larger. The slight variation in responses at longer times is believed to be caused by the dominance of neptunium for total dose after about 50,000 years. Before this time, dose is dominated by technetium. The greater sensitivity of neptunium to matrix diffusion (after the leading edge of the concentration front passes through the system) is expected because of the differences discussed above the respect to matrix diffusion and sorption. The fact that matrix diffusion has much less effect on dose at the 20-km (12-mile) boundary than on the unsaturated zone radionuclide transport results (see Section 3.6) suggests that other aspects of the total system, such as engineered barrier system releases and saturated zone transport, tend to dominate the total system response.

The effect of matrix diffusion on other flow models besides the expected-value base case model was also analyzed, but not shown here because of the negligible effect. The other four flow fields in the base case (see Section 5.1) showed similar behavior to the expected-value flow field (the "base infiltration with mean fracture alpha" case), with relatively little effect on total dose rate from matrix diffusion in the unsaturated zone. The DKM/Weeps model also showed similar behavior to the expected-value base case flow field.

5.6.2 Sensitivity to Sorption

This section considers the sensitivity of performance to sorption in the unsaturated zone and does not consider changes to sorption in the saturated zone.

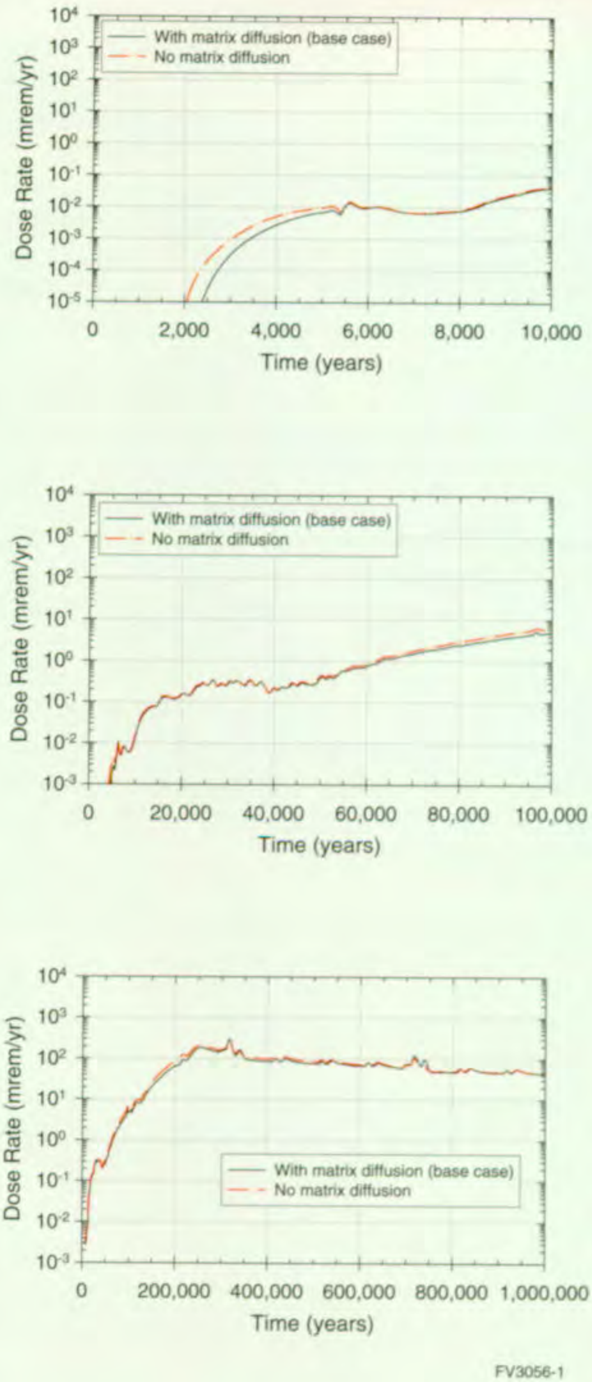

Figure 5-41. Comparison of Zero-Matrix-Diffusion (in the Unsaturated Zone) Model with Base Case Model
Effect on total dose rate at 20 km (12 miles) for three different periods.

The first sensitivity examined is to the uncertainty ranges for the distribution or sorption coefficients (K_ds) in the base case, specifically for neptunium, plutonium, uranium, selenium, and protactinium. Figure 5-42 shows the total dose rate histories resulting from using the 5th and 95th percentile values for these K_ds. The effect is negligible.

To account for other effects on sorption besides those encompassed by the base case ranges of the K_ds, an extreme alternative sorption model was examined. In particular, a model of zero sorption for the actinides (neptunium, uranium, protactinium, and plutonium) in the unsaturated zone is examined. One of the reasons that actinide sorption could disappear is that the repository may create thermal-chemical effects that could reduce sorption. One interaction that may lead to reduced actinide sorption in the unsaturated zone is an increase in pH. Higher values of pH may occur because of the interaction of water with concrete in the repository (see Section 3.3). The higher pH is believed to reduce sorption through geochemical effects on speciation of the actinides. For example, at lower values of pH (below 7), neptunium is expected to be primarily NpO_2^+, while at higher values of pH the dominant species is $NpO_2(CO_3)$ (Triay et al. 1997, p. 124, Table 47). The positively charged species of neptunium has a weak tendency to sorb onto matrix rock, but the negatively charged species is essentially nonsorbing.

Figure 5-43 compares total dose rates over the three time frames for the base case and the zero sorption case. It indicates that the early doses are not affected by sorption (primarily because of technetium), but that sorption helps to reduce the total dose rate between 50,000 and 250,000 years (primarily because of neptunium and plutonium) by about a factor of three.

The effects of neptunium sorption for this alternative model are given in Figure 5-44 over the first 100,000 years. When compared to the base case, the effect of K_d=0 of neptunium is found to be relatively small because neptunium is only weakly sorbing in the base case. Similar results are found

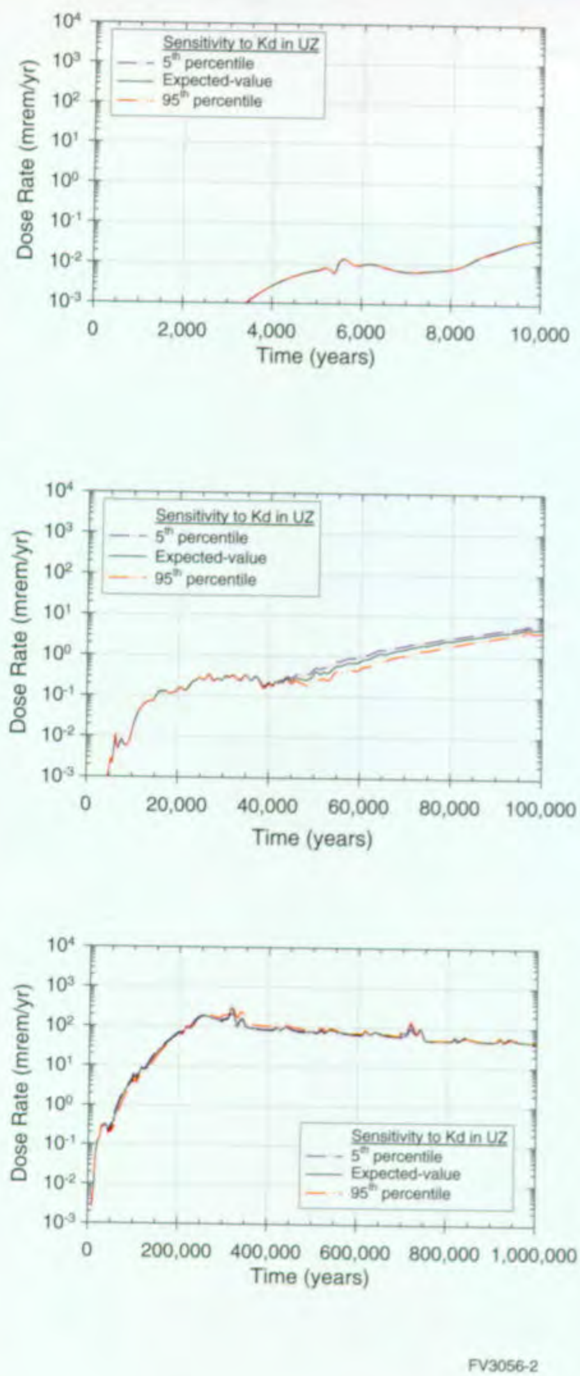

Figure 5-42. Effect of Radionuclide Distribution Coefficients in the Unsaturated Zone Matrix on Total Dose Rate at 20 km (12 miles)
Comparison of base case expected-value K_ds with the base case 5th and 95th percentile values. The primary effect is from the neptunium K_d.

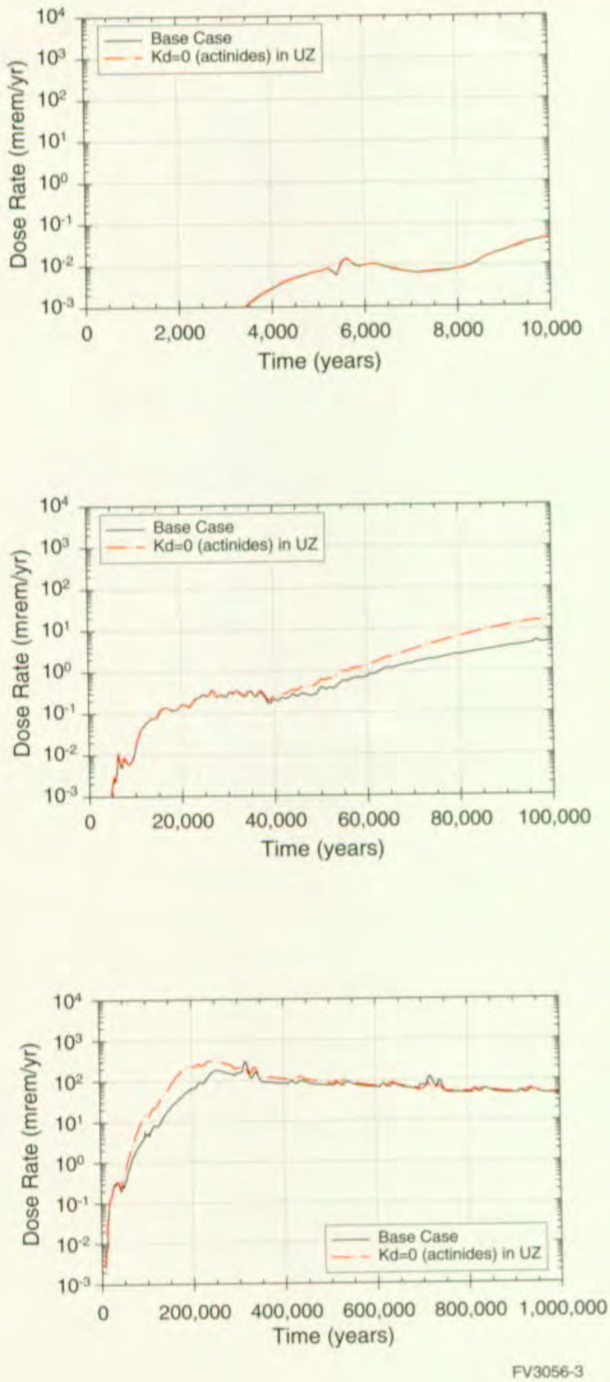

Figure 5-43. Sensitivity of Total Dose Rate to an Alternative Unsaturated Zone Transport Model that Assumes the Actinide K_ds in the Unsaturated Zone Rock Matrix are Equal to Zero
The primary effect is from the plutonium K_d.

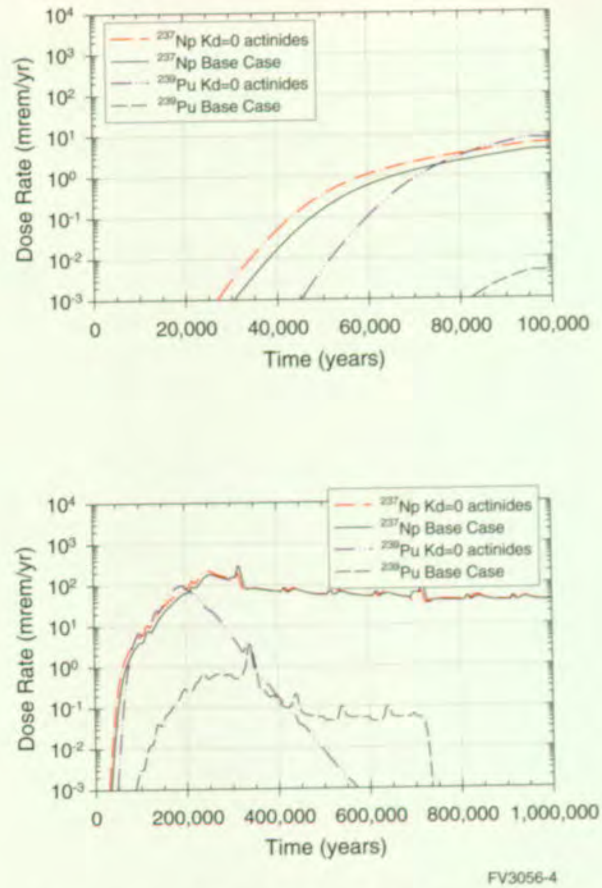

Figure 5-44. Sensitivity of Neptunium-237 and Plutonium-239 Dose Rate to an Alternative Unsaturated Zone Transport Model that Assumes the Actinide K_ds in the Unsaturated Zone Rock Matrix are Equal to Zero

for uranium, which is also a weakly sorbing element.

The effects of sorption on plutonium transport are more complex. Plutonium is a strongly sorbing radionuclide, but is also subject to colloidal transport. A portion of the colloid interaction is believed to be caused by sorptive mechanisms. Sorption on the rock matrix retards plutonium transport, while sorption on colloids facilitates plutonium transport. Therefore, the sensitivity of plutonium transport to sorption is a result of two competing effects. The overall effect of eliminating sorption on plutonium transport is shown in Figures 5-44 and 5-45 for plutonium-239 and plutonium-242, respectively. For both plutonium

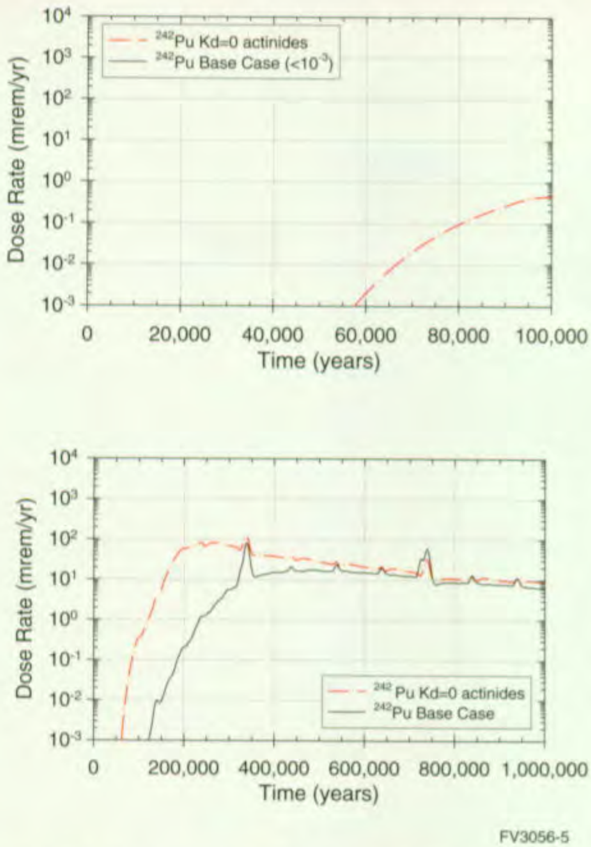

Figure 5-45. Sensitivity of Plutonium-242 Dose Rate to an Alternative Unsaturated Zone Transport Model that Assumes the Actinide K_ds in the Unsaturated Zone Rock Matrix are Equal to Zero

isotopes, the elimination of sorption causes higher dose rates. In fact, plutonium-239 dose rates slightly exceed those of neptunium-237 for the zero-sorption model between 100,000 and 200,000 years. These sensitivity results indicate that the sorption of aqueous plutonium on the rock matrix is dominant over the sorption of plutonium on colloids. Also, it is evident that the factor-of-3 increase in total dose indicated on Figure 5-43 between 100,000 and 200,000 years is caused by the plutonium isotopes becoming as important as neptunium-237.

5.6.3 Sensitivity to Combined Effects of Source Term and Unsaturated Zone Sorption

In addition to reducing sorption, increased pH from concrete dissolution may also promote waste package corrosion and waste form dissolution, as discussed in Sections 5.3 and 5.4, accelerating releases from the engineered barrier system. The dose effects at 20 km (12 miles) caused by the combined effects of zero sorption in the unsaturated zone and concrete-modified waste package and waste form degradations are shown in Figure 5-46. Dose is much higher before 10,000 years.

This early peak in dose is caused by greater releases of technetium from the engineered barrier system (see Section 5.3). Similarly to Figures 5-44 and 5-45, neptunium and plutonium are the primary causes of the later increase in doses. These increases in dose for neptunium and plutonium may be attributed to a combination of greater transport rates (see Section 5.6.1) and greater releases from the engineered barrier system (see Section 5.3). Changes in releases from the engineered barrier system indicate that plutonium-242 would always provide a greater portion of the dose than neptunium. However, sorption in the saturated zone is found to retard plutonium release to the accessible environment sufficiently so that the neptunium, which is less strongly sorbing, provides a greater portion of the dose following the technetium peak but before the arrival of plutonium.

By comparing Figures 5-14 and 5-43, it is apparent that most of the change in total dose rate for this alternative model is due to the effect on engineered barrier system releases up until 50,000 years and due to the effect on sorption in the unsaturated zone after 100,000 years. Between those two times, it is a combination of the two effects.

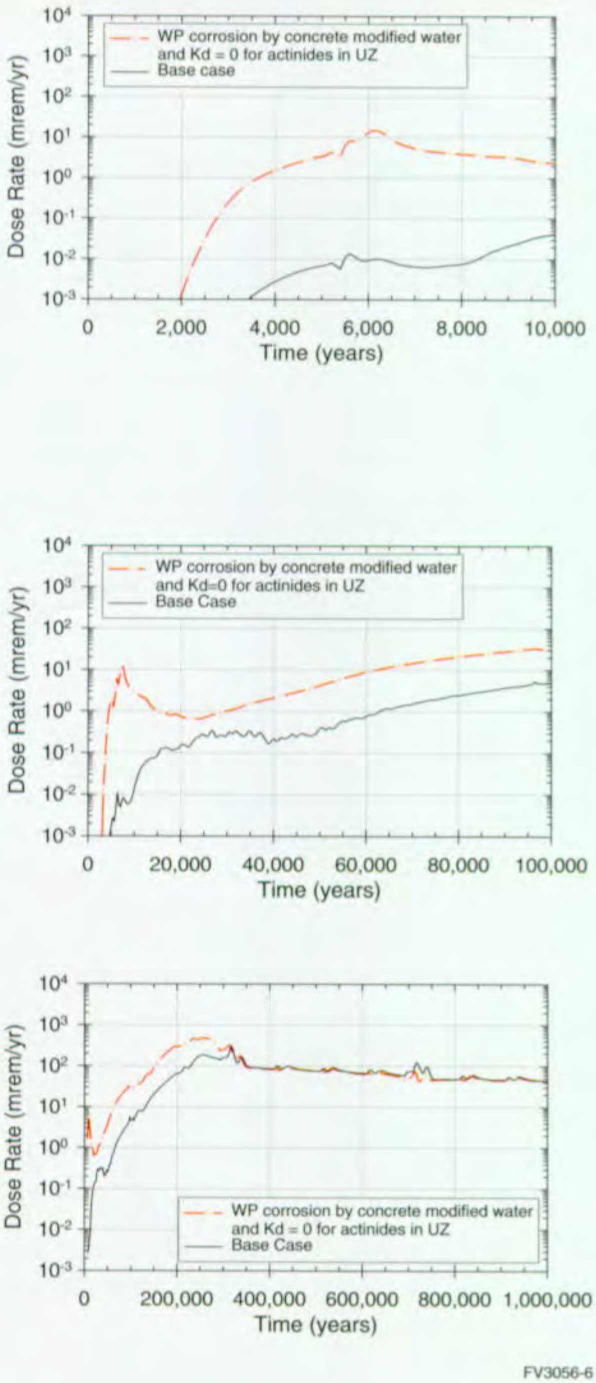

Figure 5-46. Sensitivity of Total Dose Rate to the "Alkaline Plume" Model
This model is a result of concrete degradation in the drift which raises the pH in the engineered barrier system and in the unsaturated zone. This causes an increased rate of waste package and waste form degradation and decreased (zero) sorption in the unsaturated-zone. Compare Figures 5-14 and 5-42. (UZ—unsaturated zone)

5.7 SATURATED ZONE FLOW AND TRANSPORT

Three sensitivity studies were conducted to look at the effects of base case assumptions about saturated zone flow and transport. The sensitivity studies addressed the importance of the dilution factor, the fraction of the saturated-zone flow path in alluvium, and how six flow tubes are combined to calculate the final concentration.

Assumptions about the flow model and parameters were not addressed because estimated variation in these areas would affect only radionuclide travel times and not significantly affect dilution and peak dose rate. Therefore, they are relatively less important. Also, changes in the effective-porosity parameter cause changes in the groundwater velocity. The effective porosity was considered probabilistically in the base case (Section 3.7). Assumptions about transport variables (other than the dilution factor) were not addressed for two reasons:

- These parameters also primarily influence radionuclide travel time, not dose rate
- They are defined with probability distributions and covered in the probabilistic analyses (Section 4.3)

One issue that cannot be addressed yet is the amount of dilution at the interface between the unsaturated and saturated zones. No additional dilution is allowed when contaminated water from the unsaturated zone enters the saturated zone. However, the contamination is assumed to uniformly spread through the water in each of the six areas at the base of the unsaturated zone so that dilution is implicitly considered. This assumption might overestimate dilution when a small number of containers release waste. As the number of releasing containers increases, the assumption becomes more appropriate.

5.7.1 Sensitivity to Dilution

The first sensitivity study addresses how the variation in the dilution-factor distribution affects dose rate. Figure 5-47 shows the results of the expected-value base case. Also shown are two calculations that are the same as the expected-value base case, except that one calculation uses the 5th-percentile value of the base case dilution-factor distribution and the other calculation uses the 95th-percentile value. In the dilution-factor distribution, the median value is 10; the 95th-percentile value is a dilution factor of 49, or approximately a factor of 5 above the median. The 5th-percentile value is a dilution factor of 1.7, approximately a factor of 6 below the median. The dose is proportional to the radionuclide concentrations in the groundwater, and the concentration is inversely proportional to the dilution. The 95th-percentile dilution factor does decrease the expected-value-case dose rate by about a factor of 5. The 5th-percentile dilution factor does not increase the dose by a factor of 6. The 5th-percentile dilution is small enough so that the usual method of calculating the final concentration—adding the six stream tubes—exceeded the maximum undiluted concentration in the stream tubes (Sections 3.7.2.3 and 5.7.3). Hence, the dose-rate curve for the 5th-percentile dilution factor is calculated using the maximum undiluted concentration. It is significant that the dose rate does not increase in proportion to the dilution factor at very small dilution factors. The final difference between the 5th-percentile and 95th-percentile curves ranges between approximately a factor of 10 at early times and during superpluvials, and approximately a factor of 7 at other times. This difference is enough to cause the dilution factor to be one of the most important parameters (Section 4.3).

