

tens-of-thousands to about 300,000 years; and Np-237 and Pu-242 were dominant in the period from 300,000 to one million years.

7.3.2.3 Key Features of the TSPA-VA Base Case Models

This section summarizes key features of the performance factors and computer codes that were used to implement the TSPA-VA. The descriptions are based on information contained in DOE98, Volume 3, Section 4. Highly detailed discussions of the performance factors were provided in the chapters of the Technical Basis Document for the VA (DOE98a), and in topical reports that were discussed as references in the Technical Basis Document chapters.

Climate

The TSPA-VA assumed there would be three characteristic climate regimes in the future at Yucca Mountain, with periodic recurrence intervals: dry (current conditions), long-term average, and superpluvial. Present conditions were assumed to prevail for the next 5,000 years. Long-term average conditions were assumed to persist for 90,000 years each time they occur, and superpluvial periods were assumed to last for 10,000 years.

Average precipitation rates in the long-term average and superpluvial periods were assumed to be two and three times, respectively, higher than present rates, which average about 170 mm/yr. Two superpluvial periods, in which glaciation is at a maximum and temperatures are a minimum, were assumed to occur in the next million years: one at about 300,000 years and the other at 700,000 years. Between the superpluvials, the 5,000-year dry periods and the 90,000-year long-term average periods alternate. Under these assumptions, about 90 percent of the next million years experiences the long-term average climate.

The water-table level was assumed to respond to the changes in precipitation, rising by 80 meters from present levels during long-term average climates and 120 meters during the superpluvial periods. One of the modeling consequences of the water-table rise is that the UZ flow path length is shortened.

Unsaturated Zone Flow and Infiltration

On the basis of site characterization data, the repository footprint was divided into six UZ flow and infiltration zones. Three-dimensional steady-state flow models were developed for fracture and matrix flow under current climate conditions and were extrapolated to the wetter climate

conditions. Average infiltration rates for the present, long-term average and superpluvial climate conditions were assumed to be 7.7, 42, and 110 mm/yr, respectively. The infiltration rates were therefore assumed to increase by factors of about 6 and 14 from the present rate, even though the precipitation rate increases only by factors of 2 and 3.

Drift Scale Seepage

Characterization of seepage into the drifts was based on modeling of a three-dimensional, heterogeneous fracture continuum surrounding the drifts. The seepage flow rate and fraction of the packages that are affected by seeps were modeled in terms of percolation flux, i.e., the water flux that arrives at the repository horizon after infiltration at the surface and flow through the UZ above the repository. Percolation flux was characterized for each of the six regions of the repository footprint and the three climate conditions, based on site data and the climate model.

The modeling showed that about 10 percent of the waste packages would be exposed to seeps during the dry-climate period, 30 percent would be exposed to seeps during the long-term average climate conditions, and 50 percent would be exposed during the superpluvial periods. The estimates of the fraction of the packages exposed to seeps had a very high uncertainty range in the TSPA-VA evaluations.

Thermal Hydrology

Thermal hydrology addresses the temporal and spatial impact of the spent fuel heat output on the natural system geologic and hydrologic characteristics and on the performance of the engineered features of the repository. Thermal hydrology models are used to calculate temperatures (waste package surface, waste form, drift wall) and relative humidities in the drifts. Values for these parameters provide information needed for other models such as the waste package degradation model and the near-field geochemical environment models. Standard models of heat transfer, and data concerning the physical properties of repository system materials, are used to characterize the thermal parameters.

Near Field Geochemical Environment

The near-field geochemical environment models calculate the time-dependent evolution of the gas and water compositions that interact with the waste package, the waste form, and other materials in the drift. The evolution of changes in gas and water composition is modeled as a sequence of steady-state conditions. The chemical, thermal, hydrologic, and mechanical factors

important to the near field environment are in reality coupled, but an integrated model of the coupling and its effects was not developed for the TSPA-VA.