5.7.2 Sensitivity to Alluvium Fraction

The second sensitivity study addresses how the fraction of the saturated zone flow path in alluvium affects dose rate. Alluvium has higher effective porosity and higher radionuclide retardation as compared to the fractured tuff, so transport tends to be significantly slower through the alluvium. Figure 5-48 shows dose-rate history curves for the expected-value base case plus two additional cases defined by setting the alluvium fraction to the 5th percentile and the 95th percentile of its base case probability distribution while keeping all other

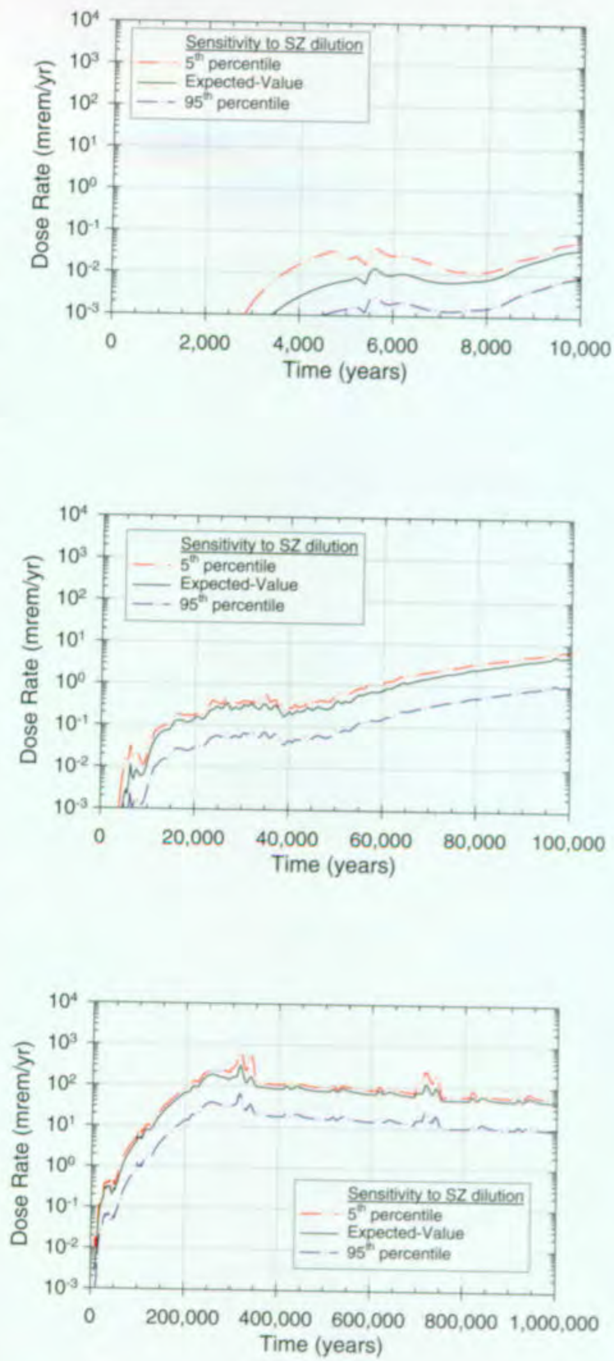

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Figure 5-47. Total Dose-Rate History Curves for the Expected-Value Base Case and Cases that Have Saturated Zone Dilution Factor at the 5th and 95th Percentiles of its Base Case Distribution (SZ—saturated zone)

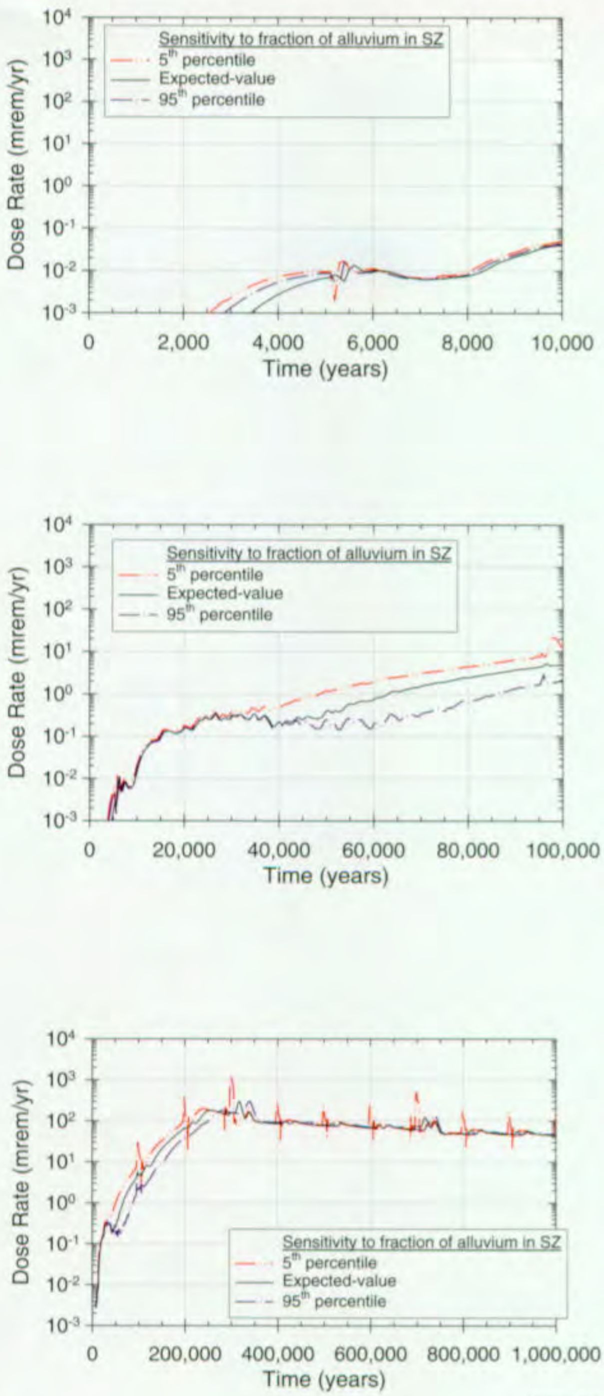

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Figure 5-48. Total Dose-Rate History Curves for the Expected-Value Base Case and Cases that Have the Fraction of the Saturated Zone Flow Path in Alluvium at the 5th and 95th Percentiles of its Base Case Distribution
(SZ—saturated zone)

parameters fixed at their expected values. Figure 5-48 shows that alluvium fraction affects the dose rate by a moderate amount. The most interesting behavior is shown in the 1-million-year plot (Figure 5-48, bottom). The spikes caused by climate changes are much more pronounced in the fifth-percentile case. This behavior results because the alluvium in the flow path has a damping effect on the climate-change spikes, so when there is less alluvium the spikes are larger.

5.7.3 Sensitivity to Method of Flow-Tube Combination

The third sensitivity study examines the effect on the dose-rate results of how the six one-dimensional flow tubes are combined, for the saturated zone, to estimate the final, diluted concentration at the biosphere/geosphere interface. Radionuclide concentrations at a distance of 20 km (12 miles) were calculated separately for each of the six flow tubes (Section 3.7). The goal is to preserve the spatial variability but allow diffusion and dispersion out of the individual flow tubes and potentially into neighboring flow tubes. Three methods are considered for calculating the concentration used for determining dose:

- The final concentrations in the six flow tubes are summed. If the sum is larger than the greatest initial concentration in one of the flow tubes, then that initial concentration is substituted for the result. It is not physically realistic for concentrations to increase above the initial maximum concentration. This method overestimates the concentration; it is the method used in the base case.

- The final concentration in the flow tube that is the greatest of all the flow tubes is selected as the maximum concentration. This method underestimates the final concentration because radionuclides from other flow tubes might diffuse or disperse into the selected flow tube. (By the principle of linear superposition of solutions, two spreading plumes that overlap must have at least as high a maximum concentration as if they did not overlap.)

- One flow tube is used in place of the six flow tubes to calculate the final concentration. This method also underestimates the final concentration, because initially all radionuclides are distributed in all the water from the unsaturated zone, thereby ignoring spatial variability and omitting any potential high-concentration regions.

The results of the concentration-combination study are presented in Figure 5-49. The first method is incorporated in the expected-value base case. The other methods are modifications of the expected-value base case. As anticipated, the first method—summing the concentrations—causes the greatest doses. The second method—using the highest concentration flow tube—causes doses that are approximately a factor of four below the first method. The third method—using one flow tube—causes doses that are approximately a factor of six below the first method. The relationship depends on the distribution of radionuclides in the flow tubes at the boundary between the unsaturated and the saturated zones. The calculated dose depends on the method of determining the final concentration. However, the differences in dose are less than the differences caused by parameters such as the dilution factor that are significant to the final results (compare Figures 5-47 and 5-49).

5.8 BIOSPHERE

Three sensitivity studies were conducted to look at the effects of assumptions about the biosphere on the TSPA calculations. Other assumptions internal to the biosphere modeling that affected calculation of biosphere dose conversion factors also were considered (see Section 3.8.3).

The sensitivity studies address how the variability in biosphere dose conversion factors influences the final calculated dose rates, how assumptions related to dilution at the well and in the biosphere affect the calculated dose rates, and how assumptions about the critical group affect calculated dose rates.

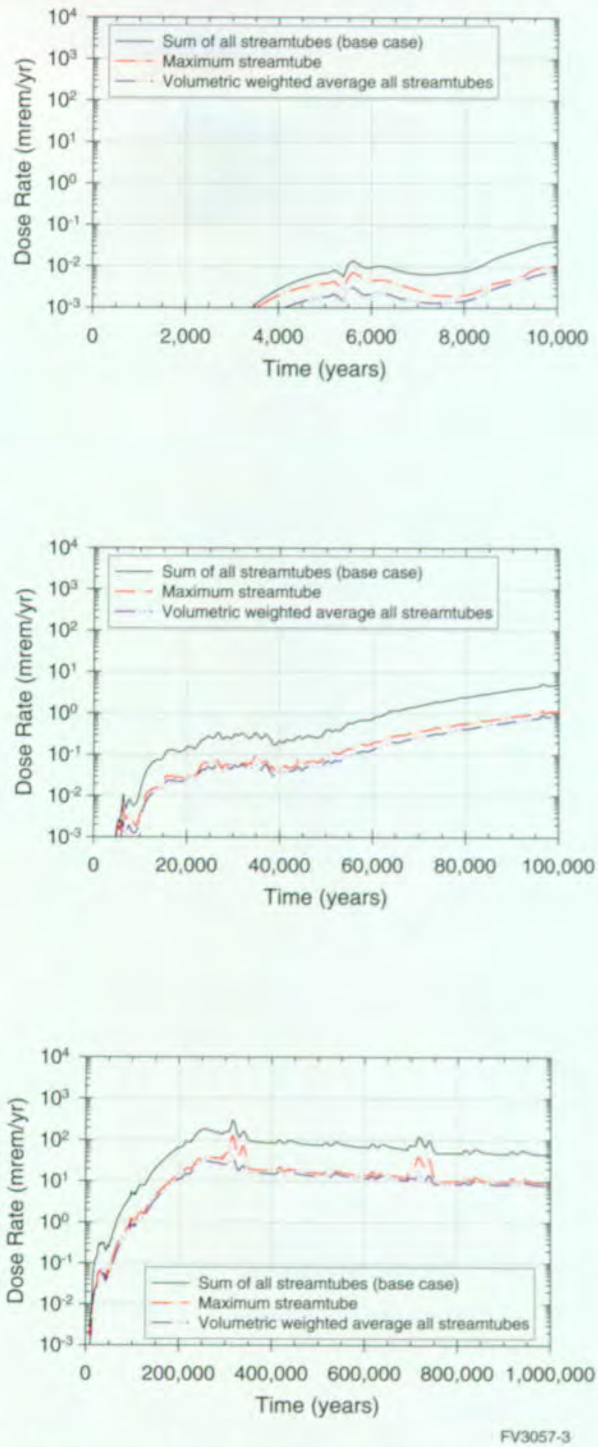

Figure 5-49. Total Dose-Rate History Curves for the Expected-Value Base Case and for Two Sensitivity Cases that Use Different Methods for Combining the Six Flow Tubes Used to Model the Saturated Zone

5.8.1 Sensitivity to Biosphere Dose Conversion Factor

The first sensitivity study addresses how the final dose calculation is affected by the spread in the distributions used to define the biosphere dose conversion factors. Figure 5-50 shows the results of the expected-value base case. Also shown are two calculations that are the same as the expected-value base case, except that in one the 5th-percentile values of each biosphere dose conversion factor distribution for each radionuclide is used and, in the other calculation, the 95th-percentile values are used. In the figure, using the 5th-percentile, biosphere dose conversion factors decreases expected-value-case dose rates by about a factor of 3; using the 95th-percentile biosphere dose conversion factors increases the dose rates by about the same amount. The spread in most of the biosphere-dose-conversion-factor distributions is about an order of magnitude (a factor of 10), and thus the difference in doses calculated by the 5th-percentile and 95th-percentile values also is about an order of magnitude.

The three curves shown in Figure 5-50 have an identical shape. The dose is proportional to the concentration of radionuclides in the groundwater, and the biosphere dose conversion factors for the radionuclides are the constants of proportionality. Although this is a linear relationship, when combined with the relatively large uncertainty in biosphere dose conversion factors as shown by the order-of-magnitude spreads in the distributions, it is enough to cause the biosphere dose conversion factor to be an important parameter in TSPA-VA. (That is, out of 177 parameters defined with probability distributions in the base case, the biosphere dose conversion factors are number three in importance over the one-million-year time frame, as shown in Figure 4-34.)

5.8.2 Sensitivity to Dilution at the Well and in the Biosphere

The second sensitivity study examines assumptions made in the base case analyses concerning well-water withdrawal, well location, and food-source

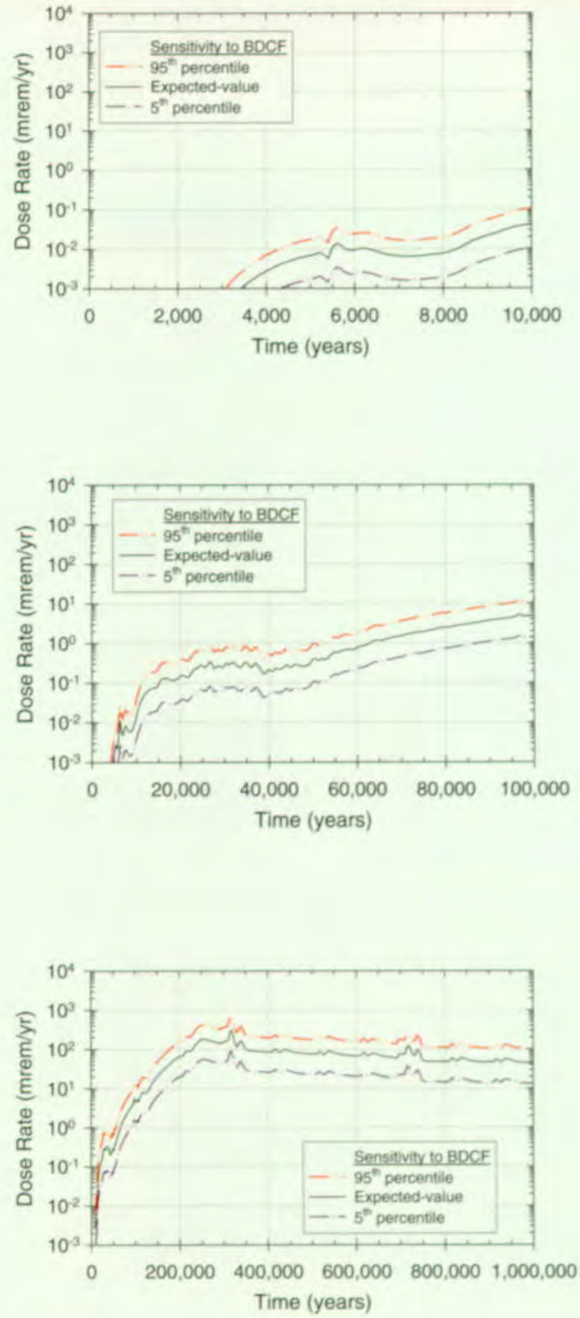

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Figure 5-50. Total Dose-Rate History Curves for the Expected-Value Base Case and Cases that Have Biosphere Dose Conversion Factors at the 5th and 95th Percentiles of Their Base Case Distributions (BDCF—biosphere dose conversion factors) location; these assumptions are related to the choice of an individual-dose performance measure. Some of these assumptions are as follows:

- The individual's water source is always at the point of maximum contamination in the aquifer.

- There is no dilution during withdrawal of water from the aquifer; that is, there is no mixing of contaminated water with uncontaminated water when the water is pumped from the ground or when the water is stored in a tank.

- The locally produced foodstuffs that this individual consumes, as determined by a survey of existing inhabitants of the Amargosa Valley region (Section 3.8.1), were all grown with this maximally contaminated groundwater. In other words, consumables, possibly grown away from contaminated groundwater, but still in the Amargosa Valley region, were considered to be contaminated.

If the plume of contaminated groundwater is large and dispersed, these assumptions are reasonable. In the saturated zone modeling, the contaminated groundwater travels in flow tubes and is typically diluted by a factor of 10. The original plume has cross-sectional dimensions of about 3,000 m (10,000 ft) horizontally and 10 m (30 ft) vertically. The plume at 20 km (12 miles) downgradient has an undefined cross section, but with a dilution factor of 10, it might have dimensions of about 10,000 m (30,000 ft) horizontally and 30 m (100 ft) vertically. At the estimate groundwater flux of 0.6 m/year (2 ft/year), such a plume would constitute approximately 180,000 m^3/year (50 million gal/year) of water. (This approximation is somewhat less than the volume actually used in the expected-value base case.)

To examine the assumptions that all drinking water and all local consumables are produced using well water containing the highest concentration of radionuclides, the average dose rate per person is introduced. To calculate an average dose rate per person, all contamination is assumed to be spread equally in all water available to a group of people. Also, each member of the group is assumed to use the same amount of water—the average water usage per person. For a given volume of contaminated water, the size of group of people can be estimated. For a given mass of radionuclides, the radionuclide concentration—and thus the average dose rate per person—can be estimated.

The mass of each radionuclide reaching the 20-km (12-mile) distance from Yucca Mountain at a given time can be taken from the TSPA-VA expected-value base case results. This mass is dissolved in a certain amount of water, for example, if the approximate base case volume is assumed, then this mass is dissolved in 180,000 m^3/year of water directly linked to the repository. If, however, it is assumed that this mass is dissolved in all of the groundwater that is currently being pumped from the Amargosa Valley, which includes water sources not linked to the repository, then this mass is dissolved in an estimated 12 million m^3/year (3 billion gal/year), resulting in a lower concentration of radionuclides in water. Note that based on this estimate of present-day water production, the average water usage for a member of the Amargosa Valley population (approximately 1,300 people) is about 9,300 m^3/year or 2.5 million gal/year.

The peak radionuclide release over a one-million-year period is estimated to occur 317,000 years in the future and the major contributor to the dose rate is neptunium-237, which reaches 20 km (12 miles) at the peak rate of 12 g/year. If this mass of radionuclide is dissolved in the amount of water corresponding to the current estimated volume of water produced in Amargosa Valley instead of the base case volume, then the resulting radionuclide concentration is 1.0×10^{-6} g/m^3. The dose rate resulting from using this contaminated water is found by multiplying the radionuclide concentration by the biosphere dose conversion factor, which for neptunium-237 is 4.6×10^{6} mrem/year per g/m^3. The result is an average peak dose rate of 4.6 mrem/year. If, however, the approximate base case volume is assumed, the resulting neptunium-237 concentration is 6.7×10^{-5} g/m^3, and a corresponding average peak dose rate is 308 mrem/year.

(This dose rate does not correspond exactly to the value calculated for the expected-value base case described in Section 4.2. The peak dose rate in the expected-value base case occurred during the long-term-average climate and was calculated for a larger groundwater flux. However, the concentrations for the various flow tubes used were combined in the expected-value base case, yielding a similar concentration and dose rate to the values presented here.) This sensitivity analysis shows that dose rate is 67 times higher for the base case than if the entire water-production dilution is assumed. Based on the current Amargosa Valley water usage per resident (9,300 m^3/year), the approximate base case volume of groundwater (180,000 m^3/year) could only support about 20 people. Figure 5-51 presents the relationship between the average dose rate per person and the size of the group using contaminated water, calculated using the peak radionuclide release from the expected-value base case results for the one-million-year period.

This analysis indicates that assumptions related to how radionuclides are distributed in the biosphere

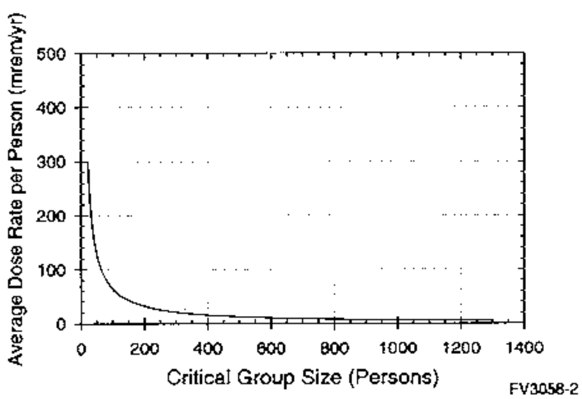

Figure 5-51. Average Dose Rate as Function of Critical Group Size for the Amargosa Valley, Estimated for the Expected-Value Base Case
The line indicates the average dose rate per person; the plateau value corresponds to the 300 mrem/year maximum dose rate calculated in the base case for the reference person. Approximately 20 people could incur this dose rate based on the calculated amount of radionuclides released to the biosphere and the average water usage per person in the Amargosa Valley. The distribution of these radionuclides among the entire population would reduce the dose rate to an average of approximately 5 mrem/year.

(i.e., the volume of water the radionuclides are dissolved into) could affect calculations of dose rates by possibly a factor of 70. It also sets approximate limits on how many people (20) can incur the maximum dose rate calculated for the base case for the assumed usage.

5.8.3 Sensitivity to Critical-Group Definition

The third sensitivity study examines the effect on calculated doses of the assumptions made for the base case regarding what constitutes the critical group. In the base case, radiation doses were calculated for a reference person based on an average resident of Amargosa Valley, using distributions for consumption of water and locally grown food taken from a survey of residents (Section 3.8.1). In this section the base case is compared with two other possible dose receptors. One alternative is a "subsistence farmer," who lives entirely on water taken from the maximally contaminated part of the contaminant plume, including all drinking water and all water used to grow produce and livestock. The other alternative is a "resident farmer," who is assumed to consume 50 percent locally grown food that is contaminated by water from the maximally contaminated part of the contaminant plume.

Figure 5-52 compares the two alternative dose calculations with the base case. The spread between the highest method (the subsistence farmer) and the lowest method (the base case) is a factor of 4 to 5. These results are consistent with the differences in these receptors types presented in Figure 3-80.

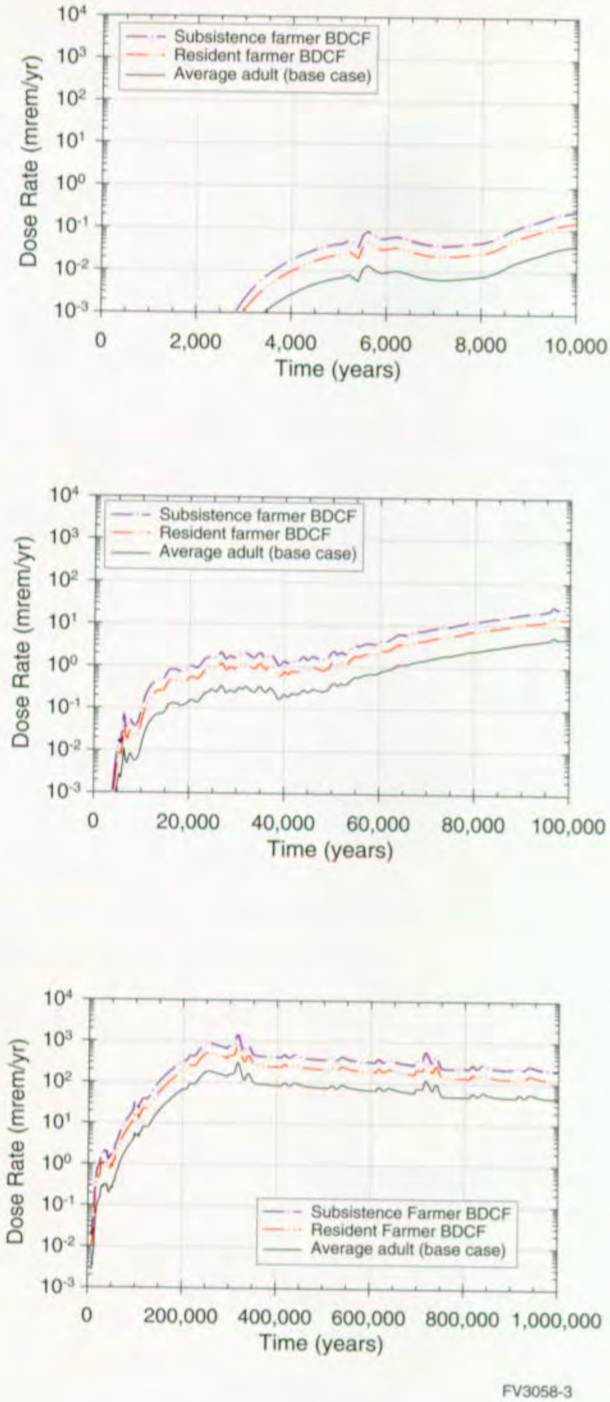

Figure 5-52. Total Dose-Rate History Curves for the Expected Value Base Case and for Two Sensitivity Cases in Which Different Critical Groups are Assumed

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6. SUMMARY AND CONCLUSIONS

The objective of the TSPA-VA was to conduct and document "a total system performance assessment based upon the design concept and the scientific data and analysis available by March 1998, describing the probable behavior of the repository in the Yucca Mountain geological setting relative to the overall system performance standards" (Public Law 104-206). This volume of the VA contains the information, analyses, and interpretations used to describe "probable" behavior of the proposed Yucca Mountain repository system.

The projected behavior of the Yucca Mountain repository system is based on the site-specific scientific data and analyses summarized in Volume 1, and in Section 3 of this volume. The Yucca Mountain repository and waste package design concepts used in this TSPA are described in detail in Volume 2, and the relevant scientific data and analyses used to evaluate the behavior of these designs are summarized in Section 3 of this volume. This information has been used to develop an estimate of the probable behavior of the repository disposal system, comprised of both engineered and natural components. The results of these analyses have been presented in Section 4 of this volume.

In examining and summarizing results describing the "probable" behavior of the Yucca Mountain repository system, it is important to understand that the evaluations are indicators of the postclosure performance impacts of the proposed repository. The analyses should not be construed as predictions in the normal sense of the word. Although an attempt has been made to incorporate the most current understanding of key processes affecting the long-term behavior of Yucca Mountain, such projections are uncertain. Although many of these uncertainties will still remain at the time of the License Application (LA), DOE will reduce the significant uncertainties sufficiently and modify the design as appropriate to provide NRC with reasonable assurance that post closure performance objectives can be met. The uncertainty is a result of several factors:

- The periods are long. The period for the quantitative analyses extends to 10,000 years and beyond. Over these long periods, there is uncertainty in the definition of likely future environments. This period is also long compared to available information about a number of principal factors affecting repository performance such as the degradation rates of corrosion-resistant metals.

- The site is heterogeneous. Water movement in the unsaturated and saturated zones at Yucca Mountain is expected to vary with location and with time because of the heterogeneous nature of the fractured rocks. Precisely defining the flow paths and volume and geochemical interactions along the flow paths is not possible.

- The system is coupled. Coupled interactions are expected to occur near the emplaced wastes. These interactions include thermal, chemical, hydrologic, and mechanical processes that can influence one another. For example, the chemical alteration of the rock caused by a temperature increase may cause the hydrologic properties of the rock to change. It is difficult to predict the magnitude and extent of these changes with any precision.

- Future populations are unknown. It is not possible to reasonably forecast changes in human activities. Therefore, the doses calculated are based on the activities of the present-day population in the region around Yucca Mountain. This assumption, which is consistent with internationally accepted recommendations (see Section 3.8), implies that the doses represent the range of likely performance of a repository for a hypothetical population.

Although these uncertainties are recognized in the analyses and interpretation of results, they do not detract from the goal of objectively evaluating how the system is likely to perform. To precisely identify the exact magnitude of the effects and consequences for disposing 70,000 metric tons of nuclear waste at Yucca Mountain is not expected by the regulators. For example, NRC has noted in 10 CFR Part 60.101 (a)(2):

> While these performance objectives and criteria are generally stated in unqualified terms, it is not expected that complete assurance that they will be met can be presented. A reasonable assurance, on the basis of the record before the Commission, that the objectives and criteria will be met is the general standard that is required. For 60.112, and other portions of this subpart that impose objectives and criteria for repository performance over long time periods into the future, there will inevitably be greater uncertainties. Proof of the future performance of engineered barrier systems and the geologic setting over time periods of many hundreds or many thousands of years is not to be had in the ordinary sense of the word. For such long-term objectives and criteria, what is required is reasonable assurance, making allowance for the time period, hazards, and uncertainties involved, that the outcome will be conformance with those objectives and criteria.

In addition, EPA has noted the following with respect to the degree of confidence that is expected (40 CFR 191):

> Performance assessments need not provide complete assurance that the requirements of 191.13(a) will be met. Because of the long period involved and the nature of the events and processes of interest, there will inevitably be substantial uncertainties in projecting disposal system performance. Proof of the future performance of a disposal system is not to be had in the ordinary sense of the word in situations that deal with much shorter time frames.

The assessment provided here is not final. It does not include the type of evaluation of a number of design options that will be conducted for the LA. Also, analyses to address design margin and defense in depth have not yet been completed. However, this assessment does include sufficient sensitivity and uncertainty analyses to illustrate the range of possible behaviors that might be anticipated. These analyses have assisted DOE in focusing the data-collection, model-validation, and design-development activities between now and the LA. While the projections of future behavior presented appear reasonable, additional information may be needed to demonstrate regulatory compliance in a licensing proceeding.

Although uncertainties exist regarding many factors affecting repository performance, these uncertainties have been reasonably bounded and analysis have been satisfactorily completed using these bounded values. Therefore some uncertainties and areas of limited data will remain as they are while other uncertainties may require further study in order to provide a sufficiently robust analysis of repository performance for the LA.

As noted above, it would be ideal to be able to make all decisions based on complete information. However, the behavior of the repository system over tens of millennia will never be known with certainty. The regulatory agencies agree that reasonable conclusions about whether a repository system can be expected to provide the public with adequate protection from nuclear waste need not include this degree of absolute certainty.

6.1 SCIENTIFIC DATA AND ANALYSES IN THE TOTAL SYSTEM PERFORMANCE ASSESSMENT FOR THE VIABILITY ASSESSMENT MODELS

One of the principal objectives of the analyses for this TSPA was to use all the available scientific data and analysis in the performance evaluation of the Yucca Mountain repository system. The data consist of a suite of laboratory, surface based, and in situ tests and observations conducted as part of the YMP. These tests and observations provided data that have been used in analytical tools consisting of conceptual, process, abstracted, and numerical models. These models, in conjunction with the test data for defining a range of parameter values, have been used as the basis for this TSPA.

An evaluation of total system performance is a projection or estimate of the expected future behavior of the repository system. With the possible exception of some naturally occurring uranium ore bodies and other natural analogs, there are no directly observable phenomena that are comparable to the potential repository system at Yucca Mountain. Because of this, it is necessary to extrapolate the projections of future behavior based on present-day observations of the same phenomena. The extrapolation of the relevant phenomena into the future is accomplished using models based on laboratory, surface based and in situ test data and observations. These models describe the behavior of the relevant processes.

Laboratory test data are valuable in improving confidence in models used for future projections of behavior because they can be conducted under well-controlled conditions and can be designed to address specific data needs.

Examples of laboratory testing that has been used in the development of models for this TSPA include:

- Corrosion testing of candidate waste package metals, including the outer corrosion allowance and the inner corrosion resistant barrier (Alloy 22)
- Intrinsic dissolution testing of the spent nuclear fuel and glass waste forms
- Alteration tests and secondary phase formation of spent nuclear fuel and glass
- Radionuclide mobility tests including solubility and colloid formation
- Radionuclide sorption tests
- Coupled thermal, mechanical, chemical, and hydrologic tests of rock alteration
- Rock matrix property tests to determine hydrologic properties

Surface based tests and observations are valuable in determining site-specific properties necessary to define how water moves through the unsaturated and saturated zones of the Yucca Mountain repository system. Surface based tests are conducted from a range of shallow (about 10 m, or 33 ft, deep) and deep (over 1,000 m, or 3,300 ft, deep) boreholes. The types of surface based tests and observations include:

- Mapping geologic features and hydrostratigraphy
- Mapping fracture characteristics and lithology
- Testing hydrologic properties
- Observing ambient water content and matrix potential in the unsaturated zone
- Measuring precipitation, temperature, and other meteorological variables
- Measuring moisture transients in the near surface (net infiltration)
- Measuring seismological disturbances
- Mapping and analyzing volcanic features
- Observing water potential in the saturated zone
- Observing ambient temperature and aqueous geochemistry
- Cross-hole testing of transport characteristics

In situ tests and observations are valuable in determining the site-specific properties necessary to define environmental conditions that control performance of the engineered barriers of the Yucca Mountain repository system. In situ tests and observations at Yucca Mountain have been made from a number of niches constructed in the Exploratory Studies Facility and other test facilities near Yucca Mountain, such as at Busted Butte. Although in situ tests are expensive because of the construction and operational costs of the underground facilities, they are the principal means of testing the host rock under conditions similar to what might be expected in the presence of a repository. Examples of in situ tests and observations include:

- Single and drift scale heater tests to evaluate the thermal, hydrologic, chemical, and mechanical response of the rock when heated

- Observations of in situ water content, chemistry, isotopes, and mechanical stress of the rock mass around the underground tunnels

- Tests of in situ hydrologic, geochemical, and transport properties of the rock mass

These observations and tests form the basis for a suite of process and numerical models that have been used to develop this TSPA analyses. Each of these models and their abstraction into the TSPA model are described in Section 3. The term "abstraction" is used to indicate the extracting of essential information. The TSPA models are usually referred to as abstracted models. Details of this abstraction are presented in the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

While these tests and observations form the scientific basis for the models used, it is important to acknowledge the uncertainty in these models. Part of this uncertainty is caused by the complexity of the individual processes and their interactions. Another component of the uncertainty is caused by the variability of the natural system and the inability to completely characterize this variability. To address the uncertainty in each of these process models, DOE has conducted a number of different expert elicitations. The results of these elicitations have provided this TSPA analyses with a reasonable range of expected parameters and models that extend the range beyond what might be determined solely by the test information.

6.2 PROBABLE BEHAVIOR OF THE REFERENCE DESIGN

The performance assessments provide a picture of the repository system evolution and probable behavior, where probable behavior is defined as the behavior specified by the base case. The assessment considers the potential for radionuclide release from the repository system and transport 20 km (12 miles) from the repository, where the radionuclides could be drawn in water from a hypothetical well. The assessment is made for the repository design concept under consideration as described in Volume 2. The behavior of other design concepts could be somewhat different. These analyses, including the sensitivity studies, provide a broad indication of the kind of behavior expected for the system, the probable behavior. This section briefly describes the design concept and discusses the probable behavior of the system.

The design concept for the Yucca Mountain repository system is to place the various waste forms, whether commercial spent nuclear fuel, defense high-level nuclear waste glass, or DOE-owned spent nuclear fuel, into large waste packages. These waste packages are made of an inner highly corrosion resistant nickel alloy (Alloy 22) and an outer layer of carbon steel. The waste packages are placed on supports within mined drifts located about 300 m (1000 ft) beneath the surface. Following emplacement of individual waste packages, the drifts are sealed to prevent access, but otherwise the facility will be designed to remain open for 100 years (and perhaps several hundred years) after the start of waste loading to allow maintenance and observation until a decision is made to permanently close the facility. This design forms the basis of this TSPA. The details of this design concept are presented in Volume 2.

The presentations in Sections 3 and 4 showed the probable behavior of the system from the viewpoint of a series of model results that, when integrated, yielded a description of the overall system behavior. This description was summarized in Section 2.4 as a series of time slices that synthesize the system response to the proposed design concept. These time slices were:

- Time of waste emplacement to time of repository closure
- Time of repository closure to several hundred years after closure
- Several hundred years to several thousand years after closure
- Several thousand years to ten thousand years after closure
- Ten thousand years to several tens of thousands of years after closure
- Several tens of thousands of years to one hundred thousand years after closure
- Several hundred thousand years to as long as a million years after closure

The assessment described in Section 4 indicates that the vast majority of radionuclides in the waste is immobile and never leaves the repository even if in contact with water. A small number of radionuclides, notably technetium-99, iodine-129, neptunium-237, and those radionuclides transported by colloids, are sufficiently mobile under some conditions that they could reach the biosphere downgradient from the repository. Therefore, these analyses indicate that the most important factors for system performance over time are the amount of water likely to contact the waste packages and the amount of waste exposed to that water. Consequently, long lived waste packages and engineered barriers or other factors that are effective in further limiting the contact of water with the waste will also be highly important to performance.

Under the base case scenario the quantities calculated to reach the biosphere are small: a negligible amount in 10,000 years and a dose rate for hundreds of thousands of years that is comparable to natural background activity. Although the assessment is conservative by virtue of overestimating the effects leading to release and transport, there are still some uncertainties in the estimates that will require further analysis. The sensitivity of the results to these uncertainties is discussed in Section 6.3.

6.3 SENSITIVITY ANALYSES OF THE REFERENCE DESIGN

One of the principal objectives of any total system performance assessment is to evaluate the significance of the assumptions made in defining the "probable" system behavior. One of the primary goals of this TSPA is to assist DOE in identifying, based on the reference repository design, the principal factors that affect the postclosure performance. A part of this objective is met by examining a range of possible outcomes predicated on a range of conceptual and parameter values. These results are presented in Sections 4.3 and 5. In addition, the possibility of some low-probability, potentially high-consequence disruptive scenarios has also been evaluated (Section 4.4). Finally, the effects of using different design options to compensate for some of the uncertainty in the various components of the natural and engineered system, and thus provide greater assurance of repository performance has been investigated (Section 4.5).

The sensitivity analyses conducted as part of this TSPA are summarized in this section. The summary includes a discussion of general insights gained during the course of the analyses. These insights include not only the direct quantitative sensitivity analyses, but also the understanding gained from a range of comparative analyses. Just as the individual results associated with a deterministic evaluation need to be tempered with realization of the uncertainty in the results, so too should the interpretation of a range of uncertainty analyses. For example, the importance of one component of the system can sometimes be masked by other, seemingly more significant components. It is also possible that the complete range of uncertainty for a particular component in the evaluation has not been fully evaluated. To address this issue, in some instances additional sensitivity analyses were conducted with wider uncertainty bounds. However, some judgment regarding the unquantified uncertainty must be used in assessing the overall significance of the principal factors in the repository safety strategy.

Finally, the significance of individual components at different locations and times must be examined because their importance can change depending on the performance measure used.

Evaluations of repository system performance are sensitive to different factors over different time frames. The analyses indicate no releases for the first several thousand years and the sensitivity studies indicate that this observation holds over a wide variety of variations of the components that could affect performance. The following summary addresses the sensitivities in the longer term. The summary has been broken into three time scales; up to 10,000 years, from 10,000 years to 100,000 years, and after 100,000 years.

6.3.1 Sensitivity Analyses over the First 10,000 Years

Over the first 10,000 years after waste emplacement, the overall performance is controlled principally by waste package failure. During this period, with few exceptions, the waste packages completely prevent water from contacting the wastes. A small fraction (on the order of 1/10 of 1 percent), of the waste packages may develop one or more small openings through the corrosion-resistant metal during the first 10,000 years. These packages would generally be located in areas of the repository either with a higher probability of more aggressive geochemical conditions or containing waste packages with a higher Alloy 22 corrosion rate. Because the reference design contains approximately 10,500 waste packages, some waste packages are anticipated to undergo more rapid degradation than others, simply because of the range in possible corrosion rates.

In addition to these "expected" failures, some low probability of failures may occur as a result of unanticipated events. These include:

- Waste packages that might fail prematurely by improper quality control of the welds or fabrication
- Impurities in the metals, leading to higher corrosion rates
- An unforeseen degradation mechanism acting on the waste package materials

Although procedures will be developed and implemented to minimize or eliminate these early failures, they have been included in this analyses in order to evaluate the sensitivity of the results to them.

Once the waste package has been breached, the next significant component is the amount of the waste form surface that is exposed and in contact with water. The amount of water entering the waste package is assumed to be proportional to the perforation area. For the commercial spent nuclear fuel and naval spent nuclear fuel, the negligible corrosion rate of the Zircaloy cladding minimizes the amount of exposed waste form surface. For the other waste forms, once the package is breached the entire surface area of the waste form is assumed to be exposed. In all of the TSPA-VA analyses, water is assumed to contact all of the exposed waste. This is a conservative assumption, but to do otherwise in a quantitative fashion would require a degree of sophistication that would be difficult to substantiate. For the highly soluble radionuclides that dominate the 10,000-year dose rates, the rate at which fuel is exposed and contacted by water dominates the release rate from the engineered barriers.

The calculated dose rate from radionuclides released from the waste package in this period is influenced by their travel time and by the degree of dilution that occurs as the radionuclides migrate to the biosphere. The travel time depends on the flow velocities in the unsaturated and saturated zones. The climate at Yucca Mountain is expected to change from the present-day dry conditions to a presumed wetter environment associated with what is believed to be the long-term average climate. The timing of this climate change and the net infiltration have a significant effect on the travel time of any dissolved radionuclides released from the engineered barriers in the first 10,000 years. Under present-day conditions, the time for 50 percent of the mass of highly soluble, unretarded species to travel from the repository to the water table is a few thousand years. It takes an additional few thousand years for 50 percent of the mass to travel in the saturated zone from the repository footprint to the assumed compliance point 20 km (12 miles) downgradient. Retarded species take significantly longer than these times. Therefore, if no releases occurred from the engineered barriers for thousands of years, then the present-day climate regime indicates that only a small fraction of the released inventory would be transported 20 km (12 miles) in the first 10,000 years. Invoking a climate change within the first 10,000 years significantly reduces these travel times. As the amount of water moving through the rocks increases, the velocity of the groundwater increases, causing a greater percentage of the groundwater to flow more rapidly through fractures in the nonwelded tuff units.

The effective amount of dilution as radionuclides move through the unsaturated and saturated rocks to the biosphere depends on several things. The dilution is influenced by the degree to which water contacting the waste is mixed with uncontaminated water in the emplacement drifts. It is also altered as the water bearing the radionuclides mixes with the water that has never entered the drift in the rock in and around the drifts, in the path to the water table, and in the paths through the saturated zone. Finally, the degree of dilution is affected by pumping at a well.

As noted in Section 4.1, the flux through the unsaturated zone within each of six different repository regions is assumed to mix the different sources of radionuclides created when each individual package is breached and begins to release radionuclides. In areas of the repository underlain by perched water bodies and/or significant lateral flow this is a reasonable assumption. At times when a significant number of waste packages have been breached and are releasing radionuclides this is also a reasonable assumption given the relatively close spacing of the waste packages. However, there is more uncertainty implicit in this assumption at early times (less than 10,000 years), when only a very few waste packages have been breached. If those breached packages are located in the southern half of the repository (where the flow is predominantly vertical), the degree of transverse mixing that might be anticipated between the breached waste package in the repository and the water table may not have been adequately captured in the model. Quantifying the effect of this uncertainty is not currently possible, given the degree of spatial averaging and numerical discretization used in the unsaturated zone radionuclide transport model. Although the significance of this uncertainty may be minimized by possible mixing in the saturated zone or at the well, it is anticipated that additional analyses will be required before licensing to define the significance of this issue.

In summary, the most significant components in the determination of the dose rates at 20 km (12 miles) from the Yucca Mountain repository over the first 10,000 years are the following:

- Seepage contacting the waste packages and waste forms
- Waste package containment, including effects both of premature waste package failures and of waste package degradation rates
- Rate of cladding failure
- Chemistry of water contacting waste package surfaces
- Travel time, including the effect of timing and the magnitude of climate change
- Degree of mixing in the unsaturated and saturated zones
- Biosphere dose conversion factor

The peak dose rates during this period are forecast to be small. The median (50th percentile) dose is 0.002 mrem/year with the 5th and 95th percentiles of 0 and 0.8 mrem/year, respectively. The mean peak dose rate is forecast to be 0.1 mrem/year. The variations in the forecasted dose results primarily in variations in the first two components. Finally, it should be noted that, in the event of immediate cladding failure, Figure 5-29 indicates that the median dose could be about 1 mrem/year in the 10,000 year time frame.

6.3.2 Sensitivity Analyses from 10,000 to 100,000 Years

It is expected that the potential future regulatory requirements for Yucca Mountain will focus on dose assessments to 10,000 years. However, analyses beyond 10,000 years will be conducted to gain additional insights into repository system behavior. It is likely that the goal of the analyses beyond 10,000 years will be determining the average behavior and its confidence interval. This is different than examining the full range of potential performance (as has been done in Sections 4.3 and 5) that has been used as the basis for defining the significance of the principal factors of the repository safety strategy. Therefore, although useful insights into factors affecting dose rates beyond 10,000 years are discussed below, the degree to which these uncertainties need to be addressed before licensing requires consideration of how these long-term projections will be used in the LA.

From 10,000 to 100,000 years after the repository is closed, the important factors controlling the dose rate at 20 km (12 miles) change from those that were important in the first 10,000 years. The principal reason is the change in the radionuclides that dominate the dose rate. As the period is extended, the controlling radionuclides generally (but not always) change from being the highly soluble, unretarded radionuclides such as technetium-99 and iodine-129 to low solubility, slightly retarded species dominated by neptunium-237. Even at extended times the high solubility radionuclides may dominate the predicted dose rate if any of the following conditions are present:

- The rate of waste package failures is low.
- The amount of seepage into the drifts or into the waste packages is low.
- The solubility of the solubility-limited radionuclides is low.

An additional explanation for the change in key components as the period of interest is extended beyond 10,000 years is the difference caused by the groundwater flow regime. While the advective travel time can make a difference when considering the present-day climate over 10,000 years, over a longer period the nonsorbed or poorly sorbed radionuclides that are released from the engineered barriers will reach the hypothetical well located 20 km (12 miles) downgradient. However, once the climate state switches to being the long-term average precipitation, the net infiltration, percolation flux, and advective velocity all increase significantly. This decreases the travel time for any radionuclides that may be traveling through the unsaturated or saturated zones. Even over this period, the more highly sorbed radionuclides are effectively immobile in the Yucca Mountain environment.

The dominance of specific radionuclides in the dose rate changes over the 100,000-year period. For example in the expected value base case realization presented in Section 4.2, the time at which neptunium-237 begins to dominate the dose occurs about 40,000 years after repository closure. Because of this change, the significant components driving the performance are some combination of those that affect the concentration and dose rate of technetium-99 and those that affect the concentration and dose rate of neptunium-237. The controlling factors are examined separately in the following paragraphs.

High solubility, low retardation radionuclides such as technetium-99 are generally called "release-rate-limited" radionuclides. The release rate from the engineered barriers is controlled by the rates of waste package failure, the surface of the waste form exposed through clad failures and contact with water. The rate of waste package failure is a function of the presence or absence of seeps and the degradation rate of the Alloy 22 metal in the waste package. If the clad is failed and water contacts the waste, the advective or diffusive travel times through the engineered barriers are generally short in comparison to the 100,000-year period of the analyses. This short travel time is, in part, a result of the conservative assumption that the water pathways inside the waste package and the water film along the waste form surface are continuous. Therefore, the advective and diffusive travel times are short for the highly soluble radionuclides released from the engineered system after the waste packages have been breached.

After the low-retardation radionuclides are released from the engineered barriers, their travel times to the 20-kilometer (12-mile) hypothetical well are on the order of several thousand years for the present-day climate. This delay time is reduced if there is a change to a wetter climate. In addition to the delay, the radionuclides released from the engineered barriers are diluted as the water containing the dissolved radionuclides mixes and disperses through both the unsaturated and saturated zones. Because not all the water that enters the drift has entered waste packages, some has diverted around the waste package, this mixing starts as soon as the water containing the dissolved radionuclides leaves the waste package and enters the drift. Because not all the water that intersects the drift has entered the drift, some has diverted around the drift, the mixing continues as the water containing dissolved radionuclides leaves the drift and enters the rock. This mixing effectively reduces the concentrations of the dissolved radionuclides.

The release rate of the less mobile radionuclides such as neptunium-237 is controlled by their solubility in the water. For these radionuclides, the solubility of the radionuclide in the water film on the waste form surface and the advective flux of water over the waste form surface control the release rate from the engineered barriers. For any repository region, the mass release rate for the solubility-limited radionuclides is a function of the cumulative number of waste packages that have been breached and the amount of water that seeps into each waste package. These aspects of the system are, in turn, controlled by the fraction of the waste packages in contact with seepage water, the amount of water that seeps, and the Alloy 22 degradation rate. This fact has been reinforced by the identification of the key parameters in the multiple-realization analyses presented in Section 4.3.

The solubility of some radionuclides, including neptunium, in the water on the waste form surface can be significantly affected by geochemical reactions of that water with the waste form as it alters. Spent fuel may progress through a complex series of alteration states. The different altered forms of the spent nuclear fuel can have neptunium retained in the solid state rather than being released to the aqueous phase. If neptunium is contained within these alteration products, then it will be immobile until the secondary and tertiary alteration phases have dissolved. The effect of these alteration phases on limiting the neptunium-237 dose has been investigated in Section 5.5.

Because the solubility limit is applied separately to each waste package, the total inventory exposed and in contact with water does not make an appreciable difference in the release rate of neptunium-237 from the repository. In part this is caused by the model not explicitly linking the solubility constraints to the clad failure model. Neptunium release remains almost unaffected by cladding degradation until the inventory in the initially breached waste packages has been depleted. With a 2 million year half-life, neptunium-237 does not significantly decay during the period of the analyses. The depletion time is a function of the amount of water that enters each waste package and the percent of the cladding that has been degraded (for commercial spent nuclear fuel). The analyses presented in Section 4.2 and 5.5 show that this time is several hundred thousand years.

Once the solubility-limited radionuclides are released from the waste package and transported through the invert, their transport through the unsaturated tuffs is controlled by the retardation characteristics of the rock. Many solubility-limited radionuclides are very highly sorbed on tuffs (they are essentially immobile in the Yucca Mountain setting). Neptunium-237 is slightly retarded on the tuffaceous rocks within the unsaturated zone. Therefore, in the time periods of interest, neptunium-237 released from the repository will travel through an average of 300 m (1,000 ft) of unsaturated tuffs to the water table. During the long-term climate state, this distance would be reduced to 220 m (720 ft) due to an assumed 80-meter (260-ft) rise of the water table. Eventually, these radionuclides will be transported through the saturated zone to the hypothetical well located 20 km (12 miles) downgradient from the repository.

Another factor that could have an impact on overall performance is the colloidal transport of plutonium. There is significant uncertainty about the nature, stability, filtration, and adsorption and desorption characteristics of naturally occurring and waste form generated colloids. For the anticipated range in colloid properties considered, the dose rate attributed to colloidal plutonium-242 and plutonium-239 is less than the dose rate caused by neptunium-237. However, either expanding the possible range in colloid properties or reducing the neptunium solubility may cause colloidally transported plutonium-242 to be the dominant dose contributor at times in excess of 100,000 years.

In summary, the most significant components in determining the dose rates at 20 km (12 miles) from the Yucca Mountain repository over the first 100,000 years are the following:

- Fraction of waste packages with seepage
- Waste package degradation rate
- Cladding degradation rate
- Dilution and mixing in the unsaturated and saturated zones
- Timing and magnitude of climate changes
- Formation and transport of radionuclide-bearing colloids
- Formation of secondary neptunium mineral phases (related to waste form degradation rate)
- Neptunium solubility (related to the alteration rate of spent nuclear fuel)
- Biosphere dose conversion factor

The median (50th percentile) peak dose rate during this period is forecast to be 0.09 mrem/year, with the 5th and 95th percentiles at 0.0 and 200 mrem/year, respectively. The mean peak dose rate is forecast to be 30 mrem/year.

6.3.3 Sensitivity Analyses for Times Greater than 100,000 Years

At times longer than 100,000 years, neptunium-237 is again the dominant radionuclide. As time proceeds, several events occur:

- The number of degraded waste packages (with at least one opening through Alloy 22 inner barrier) continues to increase
- The amount of degradation (the extent of the openings) through each waste package continues to increase
- The amount of seepage that enters each degraded waste package continues to increase
- The fraction of the waste form surface area exposed (because of cladding degradation) and in contact with water continues to increase

All of these factors tend to cause the release of neptunium-237 from the waste packages to continue to increase with time. This continued increase is illustrated in the base case deterministic results described in Section 4.2. As the releases from the engineered barriers increase with time, the concentrations and dose rates downgradient from the repository increase accordingly.

Eventually the neptunium-237 inventory will be depleted from the degraded waste packages. The length of time for depletion to occur is a function of the neptunium-237 inventory as well as the factors listed in the previous paragraph. For the base case deterministic analyses, this period is several hundred thousand years. After this time, the release rate from the engineered barriers for neptunium-237 decreases, as do the corresponding concentrations and, ultimately, dose rates.

Because the release rate depends on the seepage amount and fraction of waste packages contacting seepage water, climate affects release from the engineered barriers. In particular, during the superpluvial climate, the presumed increase in net infiltration causes an increase in percolation and, therefore, seepage. As a result, at the time of the superpluvial climate, which is assumed to be several hundred thousand years from the present, the dose rate could increase. Although this increase is in part an artifact of the presumed instantaneous change in the climatic regime from the long-term average to the superpluvial climate, some increase would be expected when the seepage increases.

As noted in the discussion of the 100,000-year sensitivity analyses, the travel time of the nonsorbed and partially sorbed radionuclides from the repository to the water table and then to the 20 km (12 mile) hypothetical well is short compared to the overall period of the analyses. However, some dilution occurs when the water containing dissolved radionuclides is released from the engineered barriers and mixes with the moving groundwater in the rock. Although not considered in the base case analyses, additional dilution could occur if the hypothetical well pumps more water than contained within the maximally contaminated zones of the aquifer.

In summary, the most significant components in determining the dose rates at 20 km (12 miles) from the Yucca Mountain repository over the first million years are the following:

- Fraction of waste packages contacting seepage water
- Waste package degradation rates
- Cumulative amount of degraded cladding
- Dilution in the unsaturated and saturated zones
- Biosphere dose conversion factors

The median peak dose rate calculated in this period is 8 mrem/year. The 5th and 95th percentiles are 0.07 and 1000 mrem/year, respectively. The mean peak dose rate is forecast to be 200 mrem/year.

6.4 PRINCIPAL FACTORS AFFECTING POSTCLOSURE PERFORMANCE

The principal factors affecting the repository system are those factors that bear directly or indirectly on one or more of the components identified in the sensitivity studies as being potentially important to performance. These factors are summarized in Table 6-1. They are grouped according to the system attributes of the *Repository Safety Strategy* (DOE 1998).

The major insights gained from the sensitivity analyses for the reference design are summarized in Section 6.3. These results have been interpreted to guide the definition of the information required to enhance confidence in the performance-assessment analyses for the LA. Each of the principal factors affecting postclosure performance was examined to determine whether that factor has a high, medium, or low contribution to the overall system performance.

The significance of the uncertainty in the principal factors on the assessment of the postclosure system performance may be identified in several different ways. One method is to examine the multiple realization performance analyses and their corresponding regression analyses presented in Section 4.3 to identify which parameters within the TSPA-VA base case simulations are most significant. This is an appropriate technique and provides very useful insights into what is driving the system behavior, but is predicated on the degree of uncertainty included in the base case models and parameters. However, it is possible that a few very uncertain parameters will mask the potential contribution to the uncertainty of other parameters. Therefore, this method can not be used in isolation or without examining the contribution of the other parameters to the system performance.

In order to examine the significance of alternative models and to examine more explicitly the role that some parameters may have in the system performance, a range of comparative analyses have been presented for each of the TSPA-VA model components in Chapter 5. These comparative analyses include sampling the uncertain parameter at

Table 6-1. Significance of Uncertainty in Principal Factors on Post Closure System Performance - Summary of Sensitivity Analysis for Viability Assessment Reference Design

Key Attribute of the Repository Safety Strategy	TSPA Model Components	Principal Factor for the Repository Safety Strategy	Significance of Uncertainty in TSPA[1] Performance Period (years) 10k	100k	1M	Reference Figure Location Base Case	Alternative Models
Limited water contacting waste packages	Unsaturated Zone Flow	Precipitation and infiltration into the mountain	L	M	L	5-5	5-1, 5-2, 5-3
		Percolation to depth	L	L	L	5-5	5-6, 5-7
		Seepage into drifts	H	H	H	4-34, 5-8	NA
	Thermal Hydrology	Effects of heat and excavation on flow (mountain-scale)	NA	NA	NA	NA	NA
		Effects of heat and excavation on flow (drift-scale)	L	M	L	NA	5-9, 5-10
		Dripping onto waste package	L	L	L	NA	5-18
		Humidity and temperature at the waste packages	L	L	L	NA	5-11
Long waste package lifetime	Near-Field Geochemical Environment Waste Package Degradation	Chemistry of water on waste package	H	L	L	NA	5-14, 5-22
		Integrity of the outer carbon steel waste package barrier	NA	NA	NA	NA	NA
		Integrity of the inner corrosion-resistant waste package barrier	H	H	M	4-34, 5-16	NA
Low rate of release of radionuclides from breached waste packages	Waste Form Alteration and Mobilization	Seepage into waste packages	L	L	L	5-26	5-27
		Integrity of spent nuclear fuel cladding	H	M	M	5-28	5-29, 5-30, 5-31
		Neptunium solubility	L	M	L	5-32	5-33
		Dissolution of uranium oxide and glass waste forms	L	M	L	4-40, 5-35	NA
		Formation and transport of radionuclide-bearing colloids	L	M	L	5-36	NA
		Transport through and out of the engineered barrier system (including waste packages)	L	L	L	5-37	5-38
Radionuclide concentration transport from the waste packages	Unsaturated Zone Transport	Transport through the unsaturated zone	L	L	L	5-42	5-41, 5-43, 5-44, 5-45
	Saturated Zone Flow and Transport	Flow and transport in the saturated zone	M	M	M	4-34, 4-40	5-47, 5-48, 5-49
		Dilution from pumping	H	H	H	NA	5-51
	Biosphere Transport	Biosphere transport and uptake	M	M	M	4-34, 5-50	5-52

[1]H —High Significance (Uncertainty in principal factor, or its absence, results in a factor over 50 increase or decrease in peak dose rate from the "expected" value)
M—Medium Significance (Uncertainty in principal factor, or its absence, results in a factor of 5 to 50 increase or decrease in peak dose rate from the "expected" value)
L—Low Significance (Uncertainty in principal factor, or its absence, results in a factor less than 5 increase or decrease in peak dose rate from the "expected" value)

discrete probabilities (principally the 5th and 95th percentiles of the distribution as well as the mean of the distribution for comparison to the expected-value realization described in Section 4.2). The results of these analyses confirmed the significance of the parameters identified in the regression analyses described in Section 4.3 and also provided a method to rank the relative contribution of each of the principal factors.

The ranking of the significance of the principal factors presented in Table 6-1 has therefore been based on both the multiple realization regression analyses and the one-off comparative analyses. Where both methods indicated the same high significance, the decision to label that factor as having a high significance was straightforward. Where the two methods gave differing results or, more commonly, where the significance based on multiple realizations was masked by more significant factors, the factor was assigned the significance based on the comparative analyses. In some cases the comparative analyses included alternative conceptual models which, if true, could lead to substantially improved or degraded performance. The results of calculations using these alternative representations (presented in Chapter 5) were also used to adjust the rankings, as appropriate. This was done to ensure that the assessment of the significance of the each principal factor is reasonably complete, is defensible and that it accommodates the full breadth of current understanding as well as the factor's potential effect on postclosure performance.

The analyses of significance have been conducted over different time scales to reflect that the behavior of the system changes with time as does the significance of the principal factors of the repository safety strategy.

Table 6-1 focuses on the principal factors affecting the expected or probable repository postclosure performance. Other factors relating to potentially disruptive processes and events have been described in Section 4.4. These consequence analyses, when combined with the understanding of the probabilities of these disruptive scenarios occurring, implies that the long-term risks to public health and safety associated with these scenarios are minimal. Therefore, the uncertainties in these processes and events (seismicity, volcanism, nuclear criticality, and human intrusion) are deemed to have a low significance on overall performance.

An evaluation of significance requires combining both objective and subjective assessments of appropriate information. To provide objective evidence of significance, a range of sensitivity analyses have been performed. These sensitivity analyses have been used to define significance with respect to the magnitude of the effect of the uncertain parameter or model on the dose rate at a hypothetical well located 20 km (12 miles) downgradient from the repository. This section reports the results of those analyses in terms of "high," "medium," or "low" sensitivity to uncertainty. For the assessment of uncertainty given in this section and in Table 6-1, the following definitions were used to assign a ranking of high, medium or low. To rate a particular factor as having high significance on dose rate, varying the range between the mean and the 95th percentile of that parameter causes a change of over 50 times in the dose rate. In addition, if the absence of such a factor could result in a change in the projected dose of a factor of over 50, it also would be deemed to have a high significance. A medium significance corresponds to a change of 5 to 50 times the dose rate and a low significance indicates a change of less than 5 times the dose rate. These criteria were applied over the range of time periods of concern: 10,000 years, 10,000 to 100,000 years, and 100,000 to 1 million years.

Objective measures of significance are useful, but they presuppose that the uncertainty in the parameter or model is well characterized. One source of uncertainty may arise from the fact that not all possible alternative conceptual models have been considered. Also, the models themselves may have inaccuracies that are not captured in the analyses. For example, a parameter may impact the results significantly over a certain range of values, but outside that range the results are insensitive to the value of that parameter. If the parameter range does not extend into the significance range, then the analyst may falsely conclude that the parameter is not important. Therefore, judgment must be used to assign significance to a particular parameter or model because a range and synergistic effects that may not have been fully evaluated in the sensitivity analyses can be an exercise in judgment. The determinations of significance in the following paragraphs have been based on objective criteria. However, the influence of judgment in modifying the significance ratings for each of principle factors is given in Section 2.2-1 of Volume 4. The changes in the values that have resulted from the incorporation of judgment are shown in Volume 4, Table 2-2.

6.4.1 Precipitation and Infiltration of Water into the Mountain

The amount of water that enters the unsaturated zone at the surface controls the amount of water that percolates to depth and, ultimately, the amount and location of seeps into the repository drifts. Generally, the lower the infiltration, the less seepage into the drifts and the less water that can contact the waste package and the waste form. Therefore, the time and location variability in precipitation and infiltration are significant factors in overall system performance. The significance of this principal factor is estimated to be higher in the period from 10,000 to 100,000 years because the solubility-limited radionuclides control the dose rate in this period. The release of these radionuclides from the engineered barriers is, in large part, a function of the volumetric flow rate into the drifts and into the waste packages. The net infiltration has a moderate significance on the projected dose rate based on variation in the range of plausible infiltration rates (Section 5.1).

6.4.2 Percolation to Depth

Percolation flux is closely correlated with net infiltration. The uncertainty in percolation flux is directly related to the uncertainty in net infiltration. Percolation flux controls the fraction of the repository with seeps and the seepage amount. Therefore, percolation is likely to have an effect similar to infiltration on the dose rate. However, the base case analyses only showed a low significance of this factor, based on the fracture flow properties, which was the only parameter related to percolation varied in this set of analyses (Section 5.1).

6.4.3 Seepage into Drifts

Seepage into the drifts is a significant factor for many reasons. Seepage controls waste package degradation because Alloy 22 only corrodes in the presence of liquid water. Following the creation of openings through the Alloy 22, the seepage volume controls the amount of water that can enter the waste package and dissolve the waste form. The flux of water into and through the waste package in turn controls the release rate of the solubility-limited radionuclides from the waste package. The high significance of seepage at all time periods can be ascertained from the Monte Carlo analyses presented in Section 4.3 (where seepage fraction or seepage amount were consistently in the top four of the most significant variables) and in Section 5.1.

6.4.4 Effects of Heat and Excavation on Flow (Drift Scale)

Heat can drive water away from the emplacement drifts and enhance flow because of condensation and refluxing in the earliest time period. Therefore the heat can modify waste package corrosion rates and other aspects important to performance. However, the effects of heat are shown to only have a low impact on repository performance in the period up to 10,000 years. In the later period, the heat will have diminished and its direct impact on the redistribution of flow will also be minor. However, heat can cause permanent effects on the flow system. For example, the heat can modify flow properties by altering minerals in fractures and creating mineral caps above the heated area, similar to those observed at geothermal sites. Likewise, excavation can alter the flow properties, increasing bulk permeabilities near the drifts. These alterations have been approximated by changing the fracture properties contained in the seepage sensitivity analyses described in Section 5.1. This aspect of altering properties of the host rock leads to an assessment of moderate importance to performance in the middle time period, as illustrated in the sensitivity analyses described in Section 5.1.

6.4.5 Dripping onto Waste Packages

In the current analyses, any seep located half way up the drift wall is conservatively assumed to directly hit the waste package. However, many seeps may simply form a film around the drift wall or fall on rubble around the waste package and not directly intercept the waste package. This diversion of water may be significant, but is believed to be less significant than the presence of the seep itself. Variations in the surface area of the waste package that becomes wetted show this factor to be of low significance to performance.

6.4.6 Humidity and Temperature Effects on Waste Packages

Corrosion of the carbon steel layer on the outside of the waste package can be initiated and sustained at values for relative humidity greater than about 70 percent and temperatures less than about 100°C (212°F). Although most of the waste package performance is caused by the very slow corrosion rates of the Alloy 22 inner barrier of the waste package, during earlier periods (less than 10,000 years) the outer barrier does provide some protection. However, the overall significance of this factor is shown to be low during all time periods (Section 5.2).

6.4.7 Chemistry Effects on Waste Packages

The chemistry of the water on the waste package surface can significantly affect the degradation rate of both the carbon steel material of the outer waste package and the inner corrosion-resistant layer. As noted in Section 3.4, the degradation rate of Alloy 22 under benign water compositions is less than the degradation rate under more aggressive chemical conditions. The presence of microbes also changes the chemistry, and thus the corrosion rates. In addition, the chemistry of the water can affect both the degradation rate of the waste form and the solubility of neptunium-237. This factor, therefore, has a high impact on performance during the first 10,000 years when water is modified as it travels through the concrete liner, but drops to low significance at later times (Sections 5.3 and 5.4).

6.4.8 Integrity of Inner Waste Package Barrier

In the reference design, the corrosion degradation rate of the Alloy 22 corrosion-resistant alloy is seen as one of the dominant factors affecting the postclosure performance. The effect is illustrated in the regression analyses presented in Section 4.3 and the comparative analyses presented in Section 5.4. Not only does the degradation control the time at which the initial opening is formed through each waste package, but it also affects the number of openings generated through each waste package. The number of openings significantly affects the amount of seepage that can enter each waste package, which in turn controls the advective release of solubility-limited radionuclides, notably neptunium-237, from the waste package. This factor has a high significance over all but the longest time frames in the analyses of reference design postclosure performance, as documented in Section 5.4.

6.4.9 Seepage into Waste Packages

After the waste package has failed because of at least one opening through the corrosion-resistant inner container, the amount of water that enters the waste package controls the advective release of solubility-limited radionuclides. However, changing the percentage of waste packages wetted and changing the resulting flux through the waste package only showed a small change in dose rates, resulting in an assignment of low for significance (Section 4.2).

6.4.10 Integrity of Spent Fuel Cladding

The degree of degradation of the cladding significantly affects the amount of the inventory potentially exposed to liquid water. The dose rates of the high-solubility radionuclides, such as technetium-99, depend on the fraction of exposed inventory. Because the dose rates during earlier periods are controlled by the high-solubility radionuclides, the significance of cladding is greater at early times. The significance of cladding is high for the first 10,000 years and relatively less significant, but still of moderate importance at late times.

6.4.11 Dissolution of UO_2 and Glass Waste Form (Waste Form Integrity)

The alteration rate of the different waste forms is relatively rapid (thousands of years) compared to the period of interest (tens to hundreds of thousands of years). Therefore, the dissolution rate is not a significant factor in the overall dose evaluation at early times. Before 50,000 years technetium is the most important contributor to dose, after that time neptunium becomes the dominant contributor. However, the possibility that secondary phases are created at the spent nuclear fuel can effectively reduce the mobile concentration of neptunium-237. Therefore, this factor has the potential to be moderately important in the period between 10,000 and 100,000 years. After the secondary phases have been dissolved, the effect is mitigated, resulting in a reduction of the significance at times greater than 100,000 years.

6.4.12 Solubility of Neptunium-237

Neptunium solubility is a moderately significant contributor to the long-term dose assessment. The contribution of neptunium-237 is less significant at earlier times when the doses are primarily attributed to more mobile radionuclides such as technetium. However, at times greater than several tens of thousands of years, or in cases where the seepage flux is high, neptunium solubility can be a significant factor in postclosure performance projections. The solubility of the secondary phases of the fuel/glass is also important, because it may lead to a delay or reduction in the overall release of neptunium at times less than 100,000 years.

6.4.13 Formation of Radionuclide-Bearing Colloids

Plutonium colloids are responsible for the peak dose in a significant number of the TSPA-VA realizations during later times. It is one of the more significant findings in TSPA-VA that, under certain conditions, colloid-facilitated transport is moderately important to performance in the time period from 10,000 to 100,000 years.

6.4.14 Transport Within and out of the Engineered Barrier System

In the current TSPA analyses, if water gets into the waste package it was conservatively assumed to contact the entire exposed waste form surface. No credit is taken for the fact that a drip entering the waste package at one opening has a low likelihood of encountering the waste form at the other end of the waste package. In addition, the time it may take radionuclides to diffuse along a thin water film layer within the waste package was not considered. Quantifying the effects of these conservative assumptions is difficult. Therefore, the transport of radionuclides through the engineered system elements beneath the waste package (sorption in the invert) for any time period was shown to be of only low significance to performance.

6.4.15 Transport Through the Unsaturated Zone

Based on the current unsaturated zone flow model, the advective travel time through the unsaturated zone depends on the percolation flux distribution that is in turn controlled by the climatic conditions. Under the current dry climate, the unsaturated zone may provide a significant barrier to the migration of highly and slightly sorbed radionuclides. Under climate states with higher percolation flux, only the highly sorbed radionuclides are still retained in the unsaturated zone. Therefore, the significance of this barrier to release to the saturated zone may change with time as the climate state varies. In addition to the transport time, the unsaturated zone dilutes the concentration of the radionuclides released from the engineered barriers. The effects of unsaturated zone transport were investigated by changing sorption coefficients, assigning a value of zero for matrix diffusion, and assigning zero for sorption coefficient of the actinides (Section 5.6). However, the range investigated for these parameters only showed this factor to be of low significance at all periods.

6.4.16 Transport in the Saturated Zone

The time it takes for radionuclides to advectively travel through the saturated zone to a distance of 20 km (12 miles) is a significant fraction of the period of interest for 10,000 years. This is especially true for those radionuclides sorbed on the alluvial sediments. For longer periods and during climate regimes characterized by the long-term average climate, travel times in the saturated zone may not provide as much of a barrier to radionuclide migration. The saturated zone also provides some dilution of those radionuclides released from the unsaturated zone. The TSPA-VA sensitivity analyses of the saturated zone flow and transport has a moderate significance relative to other factors (as noted in Sections 4.3 and 5.7) for the three time periods.

6.4.17 Dilution from Pumping

If the volume of water extracted in the hypothetical well is significantly greater than the volume of water within an individual stream tube in the saturated zone, then significant dilution can occur at the well head. The effect of this dilution would diminish as the natural dilution in the saturated zone is increased. No credit is taken for the pumping dilution in the base case analyses of the reference design. The significance of this dilution depends on the size of the pumping well and the magnitude of the natural dilution in the saturated zone. Depending upon the assumptions concerning the amount of water pumped, the dilution factors investigated showed, at all times, a high significance in changing the dose to the average members of the critical population (Section 5.8).

6.4.18 Biosphere Uptake

After water containing dissolved radionuclides is extracted from the hypothetical well, the projected dose rate is a function of the range of possible water uses and food consumption habits of the exposed population. Although there is a linear relationship between the dose conversion factor and the dose rate (for a given radionuclide concentration at the well head), the changes in the dose rates show this factor to be of moderate importance in all time periods. (Sections 4.3 and 5.8).

6.5 IMPROVING CONFIDENCE IN THE TOTAL SYSTEM PERFORMANCE ASSESSMENT FOR THE LICENSE APPLICATION

The data and analyses summarized in Section 6.1-6.4 of this volume comprise a portion of the information needed to construct a complete postclosure repository safety case. In order to provide reasonable assurance that a repository at the Yucca Mountain site will not result in a significant long-term risk to public health or safety, the postclosure safety case must include:

- Explicit evaluation of expected repository performance
- Analyses of the degree of design margin and defense in depth that could improve performance and mitigate uncertainties in performance
- Explicit consideration of processes and events that, if present, have the potential to disrupt the repository system
- Supporting information regarding long-term behavior from natural and man-made analogs
- Plans for long-term testing and monitoring of the repository system

The purpose of the TSPA-VA described in this volume has been to examine the current understanding related to the first and third elements of the safety case, that is, the expected repository performance and the process and events that, if present, have the potential to disrupt the expected repository performance. The probable performance has been described in the context of how the four elements of the repository safety strategy work in concert to first minimize the contact of water with the waste and then reduce the concentration of any radionuclides that are released from the engineered barriers.

Based on the results of the TSPA analyses presented and discussed in this volume, areas of potential improvements in the analyses can be identified. Possible model enhancements and analyses that have the potential for improving the confidence in the TSPA for the site recommendation and LA are provided in Section 6.5.1. In addition, many important observations and insights have also been provided by independent groups, most notably by the TSPA Peer Review Panel and NRC. These comments are discussed in Sections 6.5.2 and 6.5.3, respectively. Section 6.5.4 provides some concluding remarks.

6.5.1 Assessment of Potential Activities to Increase the Confidence in the Total System Performance Assessment Based on the Results of the Total System Performance Assessment for the Viability Assessment

For each of the principal factors associated with the TSPA model components, the potential significance of some of aspects of model uncertainty has not been investigated in the TSPA-VA. This is in part a function of the objective of focussing the analyses on the probable behavior of the repository system which implies evaluating the expected performance as opposed to the performance associated with all the possible alternative hypotheses and models. The sensitivity analyses presented in Section 4.3 and Chapter 5 provide an indication of the possible range of performance, but can not be construed as being exhaustive. It is expected that the TSPA analyses developed for the site recommendation and LA will address these remaining uncertainties. The following paragraphs present possible model enhancements and analyses that have the potential for improving the confidence in the future assessments of repository performance.

6.5.1.1 Precipitation and Infiltration into the Mountain: Unsaturated Zone Flow

- *Abruptness of transition of climate changes.* The relation between the magnitude of the dose-rate "spikes" and the time of transition from one climate to another is not known. It is possible that noninstantaneous transitions would lead to lower peak dose rates.

- *Climate analogs.* The current approach is to base infiltration modeling on appropriate climate analogs (i.e., places that currently have climate conditions like those expected at Yucca Mountain for the future climate). The approach could be improved by taking temperature and other factors into account in addition to precipitation when defining the analogs.

- *Timing and duration of climate states.* The current model, using three distinct climate states, is simplistic and without a strong basis. Developing a more defensible basis for the number, timing, and duration of climate states could enhance confidence in the model.

- *Infiltration uncertainty.* A more quantitative basis for the uncertainty distribution for net infiltration (i.e., the probabilities in Table 3-5) could be obtained by running the infiltration model in a probabilistic mode (e.g., using Monte Carlo simulation, which is described in Section 4.3) to derive the infiltration uncertainty from the uncertainties in the model input parameters.

- *Infiltration for future climates.* Estimates of infiltration for future climates could be improved by explicit inclusion of processes that should be different for future climates, including effects of temperature, cloudiness, vegetation type, surface water runoff/run-on, and snow cover. Even for current conditions, some experts in the unsaturated zone flow model expert elicitation (CRWMS M&O 1997o) suggested that runoff and run-on might be more important than is assumed in the current infiltration model.

- *Model testing.* Confidence in the infiltration model could be enhanced by testing it using analogs with better-known infiltration, such as Rainier Mesa and Apache Leap.

6.5.1.2 Percolation to Depth: Mountain-Scale Unsaturated Zone Flow

- *Localized flow channeling.* An important uncertainty in the mountain-scale unsaturated zone flow modeling is the effect of localized channeling of flow, and in particular, the effect of flow in discrete fractures. Current modeling uses continuum models with very coarse spatial discretization, and the adequacy of this approach is not fully established. There are indications from geochemical and isotopic tracers (chloride concentration, chlorine-36 to chlorine ratio and carbon-14 to carbon ratio) that channeling of flow might be important. Better integration of geochemical, isotopic, and temperature data could improve the calibration procedure because they provide important information about flow through fractures.

- *Perched water.* The role of perched water in unsaturated zone flow is uncertain. The current model assumes that the water is perched on a very-low-permeability underlying layer and flow is forced to go around it. Other interpretations are possible, such as mixing within the perched water and matrix flow out the bottom.

6.5.1.3 Seepage into Drifts

- *Model testing.* Confidence in the seepage model could be enhanced by more comparisons between field data and model predictions. The Exploratory Studies Facility niche test is an important first step, but it is primarily a test of the overall conceptual model of the drift opening acting as a capillary barrier. The test offers little validation of the calculated values of seepage fraction, which the TSPA results show to be the most important aspect of seepage, indeed, the most important aspect of repository performance. Seepage fraction, or the fraction of waste packages contacted by seepage water, is related to the average spacing of seeps along the drift, which is presumably related to quantities such as fracture and fault spacing, permeability distribution, and permeability correlation length. Field data relating these quantities to seep spacing, possibly from analog sites such as Rainier Mesa or Apache Leap, would lend confidence to the model.

- *Localized flow channeling.* Even more so than for mountain-scale flow, seepage into drifts is potentially strongly affected by channeling of flow and discrete-fracture effects. The adequacy of the current fracture-continuum model to represent these effects is uncertain.

- *Stability of seep locations.* A potentially important issue that was not addressed in the TSPA-VA is the stability of seep locations over time. In the present models, seeps are assumed to occur at the same locations indefinitely, so that a fraction of the waste packages (the seepage fraction) is always wet and the rest are always dry. If seep locations changed with time, more waste packages would be contacted by seeps, but only for a fraction of the time. This effect could result in more waste packages failing, but over a longer period of time, which could be important for performance.

- *Drift collapse and thermal alterations of hydrologic properties.* The effect of drift collapse on seepage has not yet been investigated. Also, thermal-hydrologic-chemical or thermal-hydrologic-mechanical alteration of hydrologic properties around the drifts could have an important effect on seepage.

- *Episodic percolation.* The potential for episodic percolation pulses at the repository is uncertain. Also, drainage of thermally mobilized water could potentially cause an increase in seepage for a period of time.

6.5.1.4 Effects of Heat and Excavation on Flow: Mountain- and Drift-Scale Thermal Hydrology

- *Conceptual model of flow.* Differences have been found between results using the dual-permeability flow model and the equivalent-continuum flow model. The dual-permeability model allows for greater mobility of water in fractures, which has a great effect on modeled condensate buildup and drainage. A related flow issue, which could be important to thermal hydrology, is channelized flow, especially in discrete fractures. Such flow could greatly increase the spatial variability in the results by increasing the range of water-flow rates seen by individual waste packages.

- *Coupled processes.* A potentially important shortcoming of the thermal-hydrologic calculations is the lack of coupling to thermal-mechanical and thermal-chemical processes. These issues have been addressed to a limited extent (see Section 3.2.1), but the full range of possible (or even likely) behaviors has not been analyzed. The Near Field Environment Expert Elicitation (CRWMS M&O 1998d) will aid in considering these issues, but it was not completed until after the TSPA-VA analyses were finished. Therefore, implementing the recommendations from the expert elicitation remains to be done.

- *Thermal alterations of hydrologic properties.* Thermal alterations of flow and thermal-hydrologic-chemical or thermal-hydrologic-mechanical alterations of hydrologic properties are potentially important. In the current TSPA structure, these effects fall under the thermal-hydrology component, but thermal-hydrologic effects could be more closely coupled with mountain-scale unsaturated zone flow and transport if necessary.

- *Infiltration rate.* As shown in Section 3.2.3, infiltration rate influences computed repository temperatures. Within the range of infiltration uncertainty incorporated in the base case, the changes in temperature are not large. However, as shown in Section 5.1.3, temperature changes can still have a significant effect on computed dose rates during approximately the first 20,000 years.

- *Matrix hydrologic properties.* Almost all of the available data on matrix hydrologic properties have been obtained from drying experiments. For modeling the thermal-hydrologic behavior during rewetting and condensate drainage, data obtained from wetting experiments would be more appropriate. Wetting and drying properties can be very different because of hysteresis. The thermal-hydrologic property set mentioned in Section 3.2.3 was intended as a better representation of wetting hydrologic properties. It was found to make a significant difference to waste package failure and calculated doses for approximately the first 20,000 years, especially in the part of the repository that lies in the Topopah Spring lower nonlithophysal hydrogeologic unit (primarily Region SW of Figure 3-20).

- *Thermal response of different hydrogeologic units.* The single heater test and the heated-drift test are both located in the Topopah Spring middle nonlithophysal hydrogeologic unit. However, in the current repository design most of the emplacement drifts are in the Topopah Spring lower lithophysal unit, with some drifts in the middle nonlithophysal unit and others in the lower nonlithophysal unit. Confidence in the models would be enhanced by obtaining thermal-response data in the other two units.

- *Continued in-situ testing of thermal hydrologic models–single heater test.* The small-scale, single heater test at the Exploratory Studies Facility has provided valuable thermal-hydrologic information, including that the dual-permeability flow model and hydrologic properties used in the base case fit the measurements reasonably well, at least for the Topopah Spring middle nonlithophysal hydrogeologic unit. The single heater test results helped to guide the selection of hydrologic-property sets thermal-hydrologic models using the base case hydrologic properties match the test results better than an earlier, preliminary-base case property set (CRWMS M&O 1998i, Sec. 3.4.5).

- *Continued in-situ testing of thermal hydrologic models–drift scale test.* An important source of new information for improving thermal-hydrologic models will be the heated-drift test at the Exploratory Studies Facility. This test will provide information on drift-scale movement of heat through rock at Yucca Mountain and its impact on the flow system above and below an emplacement-sized drift. The test will include a detailed investigation of the heating period and movement of heat-driven water as well as the cooling period and subsequent rewetting analogous to the processes that would occur in the repository. Indirect measurements to detect water flow into the drift during heating will provide crucial information related to thermal refluxing processes driven by larger-scale heat-transfer processes. The data obtained from the heated-drift test will:

- Allow important verification of the conceptual flow models currently being used (or show the degree to which the current models are not adequate).

- Provide information on the effective hydrologic properties during the various stages of heating and cooling.

- Provide information on the spatial and temporal extent of mechanical and chemical changes to the fracture-flow system surrounding the heated drift. It is important to note, however, that only the results of the early heating period of the heated-drift test will be available in time for the TSPA for the LA; information on the cooling period will not yet be available.

6.5.1.5 Chemistry of Water on Waste Package: Near-Field Geochemical Environment

- *Effects of concrete-modified water.* The potential impacts from concrete-modified water compositions assessed in the sensitivity results discussed above and in Sections 5.4 and 5.6 appear to be greatest for the 10,000-year time period. The greater effects during the first 10,000 years are mainly caused by impacts to waste package performance because earlier and more frequent failures allow greater exposure of the waste form inventories at earlier times. These results suggest that further consideration be given to substituting other ground support materials for the concrete in the VA design. However, there are uncertainties within the models currently employed to evaluate the near-field geochemical environment. To better constrain changes from materials evolution in the drift, further improvements could be made in the following areas:

  –Thermal-chemical data (both equilibrium and kinetic) for phases in the cement system, in particular at the higher temperatures expected for this system

  –Gas composition changes in the unsaturated, thermally perturbed system

  –Development and implementation of a two-phase, reactive transport model of concrete alteration

- *Precipitate and/or salt build up in the waste package.* Alternate conceptual models for water in the drifts and on the waste package may need to be considered. Similarly, preliminary bounds on biomass production suggest that microbial growth may be nutrient limited within the potential drifts for long periods.

- *Thermomechanical-hydrochemical coupling.* Coupled models would benefit from incorporation of more details of drift materials, both their physical-mechanical evolution and chemically-induced changes to the hydrologic properties of the engineered materials. These aspects would improve the description of water flow and radionuclide transport pathways and allow for development of more specific near-field geochemical environment scenarios.

- *Conceptual model of the near-field geochemical environment.* The updated near-field geochemical model could consider such topics as:

  – Heterogeneity of water composition flowing through the fracture system and interacting with the drift environment

  – Explicit CO_2 evolution from water and minerals coupled to the gas flow in the thermohydrologic system

– Explicit coupling of flow and geochemical reactions

– Thermal aging of all emplaced materials in dividing concrete

– Biomass production and specific microbial activity for local waste package corrosion effects

6.5.1.6 Waste Package Degradation

- *Chemical and electrochemical conditions.* The uncertainty in the current Alloy 22 corrosion model is mostly caused by the uncertainty in the local chemical and electrochemical conditions on the inner barrier and limited data on the long-term behavior of this material in the expected environment of the repository.

- *Corrosion rate of Alloy 22.* The Alloy 22 corrosion uncertainties include the general corrosion rate, localized (pitting and crevice) corrosion rate, localized corrosion initiation threshold, and understanding of the pitting and crevice corrosion stifling process.

- *Potential salt build up.* For waste packages under dripping conditions, the potential exists for salt deposit buildup on the package surface, which could produce concentrated salt solutions near the deposit. This condition could enhance corrosion of the carbon-steel outer barrier, thereby exposing the inner barrier to corrosive conditions earlier than the case without salt deposits. Long-term corrosion of carbon steel under dripping and in the presence of salt deposits is uncertain.

- *Potential galvanic coupling.* The effect of potential galvanic coupling between the Alloy 22 inner barrier and the carbon-steel outer barrier is also uncertain. The most important issue with this process is hydrogen embrittlement of Alloy 22, which could result from hydrogen pick-up by Alloy 22 over a long period and subsequent hydride precipitation inside the alloy, thus potentially shortening its lifetime. Potentially enhanced corrosion of the outer barrier from a galvanic coupling with the inner barrier is another area of uncertainty.

- *Incomplete annealing of welds.* Strengthening of the closure weld could lead to incomplete annealing of the weld. In this case, the closure weld could be subject to stress corrosion cracking.

- *Microbiologically influenced corrosion.* Microbiologically influenced corrosion was not included in the base case in part because the potential for this corrosion of the waste packages was discounted in the recent expert elicitation (CRWMS M&O 1998b). However, the process has uncertainty and could affect long term performance.

- *Long-term structural integrity.* The long-term structural integrity of the waste package is uncertain as to timing and effect. After substantial progress of degradation and under static loads from rockfall, the waste package is expected to lose its structural integrity and collapse, no longer providing a physical barrier to water ingress and radionuclide release. The threshold for waste package structural failure is currently unknown and has a potential impact on long-term repository performance.

6.5.1.7 Waste Form Alteration and Mobilization Models

- *Degradation of invert.* The invert is assumed not to degrade with time but to retain the same transport characteristics for the duration of the analyses. However, the chemistry of the invert will probably change as the system is heated during the thermal period. Heating may alter the transport characteristics of the invert. Likewise, the invert sorption characteristics are not well known and are under study. The impact of more complex invert analysis on overall system performances is not expected to be significant because of the small transport length involved relative to the total transport length.

- *Degradation of drifts.* The base case does not directly account for rock falling into the drift at later times, which is an expected condition. Although the cladding model evaluates the effects of rockfall on the cladding, the rest of the analyses do not incorporate rockfall at later times. A sensitivity to rockfall alteration of the thermal-hydrologic characteristics has been presented in Chapter 3 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

- *Secondary phases.* The effect of the secondary phases on subsequent radionuclide transport has been initially analyzed in Section 5.5 of the TSPA-VA. Neither the dissolution rate of the secondary phase nor the identity of radionuclides that will become fixed in the secondary phases has been well characterized. This process could have the effect of delaying radionuclide releases.

- *Seepage into the waste package.* The amount of seepage into the waste package and onto the waste form is a significant factor in releases from the engineered barrier system. How much water actually enters the waste package once it is breached is uncertain.

- *Transport resistance of failed Zircaloy cladding.* Diffusive transport out of a locally failed fuel rod is not explicitly considered and could significantly reduce release rates.

- *Chemistry-dependent solubility limits.* Additional modifications to the waste form model to improve confidence in the model include solubility-limited radionuclide distributions that are explicitly dependent on the pH and the total water composition as indicated by changing phase relations in the chemical system.

- *Chemistry-dependent sorption in the engineered barrier system.* Identifying sorption mechanisms and evaluating the dependence of sorption on the composition and mineralogy of the specific material would help to better incorporate sorption processes into the model.

- *Cladding degradation.* The current version of the cladding model provides benefit to performance but is based on assumptions about cladding degradation modes and rates. Potential cladding-failure mechanisms, such as localized corrosion and their rates, are important remaining uncertainties in the cladding analyses.

- *Water contact of waste form.* In the TSPA-VA analyses, the entire waste form surface is assumed to be exposed to aqueous conditions after the cladding fails. Detailed analyses of possible alternative models of water flow in the degraded waste package, including alternative conceptual models such as "bathtub" model would aid in evaluating the significance of the water contact mode.

- *Use of natural analogs.* The analysis of numerous uncertainties in the models, data sets, and assumptions for predicting repository performance would benefit from the use of natural analogs. Waste form degradation and mobilization and transport of radionuclides are very long-term phenomena that are difficult to evaluate with a high degree of certainty, especially if limited to information gained through short-term laboratory and field studies. Natural formations and deposits that have existed for very long periods of time show how these processes have taken place and the factors that most affect the processes. The natural "reactors" at Oklo in Gabon, Africa, have been studied extensively to evaluate the mobilization and transport of radionuclides, including plutonium, in both reducing and oxidizing environments (see, for example, Jakubick and Church 1986 p. 1; Curtis et al. 1989 p. 49; Brookins 1990 pp. 285-287; Cramer and Smellie 1994). Pena Blanca in

Chihuahua, Mexico, is a natural, high-grade uranium deposit located in unsaturated tuff that is believed to be a good analog to the Yucca Mountain repository. Further analog-oriented studies of this 8 million-year-old-deposit may provide important transport information (Murphy 1995 p. 44; Murphy et al. 1997 pp. 105-111). In addition, DOE participated in a three-year analog study at Pocos de Caldas, Brazil, which focused on radionuclide transport issues (Chapman et al. 1991 p. v).

6.5.1.8 Unsaturated Zone Transport

- *Effects of thermal-hydrologic-chemical alteration.* The current model does not account for alteration of the unsaturated zone because of thermal alteration of minerals, chemical interactions of repository materials, mineral dissolution, and precipitation. These effects are potentially important to transport behavior in the unsaturated zone. Thermal-chemical alteration could cause reduced matrix sorption and fracture/matrix interaction. The potential effects associated with alteration of the host rock and radionuclide transport pathways caused by alkaline plumes derived from the concrete masses and extending into the geosphere is not well constrained. The development of a coupled, reactive transport representation would aid the assessment of these potential effects. These effects could cause increased release rates from the unsaturated zone for base case transport results; however, the sensitivity studies (illustrated in Section 5.6) are expected to bound most of these potentially nonconservative interactions.

- *Colloid Filtration.* The ability of colloids to facilitate radionuclide transport is a function of their ability to migrate over large distances without being filtered by the host rock. This filtration effect was not included in the TSPA because of inadequate information to bound the mechanism. The filtration effect is particularly important for the fraction of radionuclides that are irreversibly bound to colloids. The assumption that there is no colloid filtration is conservative for transport in the unsaturated zone.

- *Fracture Sorption.* The effects of higher infiltration evaluated in this TSPA imply that transport will be fracture-dominated in many of the unsaturated zone units. Minerals that line fractures are known to sorb radionuclides, but more information is needed to define the distribution and character of the fracture materials so that sorption on fracture surfaces may be included. However, the assumption that radionuclides do not sorb onto fracture surfaces is conservative for transport in the unsaturated zone.

- *Matrix Diffusion.* Radionuclide transport in the unsaturated zone can be sensitive to changes in matrix diffusion depending on the nature of the release from the engineered barriers. The way in which the fracture/matrix contact area is used for calibrating the flow model suggests that some type of coupling strength may be appropriate for matrix diffusion. Sensitivity studies presented in Section 5.6 suggest that the influence of matrix diffusion on total system performance is small.

6.5.1.9 Saturated Zone Flow and Transport

- *Additional potentiometric and transport data.* Data are lacking in the saturated zone from approximately 10 km (6 miles) downgradient of the repository to the 20-km (12-mile) boundary used in the analyses. Available data in water levels, hydrochemistry and the characteristics of the alluvium are lacking in this area. Water-level measurements would improve understanding of the groundwater flow directions in this area and provide additional data for flow modeling calibration. There is uncertainty about where flow in the shallow saturated zone enters the alluvium along the flow path from the repository or even if flow occurs in the alluvium within 20 km (12 miles) of the repository. This uncertainty is particularly important given the potentially higher sorption coefficients of some radionuclides such as neptunium in the alluvium. Hydrochemical data could provide information to constrain flow paths in the model for the saturated zone in this area. Additionally, there are little site-specific data on the hydraulic, mineralogic, or geochemical characteristics of the alluvium in the saturated zone from this area.

- *Geochemical and isotopic data.* Additional geochemical and isotopic data from the saturated zone could provide important constraints on conceptual models of groundwater flow and on some key parameters for performance assessment. Enhanced vertical resolution of hydrochemical sampling could provide information on the degree of mixing in the flow system, with implications for the amount of transverse dispersion that would occur in contaminant transport from the repository. The data could also provide a better understanding of the flow paths in the saturated zone downgradient from the repository and of the magnitude of recharge to the system from Fortymile Wash. Reliable age dating of groundwater along the flow path from beneath the repository would constrain the travel times through the system and the appropriate range of values of effective porosity in the fractured volcanic tuff units. By inference, these data would provide information on the process of matrix diffusion in the fractured units. Additional electro-chemical-potential data to determine oxidation/reduction states in the saturated zone could improve sorption and solubility parameters for calculating radionuclide transport, as well as aid in the understanding of the connection between the shallow and deep aquifers at the site.

- *Three-dimensional flow and transport model.* A three-dimensional flow model for the saturated zone could provide the basis for radionuclide transport simulations that explicitly model relevant processes. An improved, site-scale flow model should be consistent with all the available data from the site. In addition, development of the flow model should be focused on simulation capabilities that are important for accurate transport modeling. These capabilities include incorporating variability and uncertainty in aquifer properties and numerical methods for simulating solute transport with minimal numerical dispersion.

- *Dispersivity and dilution.* The dilution factor for the saturated zone has been shown to be a sensitive parameter in TSPA-VA. The appropriate range for dilution and vertical transverse dispersivity is also uncertain. Reduction in the uncertainty in this parameter can benefit from inferences from analog sites or possibly from analyzing natural solute tracers in the saturated zone at the site. Additional effort could be devoted to evaluating potential analog systems of saturated flow in fractured media or in highly heterogeneous porous media.

- *Effects of climate change.* Climate change has an effect on the saturated zone that was simplified in the TSPA. For example, changes in flow paths were not considered. Changes in groundwater flux were based on regional-scale modeling but without an estimate of uncertainty. The effects of additional discharge locations and the presence of surface water were not considered. The effects of additional recharge at Yucca Mountain, Fortymile Wash, and the regions that are up gradient were also not considered. The saturated zone modeling should attempt to better reflect the wetter climates that might be normal for the Yucca Mountain region.

- *Colloid transport.* More realistic models of colloid-facilitated transport should be implemented in the saturated zone models. These models should be tied to site-scale observations of colloidal transport.

6.5.1.10 Dilution from Pumping

- *Geosphere/biosphere interface.* The geosphere/biosphere interface was defined as a well located at the point of highest concentration of contaminants in the groundwater. Natural discharge points were not considered. Dilution from mixing contaminated water with uncontaminated water during well pumping was not considered. An improved definition and modeling of this interface would lend further credibility to the calculations.

- *Pumping from Wells.* Pumping from a well could mix contaminated and uncontaminated waters—as might storing water from multiple sources in tanks—diluting any contamination. These effects have not been included in TSPA-VA (although they have been estimated in a sensitivity study presented above in Section 5.8.1). If it is determined that dilution of radionuclides by the natural environment during transport is less than the values used in TSPA-VA, it would be reasonable to examine the dilution produced during well withdrawal and storage.

6.5.1.11 Biosphere Transport and Uptake

- *Site-specific data.* The biosphere modeling indicated that the pathways that contribute the most to the calculated radiation dose rate are drinking-water ingestion, leafy-vegetable ingestion, and meat ingestion. The parameters that create the most uncertainty in these pathways are the related consumption rates, crop-interception fraction, crop-resuspension factor, grain irrigation rate, animal-uptake scale factor, and egg yield. Some of these pathways and parameters were defined based on the regional-survey data; however, additional site-specific data could improve the dose-rate calculations. The crop-interception fraction could be measured. The scaling factors for animal uptake and soil-to-plant transfer could be refined to appropriately reflect the bioavailability of radionuclides in the Amargosa Valley region and the vicinity of Yucca Mountain in general.

- *Definition of critical group and individual receptor.* For TSPA-VA, the receptor for radionuclides released from a repository at Yucca Mountain is an average member of the critical group (the reference person), to be consistent with guidance from the National Research Council (1995; p. 52). Several simplifying assumptions are made concerning the critical group: it comprises only adults; the members have habits similar to those people residing in the Amargosa Valley region today; all water for drinking and production of (locally produced) foodstuffs is the most contaminated water at 20-km (12-miles) downgradient from Yucca Mountain. It is recognized that some persons in the Amargosa Valley might be more susceptible to radionuclide contamination than the average person. Consideration of a child receptor in particular would improve the estimation of radiological effects on the critical group. It is also recognized that a better understanding of water usage and food-distribution patterns would improve the estimation of radiological effects.

- *Biosphere changes with climate.* Attempting to quantify the effect of the technological and societal advances or declines over many thousands of years is speculative. To address the influence of climate on the critical group, a comparatively small survey was conducted in Lincoln County, Nevada. This area was selected as an analog because the climatic conditions in Lincoln County are similar to those predicted for a future Amargosa Valley region. Lincoln County is approximately 200 km (120 miles) northeast of the repository site. This county is generally higher in elevation and is characterized by cooler temperatures and higher precipitation (mean annual rainfall is 2-3 times greater, depending upon elevation) than the Yucca Mountain area. However, the results from this survey were being analyzed and are not included in this report. The data gathered in the Lincoln County survey should be available to support a future TSPA.

- *Long-term build-up of radionuclides at natural discharge locations.* Modeling did not include assessments of surface soil contamination and subsequent buildup from natural discharge locations (e.g., from Franklin Lake Playa). Long-term buildup of radionuclides in natural discharge areas could alter dose calculation, especially if natural discharge affects sites of future habitation. Of interest is the amount that radionuclides might be concentrated at a natural discharge site; how these sites might be used by future inhabitants; and how much these radionuclides might subsequently be dispersed via wind or other erosional processes.

- *Well-location assumption.* All of the locally produced food consumed by the reference person was assumed to be grown with water having the same maximum levels of contamination. This assumption might be appropriate if the contaminant plume is large and well-dispersed; however, in the TSPA-VA modeling, the contamination plumes generated by a repository at Yucca Mountain would not cover the entire Amargosa Valley. This assumption might also be appropriate for a subsistence farmer; however, no one currently in the Amargosa Valley region fits this profile. Therefore, to be consistent with the current demographics and population locations, food sources could be more carefully investigated to estimate the amount of possible contamination.

- *Soil build-up.* Using contaminated water for crop irrigation can cause the buildup of radionuclide in the soils. These radionuclides can then produce an external dose and may cause additional internal doses if resuspended and contaminated soil particles are inhaled or ingested. The process of retention and buildup of radionuclides through continued irrigation with contaminated groundwater and the subsequent impacts on the various pathways defined for the biosphere could be further investigated.

6.5.2 Insights from the Total System Performance Assessment Peer Review Panel

An independent peer review of the TSPA-VA is being conducted. This peer review has the objective of providing a formal, independent evaluation and critique of the TSPA-VA in order that technical issues associated with the approach, methodology, and assumptions can be addressed prior to initiating the TSPA for the site recommendation and license application. The peer review has consisted of three interim reports to date (Whipple et al. 1997a, 1997b, and 1998) and will culminate in a final report due to be completed by the end of 1998. The Peer Review Panel consists of 6 individuals who have technical backgrounds that span the major disciplines of significance to postclosure performance, namely geohydrology, geoengineering, geochemistry, materials science, materials engineering, health physics, and risk assessment.

The Peer Review Panel has reviewed preliminary draft materials describing the approach, methodology, and assumptions to be used in the TSPA-VA as well as presentations made by the TSPA analysts to external review organizations such as NRC and the NWTRB. These materials have been supplemented by interactions of the panel members with TSPA-VA and other Yucca Mountain project staff. Although the review of the panel has been limited to date, the panel was impressed with the initial draft of Volume 3 of the Viability Assessment (Whipple et al. 1998, p. 45). However, the panel did not comment on the draft documentation of the TSPA-VA at the time of their third interim report. Ultimately the TSPA-VA Peer Review Panel will review this Volume 3 of the VA and the associated *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) which is being prepared concurrently. This review will be completed in fiscal year 1999, with the results being used to assist DOE in prioritizing the development of the approach and methods to be used for the TSPA for the site recommendation and LA.

Based on the Peer Review Panel's reviews of the preliminary materials available to them, they have provided a range of observations, conclusions, and recommendations that must be considered in the development of the approach and methodology for the next iteration of the TSPA for the site recommendation and LA. The conclusions of the panel have been grouped into the following categories:

- Physical events and processes considered
- Use of appropriate and relevant data
- Assumptions made
- Abstraction of process models
- Application of accepted analytical methods
- Treatment of uncertainties
- Other issues

In the following paragraphs, these issues are discussed with the goal of identifying the additional models and analyses that will ultimately be required in the development of the TSPA for the site recommendation and LA. These issues may be treated as critiques of some of the assumptions made in TSPA-VA and as such some discussion is included to note the potential significance of the criticism to the conclusions reached in the TSPA-VA.

6.5.2.1 Physical Events and Processes Considered

The panel notes in their review of preliminary draft materials that the TSPA-VA has not fully addressed the potential effects associated with a number of processes. The examples they cite include coupled phenomena such as the chemical and mechanical interactions in the thermohydrologic analyses, degradation of the drift with time and the effects this may have on waste package performance, and dispersion and dilution of radionuclides in the groundwater especially at early times when small source areas may be more likely. In addition, the panel believes that too much attention may have been devoted to the potential consequences associated with low-probability disruptive events such as volcanic events.

It is acknowledged that the TSPA-VA has not addressed all processes that must eventually be evaluated as DOE proceeds from the VA to the site recommendation and licensing. Nevertheless, bounding sensitivity analyses have been performed of the potential effects associated with coupled processes by varying the fracture characteristics of the rock mass around the drift opening. These changes in fracture properties would be the principal effect of the coupled phenomena identified by the panel. The analyses presented in Section 5.1 illustrate that, within the range of parameters examined, these effects are minimal on the dose rate.

Although additional analyses are also required of the effects of drift degradation to support the site recommendation, assuming the VA reference design of no long-term support or backfill placed in the drift, some sensitivity analyses of these effects have been examined in the VA. For example, the chemical degradation of the drift liner has been examined in Section 5.3. The mechanical effects associated with drift degradation are examined in the rock fall disruptive event scenario described in Section 4.4.3. The hydrologic effects associated with drift degradation were addressed in the same sensitivity analyses related to coupled processes described above. Finally, the potential thermal effects associated with the degradation of the drift have been examined in Chapter 3 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) and found to be inconsequential.

DOE agrees with the panel's observation that the degree of dispersion and dilution in the transport modeling should be related to the fraction of the area of the repository represented by degraded waste packages. This issue will be addressed in the analyses planned for the site recommendation and the LA. It is also important to note that the dispersion/dilution models used in the unsaturated and saturated zones as well as the assumptions made regarding the probability of the single well intersecting any dispersed plume in the alluvial aquifer must be internally consistent. In the case of TSPA-VA, it was conservatively assumed that the well intersected the highest concentration area of the plume and that there was minimal dilution in the saturated zone. This latter assumption was predicated on an expert elicitation of the saturated zone flow and transport model, in which the experts assumed that the source region was relatively large (on the order of several hundred meters) and therefore the additional dispersion effects expected in the saturated zone were minimal. If, as pointed out by the Peer Review Panel, the source region in the unsaturated zone is small, then the consistent saturated zone model should include the expected dispersive effects which are much larger (Whipple et al. 1998, p. 38). This alternative representation has been acknowledged in the assessment of the significance of unsaturated zone transport in Section 6.4. It is acknowledged that refined unsaturated and saturated zone transport models need to be investigated as part of the site recommendation and LA.

Some disruptive events are of very low probability. The probabilities of some disruptive events of interest to the long-term performance of the Yucca Mountain repository system are described in Section 4.4. Although DOE agrees that these low probability events pose very low risk to the public (as documented in Section 4.4), they are of interest to both the public and regulatory agencies such as NRC. Given this interest, DOE believes this effort has been given about the right level of attention in the TSPA-VA.

6.5.2.2 Use of Appropriate and Relevant Data

The panel notes in their review of preliminary draft materials that the TSPA-VA lacks site-specific data citing as examples the saturated zone from Forty Mile Wash to the Amargosa Valley, soil properties for determination of sorption characteristics, data on colloid transport, and radionuclide plant uptake factors. They also note that experimental data are needed to confirm the processes that control neptunium solubility and their belief that there are insufficient data to support the selection of the materials for use in the final waste package design. Finally, the panel believes that there is an over-reliance on the use of data generated by YMP scientists and only limited use of the published literature.

As DOE moves forward from the VA to the site recommendation and LA, all relevant site-specific data must be used to improve the basis for the process models. A wealth of site-specific, laboratory-derived and published literature information has been used in the course of developing the TSPA-VA. The unsaturated zone flow model has been based on extensive surface-based geologic and hydrogeologic testing, in situ observations of thermo-hydro-chemical conditions that are indicators of water movement, and laboratory studies of fundamental processes affecting water movement in fractured unsaturated media. The thermal hydrologic models have been based on in situ tests such as the single heater test and laboratory tests. These models have been compared to a wide range of similar models available in the published literature to add confidence in their relevance to the performance of the Yucca Mountain repository system. Similarly, a significant amount of laboratory testing has been conducted to evaluate the degradation characteristics and rates of the materials that are proposed to contain the wastes, in particular the candidate metals for the waste package. These tests have been complemented by information available in the published literature.

DOE will carefully consider the panel's input with regard to data availability. However, it is appropriate to acknowledge that uncertainty exists in many of the models that are used to evaluate the behavior of the Yucca Mountain repository system. This uncertainty is documented throughout this volume and in the companion *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i). Some of these uncertainties can and indeed will be reduced in the next years with an appropriate combination of site-specific field testing, laboratory testing and literature data. These additional data will be used to enhance the current understanding in those factors most significant to long term performance for use in the site recommendation and LA. The activities identified as being most important for continued testing are discussed in Volume 4, which was not available for the Peer Review Panel to review.

Despite the desire to reduce the uncertainty in many of the key components of the TSPA, it must nevertheless be acknowledged that some uncertainty will remain. As noted in the introduction to Chapter 6, this uncertainty is the result of the time and space scales of interest. The goal of the TSPA analyses is to reasonably bound this remaining uncertainty. DOE believes that the uncertainty has been reasonably bounded in the TSPA-VA. Even as efforts are made to reduce the uncertainty in the coming years, DOE will continue to assure that the remaining uncertainty, and its potential effects on postclosure performance, are appropriately addressed in the TSPA for the site recommendation and LA.

6.5.2.3 Assumptions Made

The panel notes in their review of preliminary draft materials that the TSPA-VA made some assumptions that need to be appropriately justified. They cite an example of the fully coupled thermal-hydraulic-mechanical-chemical interactions being of second order importance. In addition, they believe that incorporating other assumptions could provide useful insights into repository performance. They cite an example of prolonged ventilation of the repository.

DOE agrees that all assumptions need to be appropriately justified and their significance to long term performance evaluated. The purpose of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) is to provide the appropriate rationale with the supporting data, analyses and references to relevant project information, including the process model for each TSPA component. In the particular example cited by the panel, the significance of the complex process interactions mentioned has been evaluated with a range of individual sensitivity studies. For example, the seepage model has been modified to account for either an increase or decrease in fracture permeability and capillarity that might be the result of these coupled processes. These results indicate that the overall performance is not significantly sensitive to such changes, which is the basis of the assumption. DOE acknowledges that it will be very difficult to predict with any degree of robustness the exact thermal-hydraulic-mechanical-chemical interactions that might occur in each drift in the vicinity of each waste package. However, such fidelity in the analysis is not required. What is required is to reasonably bound the potential effects so that informed decisions can be made about the potential safety of the repository system.

DOE agrees that a range of alternative assumptions needs to be evaluated to gain sufficient understanding of how the system is likely to perform. Many of these alternative assumptions have been explored in Chapters 4 and 5 of this volume. Most of these have relied on the VA reference design, which does not include long term ventilation. Many others, including a range of alternative designs such as the long-term ventilation design mentioned by the panel, are worthy of investigation. These alternative designs and design features will be a major emphasis of the Project in fiscal year 1999 as DOE moves forward to selecting the appropriate site recommendation and LA design.

6.5.2.4 Abstraction of Process Models

The panel notes in their review of preliminary draft materials that the abstraction process used in the TSPA-VA needs to be reviewed. In particular they note that an abstracted model should only be used if the process-level model upon which the abstraction is based confirms its use is justified. They also note that there needs to be appropriate integration of the abstracted models among the different groups developing the TSPA. They cite an example related to the consistency between unsaturated zone and saturated zone transport model assumptions.

The abstracted models used in TSPA-VA have been based either on an underlying process model or the basic data that describe the relevant process. All models, whether process-based or abstracted, are abstractions of reality. All models, whether process-based or abstractions, need to be compared to appropriate in-situ or laboratory observations. Unfortunately, there are minimal opportunities to compare the results of the models over the spatial and temporal scales of interest to direct observations, with the exception of some natural analogs.

The *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) describes the details of the abstraction for each of the components of the TSPA-VA. In the case of the abstracted models of unsaturated zone flow, seepage into drifts, thermal hydrology, waste package degradation, cladding degradation, unsaturated zone transport, and biosphere transport the abstracted models are based directly on the underlying process model. In some cases the model was directly used in the TSPA analyses while in other cases a response surface capturing the relevant results of the process model was used. In the case of abstracted models of climate change, waste form degradation, radionuclide solubility, colloid stability and mobility, and saturated zone transport the abstracted models are based directly on available externally published or project-generated global, regional, site and laboratory data. Therefore, DOE believes there is a good, even if preliminary, technical basis for all the models used in the TSPA-VA.

Nevertheless, DOE understands the Peer Review caution that all models and assumptions need to be appropriately justified when used in the development of a license application. Therefore, a range of activities that are summarized in Volume 4 will be undertaken over the next few years with their aim being to enhance the confidence in all models used to make projections of repository performance.

DOE agrees that the assumptions made between different abstracted models need to be self consistent. Every effort was made in the TSPA-VA to assure this. For example, if different assumptions were made about infiltration rates, these were consistently applied through the system to evaluate changes in percolation flux, seepage fraction and flux, thermal hydrologic response, unsaturated zone groundwater velocity, and the water table elevation. An additional example is the propagation of transient changes through the system. If the climate was assumed to change at a particular time, then this change was propagated through changes in the unsaturated zone flow and transport, seepage, water table elevation and saturated zone flow and transport. DOE will continue to strive to assure that assumptions made between different components of the TSPA remain consistent.

An additional consistency issue noted by the panel centers around the depiction of contaminant plume dispersion in both the unsaturated and saturated zones. DOE agrees these need to be consistent. As noted above in the discussion in Section 6.5.2.1, the TSPA-VA assumed that the pumping well directly intersected the maximum concentration of the plume, that there was no mixing in the pumping well and that there was minimal dispersion in the saturated zone. This was the result of basing the dispersion effects on an expert elicitation which assumed the plume size entering the saturated zone was on the order of several hundred meters in width, that is the plume had already been dispersed in the unsaturated zone. As the panel points out, this assumption, may be appropriate at late times (several hundred thousand years) when waste packages located over a significant fraction of the area of the repository may be expected to be contributing to releases. However, this assumption is probably inappropriate at early times (less than 10,000 years) when only a very few breached waste packages are contributing to the dose rate. To be consistent, if a model of transport from discrete waste packages is invoked in the unsaturated zone, then the likely dispersion in the saturated zone and well intersection probability should be included in the analyses. These will be investigated in the TSPA for the site recommendation and LA.

6.5.2.5 Application of Accepted Analytical Methods

The panel notes in their review of preliminary draft materials that the TSPA-VA may contain models that contain uncertainties. In some cases they note that these models may have been applied without recognition of their potential limitations. They comment that these limitations could call into question the usefulness of the sensitivity and uncertainty analyses that have been conducted. They cite several models that exemplify their concern:

1. The saturated zone flow model

2. The thermal hydrology model including the coupled effects of thermochemical and thermo-mechanical interactions

3. The unsaturated zone flow model including the seepage model

4. The unsaturated zone transport model including the mechanisms of colloid migration

5. The waste package degradation model including the effects of crevice corrosion, chemical conditions on the waste package surface, stress corrosion cracking, and the generation of corrosion products in crevices between the outer and inner waste package materials

6. The cladding degradation model including its uncertainty

7. The saturated zone flow and transport model including greater resolution of the numerical representation

In addition to these individual models, they note that the relevant hypotheses, models and abstractions need to be verified by comparison to appropriate site-specific, laboratory or literature data.

DOE agrees that uncertainties exist in all the cited models. These uncertainties have been the focus of a number of workshops that DOE has held on the key components of the TSPA (see Table 2-2 for a complete listing of these workshops and expert elicitations). These uncertainties have also been documented in the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i).

One of the goals of the TSPA-VA is to examine the significance of these uncertainties on postclosure performance to assist in identifying the key factors requiring additional information prior to the submittal of the site recommendation and LA. As noted in Section 4.3 and Chapter 5, the significance of these uncertainties was investigated using both a range of parameters and models to describe the base case performance, as well as a suite of alternative models that were tested to determine whether differing assumptions would significantly change the results.

In all of the examples cited by the Peer Review Panel, at least one and in many cases several sensitivity analyses have been conducted to investigate the potential significance of the noted uncertainty. For example, a range of different seepage models has been studied to accommodate the most significant aspect of coupled processes (Section 5.1). A range of different assumptions regarding the waste package degradation model (including the fraction of the waste package with seepage, the size of the patches, and the distribution of the local chemistry on the waste package surface) has also been investigated (Section 5.4). Analyses have been performed using a range of distributions for the Zircaloy cladding degradation (Section 5.5). These included one case in which it was assumed there was no performance credit for the cladding.

DOE believes that the above sensitivity analyses and others described in Chapter 4 and 5 capture the range of likely performance of the repository system. Although it is possible to combine low probability models and parameters to further investigate the full range of performance, the results of these scenarios would have to be appropriately weighted by their low probability. Such analyses may be required in the licensing of the Yucca Mountain repository, but they were not the focus of the Viability Assessment, which was designed to investigate the probable behavior of the repository system.

DOE agrees that uncertainties other than those quantified in the TSPA-VA do exist. The panel correctly points out some of these. These uncertainties, as well as their potential significance need to be investigated prior to the site recommendation and LA.

In several instances, the Peer Review Panel notes the need to reduce the uncertainty that has been identified (whether that identification was by DOE, by the Peer Review Panel, by the NWTRB or by NRC). Although in many areas, there will be significant work performed over the next several years to minimize the uncertainty in models and abstractions for use in the TSPA for the site recommendation and LA, DOE acknowledges that even after this effort, uncertainties will still remain. The goal of the next TSPA analyses will be to address the significance of the remaining uncertainties and assure that the site and associated engineered barriers meets with a sufficient safety margin all applicable environmental and safety standards.

6.5.2.6 Treatment of Uncertainties

The panel notes in their review of preliminary draft materials that the TSPA-VA has "lumped" many types of uncertainties and that the results may be inappropriately insensitive to some aspects of the actual behavior of the repository system. In addition, the panel observes that the TSPA-VA has not provided adequate consideration of the multitude of factors that are necessary for estimating dose rates to the public.

DOE agrees that many types of uncertainty have been evaluated as part of the TSPA-VA. Assuming that all the relevant uncertainties were adequately characterized, quantified and incorporated in the models used for projecting system behavior, the system model should be able to determine the relative sensitivity of the system behavior to each of the model components. However, it must be acknowledged that not all uncertainties of all types have been included in the single system model used in TSPA-VA. To better investigate the complex interrelationships between the different system components, a number of alternative model sensitivity analyses were developed and documented in Chapter 5. These analyses were conducted in part to examine whether the initial results were indeed "inappropriately insensitive". DOE believes that the combined suite of sensitivity analyses have captured the range in probable behavior of the Yucca Mountain repository system.

DOE is aware of the many factors that contribute to the estimate of dose from a given radionuclide concentration in the groundwater. All of these factors are included in the model used to estimate biosphere dose conversion factors. Reasonable ranges for the key factors have been used in the development of the distribution of biosphere dose conversion factors. These distributions will be examined, including the sources and magnitudes of their uncertainty, prior to the development of the TSPA for the site recommendation and LA.

6.5.2.7 Other Issues

Based on the preliminary information available to the panel, they questioned the apparent focus of the TSPA-VA almost exclusively on the time period from 10,000 to 1 million years. They also note the need for additional supporting research in a number of areas, including water compositions in contact with the waste package, critical crevice corrosion temperatures, neptunium solubility and technetium sorption on degraded waste package materials.

DOE does not believe it focussed on any particular time frame in the analyses presented in Chapters 4 and 5. Generally, the results are displayed and discussed in three different time periods, the time from closure to 10,000 years, the time from 10,000 to 100,000 years and finally the time period from 100,000 to 1 million years. This has been done for several different reasons. First, the significance of the overall system response is difficult to examine in a single time period. Second, different insights are gained by examining the results over different time periods. Third, DOE does not know over what time period the environmental and regulatory performance objectives will be applied although preliminary indications are that the 10,000 year period will likely be addressed with quantitative performance requirements in the new regulations being prepared for Yucca Mountain. Finally, it is believed that conducting analyses to the time of the peak dose, whenever that occurs, allows some confidence in a more comprehensive, even if more qualitative, evaluation of the performance of the entire system.

As noted by the panel, the time period of interest is an important aspect of the defense in depth strategy that will evolve during development of the license application, and will undoubtedly be the subject of extensive discussion between NRC and DOE during licensing proceedings. The performance allocation approach presented in Volume 4 is consistent with the general approach taken to date, which assumes that both 10,000 year performance and peak dose are important performance measures.

DOE acknowledges the need for additional investigations in a number of areas, including those mentioned in the panel's report. The investigations that are planned over the next few years are those that would significantly improve upon the current understanding and which are significant to the system performance. The planned activities are described in Volume 4.

In summary, interactions with the Peer Review Panel continue to provide useful feedback as DOE moves forward to develop the basis for the next iteration of TSPA. Their Third Interim Report points to a number of areas where attention will need to be placed as the basis for the site recommendation and license application is prepared.

6.5.3 Comments from the Nuclear Regulatory Commission

NRC has recently provided DOE with initial comments on the TSPA-VA (Letter from Michael J. Bell, NRC to Stephan J. Brocoum, DOE, dated July 6, 1998). These comments were based on a series of three Technical Exchanges held between DOE and NRC staffs and their contractors, the most recent being in March, 1998. As noted in the letter, NRC comments are intended to facilitate DOE's effort to identify the future work that may be needed to develop a complete and acceptable license application.

DOE appreciates the opportunity to interact with NRC staff on the significant issues associated with assessing the long-term performance of the Yucca Mountain repository system. The comments provided by NRC during the Technical Exchanges and the above letter, in conjunction with the recently published Issue Resolution Status Report for Total System Performance Assessment and Integration (NRC 1998a) assist DOE in identifying the most relevant issues to NRC which can then be a basis for continued discussion and ultimately closure.

In the following paragraphs, each of the NRC comments is presented and briefly discussed. Where appropriate, the reader is referred to Volume 4, which contains descriptions of the activities DOE intends to conduct between now and the completion of the Site Recommendation Report and the submittal of the LA.

6.5.3.1 Radionuclides Tracked in the Performance Assessment

DOE concurs that the approach used for screening out unimportant radionuclides from the TSPA analyses needs to be well documented and the potential impacts of not considering the full suite of possible radionuclides present needs to be demonstrated. The radionuclide screening process is summarized in Section 3.5 of the TSPA-VA. The details of the implementation of this process and the small differences arising from not considering a full suite of radionuclides are discussed in Chapter 6 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i). Given that the suite of potentially important radionuclides may vary with the specific design, modeling assumptions, and scenarios analyzed, it is agreed that each TSPA analysis should address this issue and justify the inclusion or exclusion of certain radionuclides. DOE will continue to carefully review the assumptions of the radionuclides included in the TSPA analyses for the site recommendation and LA.

6.5.3.2 Consideration of all Significant Features and Processes in the Performance Assessment

DOE agrees that the rationale for including or excluding any potentially significant feature, event or process needs to be technically justified and clearly articulated. The rationale used in the TSPA-VA has been presented in the detailed descriptions of each component model documented in the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O1998i). Acknowledging that there is uncertainty in exactly what features, events and processes are most significant to long-term performance, a range of total system sensitivity analyses have been conducted in the TSPA-VA (Chapter 5). These analyses, together with subsystem and component model sensitivity analyses that are presented in the *Total System Performance Assessment-Viability*

Assessment (TSPA-VA) Analyses Technical Basis Document (CRWMS M&O 1998i) form the basis for the determination of importance. As DOE proceeds from the VA to the site recommendation and LA, these analyses will be expanded to include other potentially important features, events and processes, along with appropriate auxiliary analyses to support the resolution of relevant sub-issues in some of the NRC KTIs. All of these analyses will be documented in the LA.

6.5.3.3 Model Abstraction

DOE agrees that modeling assumptions should be consistent across different process models, unless there is a defensible technical rationale. An example of a case where different assumptions may be appropriate is the spatial and temporal averaging of hydraulic properties. Flow properties applicable at the scale of tens of meters (e.g., analyses in the vicinity of the drift) would be different than hydraulic properties hundred of meters (e.g., analysis of flow from the repository to the water table). Although the two parameter sets should be consistent, they may be different.

Every effort is being made through the document review process to ensure consistency of key parameters across both process models and abstracted models used in the TSPA-VA. For example, the climate change is propagated through the system by affecting the infiltration rate, percolation rate, seepage rate, groundwater velocity, water table rise and saturated zone flux. DOE will continue to rely on its internal and external review processes to ensure that the individual models and their coupling are appropriately consistent. In addition, DOE will use pre-licensing interactions and technical exchanges with NRC to check its effectiveness, as it proceeds with preparing the TSPA for the LA.

6.5.3.4 Documentation of Assumptions

DOE agrees that detailed documentation of the assumptions used in performance assessments is of fundamental importance to the transparency (and credibility) of the TSPA. This was the objective of producing a companion document, the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i), which augments the TSPA-VA documentation contained in Volume 3 of the VA. This documentation has undergone several internal reviews, with part of the review criteria being based on the NRC criteria noted in their Issue Resolution Status Report on Total System Performance Assessment (NRC 1998a). The documentation of the TSPA-VA model assumptions will ultimately be reviewed by the TSPA-VA Peer Review Panel, NRC and their contractors, the NWTRB, the State of Nevada, and by other interested parties. All comments received will help DOE ensure that the documentation for the TSPA for the site recommendation and LA is sufficient to facilitate regulatory review.

6.5.3.5 Transparency and Traceability of Analysis

DOE concurs with NRC comment that the performance assessment results should allow the importance of alternative models to be evaluated. This was the goal of the sensitivity analyses presented in Section 5. Based on previous NRC comments on this subject, as those from the NWTRB on this same subject, DOE is making a concerted effort to improve the graphical presentation and documentation of these diverse and multifaceted analyses. DOE expects to discuss this issue at future Technical Exchanges with NRC on their TSPA Methodology Issue Resolution Status Report, and through DOE's internal review process for technical reports.

6.5.3.6 Container Life

DOE agrees that the technical basis for the degradation characteristics and rates of the candidate waste package materials needs to be adequately justified. The basis for the model used in the TSPA-VA is described briefly in Section 3.4 of Volume 3 of the VA and in Chapter 5 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) along with the supporting documentation that is cited in these documents. DOE agrees that the basis must be enhanced further in the next few years leading to the LA design and performance assessment. The work to provide this additional confidence is described in Volume 4.

6.5.3.7 Role of Rockfall in Assessing Waste Package Lifetime

DOE agrees that rockfall effects need to be considered in the design and performance of the Yucca Mountain repository system. In the TSPA-VA this has been addressed with some explicit analyses of seismically-induced rockfall damage in Section 4.4.3 and an implicit incorporation of early waste package failures caused by unanticipated processes. These early juvenile failures were incorporated in the base case analyses after the NRC-DOE Technical Exchange in March, 1998. Continued definition of the potential consequences of early waste package failures will occur as part of the multiple barrier analyses expected to be performed for the TSPA for the site recommendation and LA. In addition, DOE plans to pursue dialog on this topic through its interactions with NRC in future Technical Exchanges.

6.5.3.8 Effectiveness of Engineered Barriers in the Event of Volcanic Activity

In the TSPA-VA, DOE has analyzed the potential consequences associated with low probability volcanic activity at Yucca Mountain. In these analyses, reasonably conservative assumptions were made regarding the potential waste package failure modes (presuming that such an event occurred.) Although alternative failure modes could be postulated, DOE believes that the analyses presented in the VA are sufficient to bound the potential risk (equal to the probability of the event times the consequences of the event) of the direct volcanic intrusion scenario. Alternative failure modes may be addressed in the TSPA for the site recommendation and LA and DOE plans to pursue dialog on this and other disruptive events and processes with NRC staff and their contractors as part of the ongoing DOE/NRC Technical Exchanges on TSPA and future interactions on the NRC TSPA Methodology Issue Resolution Status Report (i.e., as part of resolution activities on the scenario analysis subissue).

6.5.3.9 Neptunium Solubilities

DOE concurs that the basis for the selection of the neptunium solubility used in the TSPA analyses needs to be supported by applicable measurements under suitable conditions. The rationale for the statistical distribution used in TSPA-VA has been based on available laboratory measurements combined with information on the stability of various neptunium phases in different geochemical environments. This rationale is presented in Section 3.5. Because of the importance of neptunium to long-term dose assessments (see Section 5.5), DOE will continue to refine the estimates of the range of neptunium solubility values for use in the TSPA for the site recommendation and LA.

6.5.3.10 Matrix Diffusion

DOE agrees that the technical basis for differing assumptions about matrix diffusion need to be documented. These have been included in Section 3.6 and in Chapter 7 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i). In addition, a range of sensitivity analyses have been conducted with differing assumptions about matrix diffusion. These results are presented in Sections 5.6 and 5.7. DOE will continue to review this issue with NRC in future Technical Exchanges and Appendix 7 meetings to ensure that a defensible and reasonably conservative representation of this transport process is included in subsequent TSPAs for the repository.

6.5.3.11 Saturated Zone Transport

NRC notes that there are limited data to define the saturated zone transport along the groundwater pathway south of the repository to the postulated receptor location at 20 km (12 miles). DOE also recognizes the current dearth of hydrogeologic data in this region of the flow system. DOE also agrees that both the incorporation of additional field data, such as the data to be collected from the Nye County well network, and models which incorporate the uncertainty in the flow and transport characteristics are essential to the preparation of a complete and defensible TSPA for the proposed repository system.

6.5.3.12 Radionuclide Retardation

DOE agrees that the technical basis for sorption coefficients used in the TSPA analyses needs to be improved and more substantive. For the TSPA-VA, this basis is presented in Chapter 7 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i). The basis for application of any laboratory-derived sorption coefficients, and the analogs that are used to define correlations, needs to reasonably represent the uncertainty in this parameter. DOE will continue its effort to enhance the documentation of the basis for all assumptions as it moves forward from the VA to the site recommendation and LA.

6.5.3.13 Treatment of Colloids

DOE agrees that the mechanistic basis for the partitioning coefficient for plutonium sorption onto colloids, as well as the stability and filtration characteristics of these colloids, needs to be better defined between the VA and the site recommendation and LA. Efforts to better characterize the colloid distribution and reversibility are planned. These efforts are summarized in Volume 4.

6.5.3.14 Basis for Assigning Probabilities to Corrosion Potential Values

NRC notes that specific probabilities of different corrosion potentials, that have been derived from expert elicitation, have been used in the waste package degradation model of the TSPA-VA. The basis for these distributions is presented in Chapter 5 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i) as well as the expert elicitation report on waste package degradation (CRWMS M&O 1998b). As described in Volume 4, continued testing is underway to verify the corrosion potential distributions that should be used in future TSPAs.

6.5.3.15 Uncertainty in the Results of Expert Elicitation

NRC comments that the significance of the uncertainty representations in the expert elicitation results should be propagated through the total-system calculation. They cite an example of the point-value probabilities used to define the corrosion potential of Alloy 22. DOE concurs with this assessment. In fact, the sensitivity to this parameter was evaluated and documented in Chapter 5 of the *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (CRWMS M&O 1998i). As DOE continues toward the development of the site recommendation and LA, the uncertainty in all significant factors affecting postclosure performance will be investigated.

6.5.3.16 Development of Expert Elicitation Results for Use in Performance Assessment

NRC comments that in some cases, the cited example being the Probabilistic Seismic Hazard Analysis, the panel of experts did not have a clear understanding of how their results would be used in the performance assessment. In all of the elicitations conducted specifically for TSPA-VA, however, the CRWMS M&O presented the goals and objectives of the elicitation as well as exactly how the results were to be used. The goal of the cited seismic hazard analysis was focussed on inputs to the preclosure design and safety aspects of the surface and subsurface facilities as opposed to the postclosure performance assessments. DOE agrees that experts used in formal elicitations should be informed at the outset how the results of the elicited judgments will be used.

In summary, DOE finds the comments that NRC (and their contractors at the Center for Nuclear Waste Regulatory Analyses at the Southwest Research Institute) have made during and after the Technical Exchanges held over the last year to be very constructive and helpful. DOE anticipates that NRC will comment on the Viability Assessment when it is formally released. DOE believes that these comments in conjunction with meetings on each of the individual Issue

Resolution Status Reports will assist DOE in working towards a defensible LA.

6.5.4 Concluding Remarks

Current data and analyses indicate that the Yucca Mountain site provides favorable features for limiting the contact of water with the waste. The location of the site in a semiarid region and the nature of the site itself limit the amount of water that can reach the repository. The site provides a thick unsaturated zone where the waste can be placed deep below the surface and well above the water table; the waste packages would therefore be protected from changes in conditions at the surface while still being kept well away from the saturated zone. The site would therefore provide predictable and stable environments for design of engineered barriers that could further limit the exposure of waste to water.

Performance assessment and design studies indicate that there are a number of options for the design to keep water away from the waste. They indicate, for example, that the highly corrosion-resistant inner container and the thick steel outer container of the reference design of this VA each provide effective barriers against water. Although as noted above there are some issues that must still be addressed, the estimates indicate that robust waste packages could be designed to remain intact for thousands of years in the repository environments. The studies also indicate that the spent nuclear fuel cladding would likely provide an additional barrier to water contacting the waste, once the outer and inner waste package barriers are breached.

In addition to expected repository performance, the postclosure safety case also explicitly considers processes and events that could disrupt a repository at the Yucca Mountain site. These include disruptive natural processes (seismicity and volcanism), potential human intrusion associated with exploration for natural resources, and nuclear criticality. Each of these potentially disruptive processes and events has been examined in this volume. The analyses presented in Section 4.4 indicate these disruptive scenarios introduce no substantial increase in the risk to long-term public health or safety.

As noted in the Introduction (Volume 1), a number of investigations are required to assist DOE in developing the safety case for the Yucca Mountain repository system. The safety case is comprised not only of evaluations of total system performance similar to those documented in this volume, but also of evaluations of multiple barriers, treatment of the safety margin, and use of natural analogs as qualitative indicators of expected behavior. In addition, the total safety case must consider the operational and preclosure safety of the workers and the public, rather than solely the long-term postclosure aspects evaluated in TSPA.

The goal of each successive total system performance assessment iteration is to refine the analyses based on improved site understanding and the maturation of the design concepts for the engineered system. Using the results of the performance assessments documented in this volume and summarized in the previous portions of this section, the key attributes and principal factors likely to affect the LA performance assessment have been identified and prioritized.

The effects of uncertainty included in the analyses of the principal factors affecting postclosure performance projections for the reference design were identified and ranked in Section 6.4. These rankings provide a partial basis for prioritizing investigations to reduce the uncertainty in these factors. However, there is also additional information in the form of judgment that must also be incorporated in the rankings. Finally, although uncertainty in a specific factor may have a large effect on the final result of a TSPA analysis, reduction of that uncertainty may not be possible. This may be true because of prohibitive cost of a study, lack of technology to address a specific question, or lack of adequate time to complete a study before the LA submission deadline.

Therefore, prioritization of the principle factors, and the work needed to characterize each of them in the TSPA, is conditional based on the reference design and the TSPA modeling and assumptions. Reduction in uncertainty may not be practical, or even possible, for some of the factors that are most important to performance. Some of the factors that are most important to performance may already be sufficiently well characterized. The importance of some factors may be obviated by a change in the repository design. Only where data are deemed inadequate and can be improved from a practical perspective, and should be improved from a licensing perspective, will additional work be performed.

The TSPA-VA indicates that for the first 10,000 years the expected dose rate derived from the repository at Yucca Mountain is essentially zero. Even within the first 100,000 years, the anticipated dose rate from a repository at Yucca Mountain is less than the national average for radiation from non-medical sources (approximately 300 mrem/year). Only at time scales of greater than 100,000 years does the additional dose rate from a repository approach the average background dose rate. The TSPA-VA also finds that the engineered system in the reference design has considerable impact on repository performance for a long period, on the order of several hundred thousand years. For even longer periods of time, however, the natural system dominates performance. The TSPA-VA shows that some factors and how they are modeled can have especially important influences on performance: for example, the corrosion characteristics of Alloy 22, the dilution of radionuclides in the transport pathways, and the transport of radionuclides as colloids. The TSPA-VA reinforces some ideas about performance; for example, that it is important to isolate waste from advective flow and that, for a dose-based standard, the rate of releases of radionuclides from a repository and the dilution in the environment are important. And finally, the TSPA-VA points out areas where improvements could be made to the TSPA models and data, including the most important uncertainties that could be reduced and the most important assumptions that could be addressed in the future.

The prioritization of the information needed to address the principal factors affecting expected postclosure performance allows focusing the testing and analysis programs on the key remaining questions related to repository performance. This prioritization and the rationale behind the allocation for each principle factor is given in Volume 4. The schedule for addressing these issues is also found in Volume 4.

The analyses of the TSPA-VA will provide a large part of the basis for the work required for construction of the TSPA for the site recommendation and the LA. However, as discussed above, the specific analysis results do not tell the entire story. In some cases, there was little applicable data available at the time for developing a model of a given factor. In other areas, because site characterization and design activities are still ongoing, new information has indicated that the ranges of values used may have been incomplete or too large. In each such case, it was necessary for the analysts to use judgment, based on their experience, to interpret the validity of their results. Where judgment has been used to modify the relative influence of uncertainty of a factor to be different than that shown by analysis (as documented in Volume 4), there is clearly a need to improve the confidence in the model for that factor. In addition, independent groups, such as the TSPA Peer Review Panel and the Nuclear Regulatory Commission, have provided suggestions of areas where more work is necessary to develop adequate TSPA models.

The next step for the Yucca Mountain TSPA team is to develop a comprehensive plan for addressing the important issues. While the general outline of this plan is contained in Volume 4, many details of the specific modeling activities must still be developed. A series of abstraction and testing activities, involving investigators from site characterization, design, and the performance assessment organizations will be convened to determine how to best incorporate the available information into appropriate representations of the total system. Because it will not be possible to develop these models to the same degree of complexity, the relative sensitivity of the factor will determine how much effort will be expended to improve any specific model or parameter. The outcome of these activities is anticipated to result in an improved set of total-system performance assessment models. These models will provide DOE with the reasonable assurance required to understand how the system will behave and the degree of safety the system will provide to the public.

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7.2 STANDARDS AND REGULATIONS

Unless otherwise dated, the *Codes of Federal Regulations* cited in this document were revised as of January 1, 1998.

10 CFR (*Code of Federal Regulations*) 60. Energy: Disposal of High-Level Radioactive Wastes in Geologic Repositories; Technical Criteria; Final Rule. 238445.

40 CFR 191. Protection of Environment: Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes; Final Rule. 1985. 221863.

48 FR 28194 (1983). 10 CFR Part 60, Disposal of High-Level Radioactive Wastes in Geologic Repositories Technical Criteria; Final Rule. 238518.

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APPENDIX A

GLOSSARY

APPENDIX A

GLOSSARY

The glossary is divided into two sections. Section A.1 is a general glossary of terms used in the TSPA-VA; Section A.2 contains a listing of statistical terms that are used in or are relevant to other statistical terms used in the TSPA-VA. Definitions are written with emphasis on the relationship of the term to the TSPA-VA process and are taken from previous performance-assessment documentation, where possible, or from standard reference materials.

Many of the definitions in this Glossary are Yucca Mountain Site Characterization Project specific.

A.1 GENERAL GLOSSARY

This section is a general listing of terms used in the TSPA-VA. Statistical terms are in Section A.2.

Abiotic	Characterized by the absence of living organisms.
Absorbed Dose	The energy absorbed from ionizing radiation per unit mass of irradiated material. Units of absorbed dose are the rad and the gray (Gy).
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	Distillation of the essential components of a process model into a suitable form for use in a total system performance assessment. The distillation must retain the basic intrinsic form of the process model but does not usually require its original complexity. Model abstraction is usually necessary to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and uncertainty analyses.
Actinides	A series of chemically similar, mostly synthetic, radioactive elements with atomic numbers from 89 (actinium) through 103 (lawrencium).
Activity	Cumulative curie count. See Radioactivity.
Adsorb	To collect a gas, liquid, or dissolved substance on a surface as a condensed layer.
Adsorbate	A substance that is adsorbed. See Adsorb.
Adsorbent	A substance upon which another substance is adsorbed. See Adsorb.
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.

Adsorption Isotherm	Relationship of the quantity of an adsorbed component to its quantity in the fluid phase (expressed in concentration) at constant temperature (i.e., under isothermal conditions).
Adsorption Coefficient	See Sorption Coefficient.
Advection	The process in which solutes are transported by the motion of flowing groundwater. Advection in combination with dispersion (hydrodynamic dispersion) controls flux into and out of the elemental volumes of the flow domain in groundwater transport models. The term convection is sometimes used for advection but is not used interchangeably in the TSPA-VA.
Advisory Committee On Nuclear Waste	A committee established under the Nuclear Regulatory Commission to provide independent reviews of, and advice on, nuclear waste facilities, including application to such facilities of 10 CFR Parts 60 and 61 (disposal of high-level radioactive wastes in geologic repositories and land disposal of radioactive waste) and other applicable regulations and legislative mandates.
Aerobic	Living or active only in the presence of oxygen, as used in reference to bacteria that require oxygen; a condition in which oxygen is present.
Air Mass Fraction	Mass of air divided by the total mass of gas (typically air plus water vapor) in the gas phase. This expression gives a measure of the "dryness" of the gas phase, which is important in waste package corrosion models.
Algorithm	(1) The set of well-defined rules that governs the solution of a problem in a finite number of steps. (2) A mathematical formulation of a model of a physical process.
Alkaline	See pH.
Alloy 22	See Inner Barrier.
Alluvium	Sedimentary material (clay, mud, sand, silt, gravel) deposited by flowing water or by wind.
Alternative	Plausible interpretations or designs based on assumptions other than those used in the base case that could also fit or be applicable based on the available scientific information. When propagated through a quantitative tool such as performance assessment, alternative interpretations can illustrate the significance of the uncertainty in the base case interpretation chosen to represent the repository's probable behavior.
Ambient	(1) Undisturbed, natural conditions such as ambient temperature caused by climate or natural subsurface thermal gradients. (2) Surrounding conditions.

Anaerobic	(1) Living or active only in the absence of oxygen; used in reference to bacteria that do not require oxygen. (2) A condition in which oxygen is absent.
Anionic	An atom or group of atoms having a negative charge.
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.
Annual Dose	For human exposure scenarios, a measure of an individual's exposure to radiation in a year.
Annual Committed Effective Dose Equivalent	Composed of terms in 40 CFR 191, Subpart B, in which an annual committed effective dose means the committed effective dose caused by 1-year intake from released radionuclides plus the annual effective dose caused by direct radiation from facilities or activities. See Effective Dose Equivalent and Committed Dose Equivalent.
Annual Frequency	Number of occurrences on an annual basis.
Anthropogenic	Alterations of the environment resulting from the presence or activities of humans.
Aqueous	Pertaining to water, such as aqueous phase, aqueous species, or aqueous transport.
Aquifer	A subsurface, saturated rock unit (formation, group of formations, or part of a formation) of sufficient permeability to transmit groundwater and yield usable quantities of water to wells and 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 proposed repository. This number is 85 metric tons of uranium (MTU) per acre and remains a constant value over time for calculations in which the amount of waste per acre in the repository is assumed to remain constant.
AREST-CT Computer Program	A general modeling code that considers both equilibrium and kinetically controlled chemical reactions between solid phases, aqueous solutions, and gas under flowing conditions.
Average Individual	An individual representative of the life style in the Amargosa Valley with regard to eating, drinking, and other activities that may be relevant in a human exposure scenario as determined by a survey of Amargosa Valley residents by TSPA-VA researchers.

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, and the human body.
Basalt	A dark, fine-grained igneous rock originating from a lava flow or minor intrusion, composed mainly of plagioclase clinopyroxene and sometimes olivine, and often displaying a columnar structure.
Base Case	The sequence of anticipated conditions expected to occur in and around the proposed 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 Model	A computer model that represents an assessment of the most likely range of behavior for the overall repository system and is a combination of the most likely ranges of behavior for the various component models, processes, and associated parameters.
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 effective dose equivalent) per unit concentration. Depends on the radionuclide(s), pathway(s), climate, and other factors. A key assumption is that the dose is a linear function of concentration at the geosphere/biosphere interface.
Boiling Regime	One of two divisions (the other being the cooling regime) used to delineate the reactions between the gas, water, and minerals in the rock that occur as the system heats and boiling of the pore water occurs through time.
Borehole	A hole drilled from the surface for purposes of collecting information about an area's geology or hydrology. Sometimes referred to as a drillhole or well bore.
Borosilicate Glass	High-level radioactive waste matrix material 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. An example is using a specified groundwater flux from infiltration as a boundary condition for a flow model.
Breach	An opening in the waste package caused by gradual degradation of the outer and inner barriers that allows the waste to be exposed, and possibly released, to the external environment.
Breakthrough	The time at which the concentration of a substance, usually in groundwater, arrives at a particular point of interest after having been tracked as it moves through space.
Breakthrough Curve	A means of describing transport of radionuclides along a geosphere pathway by constructing a curve that is a cumulative probability distribution. 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 that has traveled the pathway at any specified time.
Buoyant Convection	Fluid movement, typically in the gas phase, in response to a density gradient in a gravitational field. 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 unit weight of fuel.
Calcite	A crystalline mineral composed of calcium carbonate ($CaCO_3$).
Calihration	The process of comparing the conditions, processes, and parameter values used in a model against actual data points or interpolations (e.g., contour maps) 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 the waste (e.g., high-level radioactive waste immobilized in glass rods) that facilitates handling, storage, transportation, and/or disposal. A metal receptacle with the following purpose: (1) a pour mold for solidified high-level radioactive waste and (2) for spent nuclear fuel, structural support for loose rods, non-fuel components, or containment of radionuclides during postclosure operations.

Capillarity	(1) A phenomenon that results from the force of mutual attraction (cohesion) between water molecules in conjunction with the force of molecular attraction (adhesion) between water and different solid materials. (2) A means by which water will rise in small diameter tubes and, in combination with the effects of gravity, a means of water movement in the unsaturated zone.
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 across which water does not flow.
Capillary Force	In the unsaturated zone, the forces acting on moisture that can be attributed to the attraction between rock grain, or matrix, surfaces and water.
Capillary Pressure	The difference in a fluid pressure at a given point between a nonwetting phase such as air and a wetting phase such as water.
Capillary Suction	A condition in unsaturated rocks in which the attraction of fluids to particle surfaces is stronger than the force of gravity on the fluid.
Carbon Steel	A steel that is tough but malleable and contains a small percentage of carbon. The outer barrier of waste packages is composed of carbon steel.
Carbonate	Any compound formed by the reaction of carbonic acid with either a metal or an organic compound. Any compound containing the carbonate ion.
Carbonation	A chemical process involving the change of concrete and cement into a carbonate.
Carboniferous	Producing, containing, or pertaining to carbon or coal.
Cationic	An atom or group of atoms having a positive charge.
Center For Nuclear Waste Regulatory Analyses	A federally funded research and development center in San Antonio, Texas, sponsored by the Nuclear Regulatory Commission to provide the Nuclear Regulatory Commission with technical assistance for the repository program.
Ceramic Coating	A layer of ceramic material such as alumina that has been applied to a metallic product to protect against extremely high temperatures and corrosion.
Cladding	The metallic outer sheath of a fuel element generally made of stainless steel or 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.
Climate Proxies	The physical remains of substances that carry the imprint of past climates.
Climate States	Representations of climate conditions. Three different climate states are used to represent changes in climate over the time periods of interest: present-day dry climate, long-term-average climate (about twice the precipitation of dry climate), and superpluvial climate (about three times the precipitation of dry climate).
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 two types of waste, such as commercial spent nuclear fuel and defense 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 to limit the reactivity of individual waste packages.
Coefficient of Multiple Determination	See Section A.2 of this glossary.
Colloid	As applied to radionuclide migration, a colloidal system is a group of large molecules or small particles that have at least one dimension with the size range of 10^{-9} to 10^{-6}m that are suspended in a solvent. Naturally occurring colloids in groundwater arise from clay minerals such as smectites and illites. Colloids that are transported in groundwater can be filtered out of the water in small pore spaces or very narrow fractures because of the large size of the colloids.
Colloid-Facilitated, Radionuclide Transport Model	A model that represents the enhanced transport of radionuclides by particles that are colloids.
Commercial Spent Nuclear Fuel	Commercial nuclear fuel rods that have been removed from reactor use.

Committed Dose Equivalent	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.
Committed Effective Dose Equivalent	The sum of the products of the weighting factors applicable to each of the body organs or tissues that are irradiated and the committed dose equivalent to these organs or tissues.
Complementary Cumulative Distribution Function	See Section A.2 of this glossary.
Component Models	The 16 process models that are run separately and then combined for running in the TSPA-VA RIP computer model.
Concentration Gradient	For a substance dissolved in a solute, the change in concentration of the substance over a distance.
Conceptual Model	A set of qualitative assumptions used to describe a system or subsystem for a given purpose. Assumptions for the model should be compatible with one another and fit the existing data within the context of the given purpose of the model.
Concrete Lining	Part of the reference design in which the large majority of emplacement drifts are lined with precast concrete segments.
Conduction	Transport of heat in static groundwater, controlled by the thermal conductivity of the geologic formation and the contained groundwater and described by a linear law relating heat flux to temperature gradient.
Confidence	See Section A.2 of this glossary.
Confidence Interval	See Section A.2 of this glossary.
Consequence	A measurable outcome of an event or process that, when combined with the probability of occurrence, gives risk.
Conservative Assumption	(1) An assumption that has the effect of maximizing the calculated amount of radionuclides released from the hypothetical repository to the accessible environment. (2) An assumption that uses uncertain inputs and does not attempt to include any potentially beneficial effects.
Conservative Tracer	Substances with no retardation effect. See Tracer.
Continuous Random Variable	See Section A.2 of this glossary.
Continuum Model	A model that represents fluid flow through numerous individual fractures and matrix blocks by approximating them as continuous flow fields.

Convection	(1) Thermally driven groundwater flow or a heat-transfer mechanism for a gas phase. The bulk motion of a flowing fluid (gas or liquid) in the presence of a gravitational field, caused by temperature differences that, in turn, cause different areas of the fluid to have different densities (e.g., warmer is less dense). (2) One of the processes that moves solutes in groundwater. See Transport.
Convolution Integral Method	(1) A computational method used to calculate the radionuclide concentration in the saturated zone as it changes with time. (2) The abstraction method for the saturated zone flow and transport component model of the TSPA-VA RIP computer model.
Cooling Regime	One of two divisions (the other being the boiling regime) used to delineate the reactions between the gas, water, and minerals in the rock, which occur as the system cools after heating and boiling of the pore water occurs through time.
Correlation Coefficient	See Section A.2 of this glossary.
Corrosion	The process of dissolving or wearing away gradually, especially by chemical action.
Corrosion Allowance Material	A material that undergoes relatively uniform corrosion penetration at relatively low and predictable rates in moderately acidic, moderately alkaline, and neutral conditions (see pH). Corrosion allowance material is used as the outer barrier of the two-layer, metallic waste-disposal container and is made of carbon steel.
Corrosion Model (for inner barrier and outer barrier)	A model that includes the time histories of first and subsequent pit and patch penetrations for the waste package layers.
Corrosion Resistant Material	A material that develops a protective film on its surface, creating a high resistance to corrosion. This material, usually the nickel-base alloy, Alloy 22 (ASTM B 575 N06022), is used as the inner barrier of the two-layer waste-disposal container.
Coupling	The ability in a performance assessment to assemble separate analyses so that information can be passed among them to develop an overall analysis of system performance.
Covariance	See Section A.2 of this glossary.
Crevice Corrosion	A type of localized corrosion that forms in splits or cracks.
Critical Event	See Criticality.

Critical Group	With regard to annual dose, the maximally exposed individuals. A group of members of the public whose exposure is reasonably homogeneous and includes individuals receiving the highest dose. The individuals making up the critical group may change with changes in source term and pathway.
Criticality	(1) A condition that would require the original waste form, which is part of the waste package, to be exposed to degradation followed by conditions that would allow concentration of sufficient nuclear fuel, the presence of neutron moderators, the absence of neutron absorbers, and favorable geometry. (2) The condition in which nuclear fuel sustains a chain reaction. It occurs when the number of neutrons present in one generation cycle equals the number generated in the previous cycle. The state is considered critical when a self-sustaining nuclear chain reaction is ongoing.
Critical Population	See Critical Group.
Cumulative Distribution Function	See Section A.2 of this glossary.
Cumulative Probability	See Section A.2 of this glossary.
Cumulative Release	The sum of the radionuclide curies released over a certain period at a specific location.
Curie	A unit of radioactivity equal to 37 billion disintegrations per second.
Darcy's Law	Used in hydrology to describe fluid flow in a porous medium. Darcy's Law states that the fluid velocity is directly proportional to the hydraulic gradient between the two locations.
Data	Facts or figures measured or derived from site characteristics or standard references from which conclusions may be drawn Parameters that have been derived from raw data are sometimes, themselves, considered to be data.
Decay	See Radioactive Decay.
Deep Percolation	Precipitation moving downward, below the plant-root zone, toward storage in subsurface strata.
Defense in Depth	The term used to describe the property of a system of multiple barriers to mitigate uncertainties in conditions, processes, and events by employing barriers that are redundant and independent, such that failure in any one barrier does not result in failure of the entire system.
Defense Spent Nuclear Fuel	See DOE Spent Nuclear Fuel.

Department of Energy, U.S. (DOE)	A Cabinet-level agency of the United States federal government charged with the responsibilities of energy security, national security, and environmental quality.
Design Concept	As mentioned in the Energy and Water Appropriations Act, consists of the subsurface repository layout, the engineered barrier segments, and the waste package.
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.
Deterministic	A single calculation using only a single value for each of the model parameters. A deterministic system is governed by definite rules of evolution leading to cause and effect relationships and predictability. Deterministic calculations do not account for uncertainty in the physical relationships or parameter values.
Diffusion	(1) The spreading or dissemination of a substance. (2) The gradual mixing of the molecules of two or more substances due to random thermal motion.
Diffusive Transport	Movement of solutes due to their concentration gradient. The process in which substances carried in groundwater move through the subsurface by means of diffusion because of a concentration gradient.
Diffusivity	A measure of the rate of heat diffusion. It varies with the nature of the involved atoms, the structure, and changes in temperature.
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.
Dimensionality	Modeling in one, two, or three dimensions.
Dimensionality Abstraction	An abstraction in which there is a change in the dimensions of a problem, such as from three dimensional to two dimensional, for modeling purposes. This is done either to simplify the problem or reduce the computational requirements of the problem to implement modeling results in a more efficient or usable form.
Discrete Heat Source	An attribute of drift-scale thermal hydrology models in which the model includes a representation of heat output for discrete waste packages with varying heat outputs depending on the type and amount of waste in the package.

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 (defection) moved it. The tortuous path the solute follows through openings (pores and fractures) causes part of the dispersion effect in the rock. (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.
Disposal Container	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 disposal container is designed to contain spent nuclear fuel and high-level radioactive waste, but exists only until the outer lid weld is complete and accepted. The disposal container does not include the waste form or the encasing containers or canisters (e.g., high-level radioactive waste pour canisters, DOE spent nuclear fuel co-disposal canisters, multi-purpose canisters of spent nuclear fuel, etc.).
Dissolution	Change from a solid to a liquid state. Dissolving a substance in a solvent.
Distribution	See Section A.2 of this glossary.
Distribution Frequency	See Section A.2 of this glossary.
Disturbed Performance	Performance that is expected for the system if perturbed by disruptive events such as human intrusion, natural phenomena such as volcanism, or nuclear criticality. This is as used in a description of scenario.
Disruptive Event	An unexpected event that, in the case of the repository, includes human intrusion, volcanic activity, seismic activity, and nuclear criticality. Disruptive events have two possible effects: (1) direct release of radioactivity to the surface or (2) alteration of the nominal behavior or the system.
Domain (Model)	(1) The set of elements that a mathematical model describes. (2) Individual process areas, such as the unsaturated zone flow domain.
DOE Spent Nuclear Fuel	Radioactive waste created by defense activities that consists of over 250 different types of spent nuclear fuel and is expected to contribute 2,333 metric tons of heavy metal (MTHM) to the total repository. The major contributor to this waste form is the N-reactor fuel currently stored at the Hanford Site. This waste form also includes 65 MTHM of U.S. Navy spent nuclear fuel.

Dose	The amount of radioactive energy that passes the exchange boundaries of an organism (e.g., skin and mucous membranes) and is taken into living tissues. Dose arises from a combination of the energy imparted by the radiation and the absorption efficiency of the affected organism or tissues. It is expressed in terms of units of the radiation taken in, the body weight or mass impacted, and the time over which the dose occurs or the impact is measured.
Dose Conversion Factor	(1) Any factor used to change an environmental measurement to dose in the appropriate units. (2) The multipliers that convert an amount of radionuclides ingested or inhaled to an estimate of dose.
Dose Equivalent	The product of the absorbed dose in tissue, quality factor, and all other necessary modifying factors at the location of interest. See also Effective Dose Equivalent and Total Effective Dose Equivalent.
Dose Rate	An organism's exposure to radiation over time.
Downgradient	An area toward which water will tend to flow as the result of several factors. The most important factor is the elevation of water levels in wells in that area relative to other areas. The downgradient is the direction in which contaminants released from the potential repository at Yucca Mountain and migrating in the saturated zone might be expected to move. Based on current understanding of the hydraulic gradient below Yucca Mountain, downgradient is toward the south to southeast of the potential repository location in the area within about 5 km.
Drift	From mining terminology, a horizontal underground passage. The nearly horizontal underground passageways from the shaft(s) to the alcoves and rooms. Includes excavations for emplacement (emplacement drifts) and access (access mains).
Drift Scale	The scale of an emplacement drift, or approximately 5 m in diameter.
Drip Shield	A sheet of impermeable material placed above the waste package to prevent seepage water from directly contacting the waste packages.
Dripping Conditions	Assumed for a certain fraction of the waste packages based on water seepage into a drift. The following set of assumptions apply: (1) a small number of the waste packages will be emplaced in drifts with fractures that periodically drip water, and water may drip on a certain fraction of these packages after emplacement; (2) if water drips onto a waste package, it is 100 percent wet from the dripping; and (3) the dripping rate, frequency of drip periods, and water chemistry (especially pH and chloride concentration) will contribute significantly to waste package degradation.

Dry Climate	One of three sets of conditions used to represent climate changes through time. Representative of current climate conditions at Yucca Mountain. See also Long-Term-Average Climate and Superpluvial Climate.
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.
Dual Permeability/Weeps Model	A dual-permeability approximation of the Weeps Model. Also see Dual Permeability Conceptual Model and Weeps Model.
Edge Effects	Conditions at the edges of the repository that are cooler and wetter because heat dissipates more quickly than at the center of the repository.
Effective Dose Equivalent	The sum of the products of the dose equivalent to the organ or tissue and the weighting factors applicable to each of the body organs or tissues that are irradiated.
Effective Porosity	The fraction of a given medium's porosity available for fluid flow and/or solute storage, as in the saturated zone.
Electric Power Research Institute	A nonprofit organization that serves as a research and development consortium serving the entire power industry, from power generation to delivery, to end use products and services. This group has performed an independent performance assessment on the Yucca Mountain site.
Elicitation	See Expert Elicitation.
El Niño	A complex set of changes in the water temperature in the Eastern Pacific equatorial region, producing a warm current. This occurs annually to some degree between October and February, but in some years intensifies and causes unusual storms and destruction of marine life.
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 Drift	See Drift.

Energy Policy Act of 1992, Public Law 102-486	Comprehensive energy legislation enacted in 1992. Section 801 of the Act directs the U.S. Environmental Protection Agency (EPA) to contract with the National Academy of Sciences to provide "findings and recommendations on reasonable standards...that would govern the long-term performance of a repository at the Yucca Mountain site." The EPA Administrator is to promulgate public health and safety standards after the receipt of the findings and recommendations of the National Academy of Sciences, and these shall be the only standards applicable to the Yucca Mountain site.
Energy and Water Development Appropriations Act of 1997, Public Law 104-206	Legislation that provided that "no later than September 30, 1998, the Secretary shall provide to the President and to the Congress a Viability Assessment of the Yucca Mountain site. The VA shall include: • the preliminary design concept for the critical elements for the repository and waste package • a total system performance assessment (TSPA), based upon the design concept and the scientific data and analysis available by September 30, 1998, describing the probable behavior of the repository in the Yucca Mountain geological setting relative to the overall system performance standards • a plan and cost estimate for the remaining work required to complete a license application, and • an estimate of the costs to construct and operate the repository in accordance with the design concept."
Engineered Barrier Segments	As mentioned in the 1997 Energy and Water Appropriations Act, include (1) the invert and pedestal systems to support the waste package, (2) any packing or backfill materials that may be used within the drift, (3) and any drip shield that may be placed over or around the waste package.
Engineered Barrier System	The waste packages and the underground facility. The designed, or engineered, components of the disposal system and the waste package.
Engineered Barrier System Transport Model	A computer model that includes the key processes: (1) in-drift thermal hydrology and geochemistry, (2) degradation of the drip shield (if used), (3) degradation of the waste package and cladding, (4) alteration and dissolution of the waste form, (5) degradation of the invert, (6) mobilization of the radionuclides in the waste form, and (7) transport of radionuclides in the drift.
Enrichment	The percentage of the fuel matrix that is fissile.

Environmental Impact Statement (EIS)	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 (NEPA). The environmental impact statement describes: "...the environmental impact of the proposed action; any adverse environmental effects which cannot be avoided should the proposal be implemented; alternatives to the proposed action (although the Nuclear Waste Policy Act, as amended, precludes consideration of certain alternatives); the relationship between local short-term uses of man's environment and the maintenance and enhancement of long-term productivity; and any irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented." 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. A final environmental impact statement for the Yucca Mountain site is to be published in fiscal year 2000.
Environmental Protection Agency (EPA), U.S.	The agency charged by the Nuclear Waste Policy Act of 1982, and subsequently by the Energy Policy Act of 1992, with promulgating generally applicable standards for protection of the general environment. The proposed repository at Yucca Mountain is overseen by this agency.
Equilibrium	The state of a chemical system in which the phases do not undergo any spontaneous change in properties or proportions with time, a dynamic balance.
Equilibrium Batch Reactor	A concept describing the conditions in a computer model cell in which the value of any given parameter is homogeneous and in equilibrium throughout the cell area. Used when referring to concentration conditions within an individual cell during modeling of engineered barrier transport.
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 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, such as volcanic activity, and each branch from that node represents one of its possible outcomes. Each branch can continue to branch many times. Each possible pathway along the tree, from beginning to end of a given line of branching, represents a specific scenario.
Events	(1) Occurrences that have 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.
Expected Behavior	The nominal behavior of the repository system and the geologic barrier in the absence of disruptive events.
Expected Value	See Section A.2 of this glossary.
Expected Value Realization	The single realization derived by sampling all uncertain input parameters in the component models at the expected values of their ranges.
Expert Elicitation	A formal process through which expert judgment is obtained.
Exploratory Studies Facility	An underground laboratory at Yucca Mountain that includes a 7.9-km (4.9-mile) main loop (tunnel), a 2.8-km (1.75-mile) cross-drift, and a research alcove system constructed for performing underground studies during site characterization. The data collected will contribute toward determining the suitability of the Yucca Mountain site. Some or all of the Exploratory Studies Facility may eventually be incorporated into the repository.
External Criticality	A condition in which a critical configuration of fissile material occurs after this material is released from the waste packages. See also Criticality.
Far-Field	With reference to processes, those occurring at the scale of the mountain. The area of the geosphere and biosphere far enough away from the repository that, when numerically modeled, releases from the repository are represented as a homogeneous, single-source effect.
Fast Path	Localized unsaturated zone flow pathways that might have high advective velocities. Fast paths move water, carrying radionuclides, through the unsaturated zone more quickly than if movement were predominantly through the pores of the rock matrix. Fractures are potential fast paths.
Fault (Geologic)	A fracture in rock along which movement of one side relative to the other has occurred.

Features	Physical, chemical, thermal, or temporal characteristics of the site or repository system.
FEHM Computer Code	The Finite Element Heat and Mass transfer computer code that is a process model for unsaturated flow and transport.
Fick's Law	The mass of solute diffusing is proportional to the concentration gradient when a solute in water moves from an area of greater concentration toward an area of lesser concentration by molecular diffusion.
Film Flow	Movement of water as a thin film along a surface.
Finite Difference Computer Code	A commonly used numerical method for solving flow problems. An approximating technique in which algebraic equations are used for approximating the partial differential equations that comprise mathematical models in order to produce a form of the problem that can be solved on a computer. For this type of approximation the real world area being modeled is formed into a grid with cubical or rectangular blocks. Values for parameters, such as head, are computed at the grid nodes with the same value also being the average for the area surrounding the node.
Finite Element Computer Code	A commonly used numerical method for solving flow problems. An approximating technique in which algebraic equations are used for approximating the partial differential equations that comprise mathematical models in order to produce a form of the problem that can be solved on a computer. For this type of approximation the real world area being model is formed into a grid with irregularly shaped blocks. This method provides an advantage in handling irregularly shaped boundaries, internal features such as faults, and simulation of point sources (of contamination), seepage faces, and moving water table elevations. Values for parameters are frequently calculated at nodes for convenience, but are defined everywhere in the blocks by means of interpolation functions.
Fissile	Sometimes used as a synonym for fissionable (see Fission). Fissile material can undergo 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. Fissionable nuclides require fast neutrons to undergo fissions.
Fissile Material	See Fissile.
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 Products	A complex mixture of nuclides produced by the process of fission that includes radioactive (and some nonradioactive nuclides) as well as the daughter products of the radioactive decay of these nuclides, which can result in more than 200 isotopes.
Flow	The movement of a fluid such as air or water. Flow and transport are groundwater processes that can move potential contaminants; it usually means flow based on Darcy's law.
Flow Pathway	The subsurface course that a water molecule or solute (including radionuclides) would follow in a given groundwater velocity field governed principally by the hydraulic gradient.
Flux	The rate of transfer of fluid, particles, or energy passing through a unit area per unit time. For water, also known as specific discharge.
Fractures	Breaks in rocks caused by the stresses that cause folding and faulting. 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. Fractures may act as fast paths for groundwater movement.
Fracture Aperture	(1) The space that separates the sides of a fracture. (2) The measured width of the space separating the sides of a fracture.
Fracture Permeability	The capacity of a rock to transmit fluid that is related to fractures in the rock.
Fracture-Matrix Exchange Coefficient	(1) A multiplier used in unsaturated groundwater flow simulations that alters the geometric conductance between fracture and matrix elements to account for reduced wetting and contact area. (2) A coefficient that assists in capturing the effect of groundwater being distributed unevenly over fracture surfaces as moves through fractured rock.
Frequency Distribution	See Section A.2 of this glossary.
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 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	Pertaining to an electrochemical process in which electron flow is produced between two dissimilar metals when they are immersed in an electrolyte solution and placed in contact or are electrically connected. The electron flow results from the difference in electrical potential of the metals.

Galvanic Corrosion	Electrochemical corrosion (eating into a substance) 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.
Gaseous Diffusion	The selective transfer of gas by molecular diffusion through microporous barriers. Used to refer to the mechanism for movement of gas through concrete and rock and for movement of gas out of the waste package by means not involving water.
GENII	A deterministic computer software code that evaluates dose from the migration of radionuclides introduced into the accessible environment, or biosphere, that may eventually affect humans through ingestion, inhalation, or direct radiation. It is used to develop biosphere dose conversion factors.
GENII-S	A quasi-stochastic computer software code that can create distributions and sample them and is run in conjunction with GENII for biosphere modeling.
Geochemical	The distribution and amounts of the chemical elements in minerals, ores, rocks, soils, water, and the atmosphere, and the circulation of the elements in nature on the basis of their properties.
Geochemistry	The study of the abundance of the elements and atomic species (isotopes) in the earth. Geochemistry, or geochemical study looks at systems related to chemicals arising from natural rock, soil, soil processes such as microbe activity, and gases, especially as they interact with man-made materials from the repository system. In the broad sense, all parts of geology that involve chemical changes.
Geologic-Framework Model	A nonmathematical model of the geologic system.
Geologic Repository	A system for disposing of radioactive waste in excavated geologic media, including surface and subsurface areas of operation, and the adjacent part of the geologic setting that provides isolation of the radioactive waste.
Geologic Time	The period of time over which the earth has existed. The time scale over which geologic processes produce change. In general discussion, the term geologic time implies very long periods of time such as tens of thousands of years, hundreds of thousands of years, or millions of years.
Geosphere	The combination of the earth's rock, water, and air layers (spheres).
Glass	See High-Level Radioactive Waste Glass.

Goethite	An iron oxide mineral that is yellowish, reddish, or brownish black. It is the most common constituent of many forms of natural rust or of limonite.
Gradient	The change in value of a quantity per unit distance in a specified direction.
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. Means the same as specific discharge.
Groundwater Travel Time	The time required for a unit volume of groundwater to travel between two locations. The travel time is the length of the flow path divided by the velocity, where velocity is the average groundwater flux divided by the effective porosity along the flow path. If discrete segments of the flow path have different hydrologic properties, the total travel time will be the sum of the travel times for each discrete segment.
Handling Container	The container in which the fuel matrix and cladding are placed. If the waste form is solidified, this is called a pour container. In some cases, this is the only container for storage, handling, and transportation prior to disposal.
Heavy Metal	All uranium, plutonium, and thorium used in a nuclear reactor.
Herbivore	An organism that feeds on plants, especially an animal whose diet is exclusively plants.
Heterogeneity	The condition of being composed of parts or elements of different kinds. A condition in which the value of a parameter such as porosity, which is an attribute of an entity of interest such as the tuff rock containing the repository, varies over the space an entity occupies, such as the area around the repository, or with the passage of time.
High-Level Radioactive Waste	(1) 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 (2) Other highly radioactive material that the Nuclear Regulatory Commission, consistent with existing law, determines by rule requires permanent isolation.
High-Level Waste	See High-Level Radioactive Waste.
High-Level Radioactive Waste Glass	The waste form of defense high-level radioactive waste in which the radioactive waste is mixed with borosilicate glass.
Histogram	See Section A.2 of this glossary.

Homogeneous	Consisting of or composed of similar elements or ingredients.
Host Rock	The rock unit in which the repository will be located. For the repository at Yucca Mountain, the host rock would be the middle portion of the of the Topopah Spring tuff formation of the Paintbrush Group. See also tuff.
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 strength of the gravitational field.
Hydraulic Gradient	The change in the height of water levels with respect to the distance between two locations.
Hydrodynamic Dispersion	See Dispersion.
Hydrogeology	A study that encompasses the interrelationships of geologic materials and processes involving water.
Hydrologic	Pertaining to the properties, distribution, and circulation of water on the surface of the land, in the soil and underlying rocks, and in the atmosphere.
Hydrology	(1) The study of water characteristics, especially the movement of water. (2) The study of water, involving aspects of geology, oceanography, and meteorology.
Hydrostratigraphy	A stratigraphic classification of layered rocks based on rock characteristics and the hydrologic, or water-conducting, properties of the units.
Human Intrusion	The inadvertent disturbance of a disposal system by humans that could result in release of radioactive waste. Subpart B of 40 CFR 191 requires that performance assessments consider the possibility of human intrusion.
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.
Incolloy 625	Under past reference design specifications, the corrosion resistant inner layer of the two layer metallic disposal container. The inner layer material for calculations related to disruptive scenarios.

Infiltration	The process of water entering the soil at the ground surface and the ensuing movement downward when the water input at the soil surface is adequate. 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.
Infiltration Flux	Volumetric infiltration rate per unit area.
Infiltration Rate	See Infiltration Flux.
Inner Barrier	An inner layer of the two-layer metallic disposal container. The inner layer is constructed with corrosion-resistant material such as Alloy 22 (ASTM B575 N06022), a nickel-base, high- performance material. See also Corrosion Resistant Material.
Inner Canisters	High-level radioactive waste canisters placed within the overpack.
In Situ	In its natural position or place. The phrase distinguishes in-place experiments, conducted in the field or underground facility, from those conducted in the laboratory.
Integral-Finite-Difference Computer Code	A commonly used numerical method for solving flow problems. An approximating technique in which algebraic equations are used for approximating the partial differential equations that comprise mathematical models in order to produce a form of the problem that can be solved on a computer. Similar in capability to a finite element code in that it can handle irregularly shaped areas well. See Finite Element Computer Code.
Inventory	The amount of radioactive elements in a fuel, usually stated in curies per metric ton of heavy metal. Also termed radionuclide inventory.
Invert	A construction associated with the precast concrete structure for the purpose of providing a level drift floor and enabling transporting and support of the waste package.
Ion	(1) An atom that contains excess electrons or is deficient in electrons, causing it to be chemically active. (2) An electron not associated with a nucleus.
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.
Ionic Strength	A measure of the level of electrical force in an electrolytic solution.
Irradiated Fuel	Burned fuel. See also Burnup.
Isothermal	Pertaining to constant temperature.

Isotope	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 also Anisotropy.
Iterative	Conditions or results that are repeated in an analysis. The processes in which analysts rerun calculations or refine models as new data are gathered or new insights occur.
ITOUGH2 Computer Code	A computer code that estimates hydrogeologic model parameters for the numerical simulator TOUGH2.
J-13 Water	The groundwater taken from Wellbore J13. The chemical composition of this water is used as the standard for Yucca Mountain ambient groundwater composition for modeling purposes.
Joint	A fracture in rock, usually more or less vertical to bedding, along which no appreciable movement has occurred.
Juvenile Failure	Premature failure of a waste package because of material imperfections or damage by rockfall during emplacement.
Key Technical Issues	Issues important for assessing the long-term safety of a potential Yucca Mountain repository, as defined by the Nuclear Regulatory Commission (NRC). The issues are (1) Support Revision of the U.S. Environmental Protection Agency Standard/NRC Rule Making; (2) TSPA and Technical Integration; (3) Igneous Activity; (4) Unsaturated and Saturated Flow Under Isothermal Conditions; (5) Thermal Effects on Flow; (6) Container Life and Source Term; (7) Structural Deformation and Seismicity; (8) Evolution of Near-Field Environment; (9) Radionuclide Transport; (10) Repository Design and Thermal-Mechanical Effects.
Kinetic	Of or due to motion.
Latin Hypercube Sampling	A sampling technique that divides the cumulative distribution function into intervals of equal probability and then samples from each interval.
License Application	An application to the Nuclear Regulatory Commission for a license to construct a repository.
Line Loading Repository Design	A waste emplacement design in which waste containers are spaced very closely along the drift, with emplacement drifts relatively far apart.

Lithophysal	Pertaining to tuff units with lithophysae, voids having concentric shells of finely crystalline alkali feldspar, quartz, and other materials that were formed due to entrapped gas that later escaped.
Lithosphere	The earth's crust, as distinguished from the atmosphere or hydrosphere, and as distinguished from the deeper portion of the earth underlying the crust.
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. See Pitting Corrosion and Crevice Corrosion.
Log Normal Distribution	A distribution of a random variable X such that the natural logarithm of X is normally distributed.
Long-Term-Average Climate	One of three sets of 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. See also Dry Climate and Superpluvial 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.
Longitudinal Dispersion	(1) Dispersion of a solute moving in groundwater in the same direction as the groundwater flow path. (2) Spreading of a solute in the direction of bulk flow.
Magma	Molten or partially molten rock material that is naturally occurring and is generated within the earth.
Mass Balance	The procedure of accounting for conservation of mass, such as the mass of radionuclides released from waste packages, in real world processes or in models of real world processes.
Mathematical Model	A mathematical description of a conceptual model.
Matrix	Tuff rock material and its pore space exclusive of fractures. As applied to Yucca Mountain tuff, the groundmass of an igneous rock that contains larger crystals.

Matrix Diffusion	As used in TSPA-VA conceptual models, 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.
Mean (Arithmetic)	See Section A.2 of this glossary.
Mechanistic Analysis	An analysis of processes that is based on the well established fundamentals of the processes considered, such as: thermodynamics, reaction kinetics, mass transfer laws, heat transfer laws, etc. This is as opposed to empirical analysis, which is based on a model that has been developed from the numerical value of data taken from tests or measurements of the model.
Median	A value such that half of the observations are less than that value and half are greater than the value.
Meteorological	Of, or relating to meteorology, or to weather and other atmospheric phenomena.
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. When used in the Civilian Radioactive Waste Management Program, the term usually pertains to heavy metals in spent nuclear fuel in scientific text. In this document, MTHM is equal to MTU (metric tons of uranium).
Metric Ton of Uranium (MTU)	A metric ton, which is 1,000 kg, or 2,205 lb, of uranium in scientific text.
Microbe	An organism too small to be viewed with the unaided eye. Examples of microbes are bacteria, protozoa, and some fungi and algae.
Microbially Influenced Corrosion	Corrosion of the waste package that is enhanced by the activity of microbes.
Microbiologically Influenced Corrosion	See Microbially Influenced Corrosion.
Migration	Radionuclide movement from one location to another within the engineered barrier system or the environment.
Mild Steel	See Carbon Steel.
Mineral Assemblage	Minerals that compose a rock, especially an igneous or metamorphic rock. The term includes the different kinds and relative abundance of minerals but excludes the texture and fabric of the rock.

Mineralogical	Of or relating to the chemical and physical properties of minerals, their occurrence, and classification.
Mobile Radionuclides	Radionuclides that can move within a water system with little or no retardation.
Mobilization	The process of breaking down the waste form and releasing radionuclides. After its initial mobilization a radionuclide can be removed from transport by being precipitated or adsorbed and later become remobilized in a cycle of changes that can be repeated many times.
Model	A depiction of a system, phenomenon, or process including any hypotheses required to describe the system or explain the phenomenon or process.
Molal	Of a solution, containing one mole of solute per one kilogram of solvent.
Mole	The fundamental SI unit used to measure the amount of a substance. Avagadro's number of particles (6.023×10^{23}).
Monte Carlo Uncertainty Analysis	See Section A.2 of this glossary.
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.
National Academy of Sciences	A congressionally chartered, private, nonprofit, self-perpetuating organization of scientists devoted to the expansion of science and its use for the general welfare. This organization is mandated to advise the Federal Government on scientific and technical matters. Section 801 of the Energy Policy Act of 1992 directed the U.S. Environmental Protection Agency to contract with the National Academy of Sciences to provide, "findings and recommendations on reasonable standards...that would govern the long-term performance of a repository at the Yucca Mountain site."
National Research Council	The working arm of the National Academy of Sciences and the National Academy of Engineering that carries out most of the studies done on behalf of the academies. Most of the studies are done in response to specific questions presented by federal agencies or Congress.