Five separate but interacting models were used in the TSPA-VA to characterize the near field geochemical environment:

- Gas, water, and colloid compositions as they enter the drift
- Composition of the in-drift gas phase
- Chemistry of in-drift interactions of water with the solids and gases in the drift
- In-drift colloid compositions
- In-drift microbial communities

The near-field geochemical environment models are connected to other component models (see Figure 7-37). The near-field models receive input from the UZ and thermal hydrology models and from design parameters; they provide outputs to the waste package corrosion model, the waste form model, the UZ radionuclide transport model, and the nuclear criticality model.

Waste Package Degradation

Modeling of waste package degradation was based on waste type contained in the package, whether the packages were dripped on or not dripped on, and their location in the repository. Seepage into the drifts is modeled as a function of the infiltration rate of water and the fracture properties of the rock. With the expected percolation flux, only about one-third of the waste packages are dripped for most of the one million year modeling period. If water seeps onto the surface of a waste package, 100 percent of the surface is assumed to be wetted. Uncertainty in the corrosion rate of the Alloy 22 corrosion-resistant barrier for the waste package wall was also modeled. Corrosion of waste package materials was assumed to occur via pits and patches that always encounter seeping water. Uncertainty in waste package manufacturing defects was also addressed. The model used in the TSPA-VA assumed for the basic case that a single juvenile waste package failure occurs 1,000 years after disposal.

Cladding Degradation

Mechanisms included in models for degradation of fuel rod cladding on commercial spent nuclear fuel included some pre-disposal failures, creep failure of zircaloy at high temperatures, total failure of rods clad with stainless steel, fuel rod fracture from falling rocks, and long-term general corrosion failure. Breaching of cladding was assumed to expose all of the waste-form surface in the rod to water that had entered the waste package.

Waste Form Degradation and Mobilization

Dissolution of CSNF was modeled to be a function of pH, temperature, and total dissolved carbonate; model parameters were based on experimental data. Dissolution of vitrified high-level defense waste was modeled as a function of surface temperature and water pH, and a dissolution rate constant for metals was used for degradation of the defense spent fuel from the N-Reactor. Under the assumption that all spent fuel is exposed and wetted for rods with breached cladding, the spent fuel would be totally dissolved in about 1,000 years. Dissolution of uranium dioxide fuel is known to result in formation of secondary minerals which can trap species such as Np-237 and reduce their release, but credit for this phenomenon was not taken in the TSPA-VA modeling.

Engineered Barrier System Transport

Transport in the EBS was modeled as a series of connected mixing cells, with one cell combining the waste form and waste package, and three pathway cells representing the invert, in order to reduce numerical dispersion in model calculations. The models did not include factors that could defer and decrease radionuclide release after a waste-package wall is breached, such as low seepage rates and partial seepage into the package interior, and in-package dilution. Sorption and diffusional transport was assumed for radionuclide movement through the concrete invert. Consistent with data which indicated rapid transport of plutonium from the Benham weapon test location on the Nevada Test Site, a small fraction of the plutonium mobilized was assumed to be attached to mobile colloids.

Unsaturated Zone Transport

The radionuclide transport model for the unsaturated zone was based on the flow model for that zone. Three flow fields, corresponding to the three climate conditions, and a dual-permeability geologic regime were assumed. Radionuclide movement was modeled using a three-dimensional particle tracking model. Sorption was assumed to occur for Np-237, Pu-239, and Pu-242. Matrix diffusion and dispersion were also assumed to occur.

Saturated Zone Flow and Transport

Flow in the saturated zone was simulated using a coarsely discretized three-dimensional model which establishes the general plume direction and flow path in the geologic media.

Radionuclide transport was assumed to occur in six one-dimensional stream tubes corresponding to the six area regions defined for the repository footprint. Based on the recommendations of the saturated zone expert elicitation panel, the specific discharge in all stream tubes was assumed to be 0.6 m/yr, and a dilution factor probability range, with a mean value of 10, was assumed to apply to all of the stream tubes.

Biosphere Transport

Water used by the dose receptor was assumed to be drawn from a well 20 km (12 miles) down gradient from the repository. Dilution was assumed not to occur during pumping, so the radionuclide concentration in the water emerging from the well is the same as the stream tube concentration at the withdrawal location. The dose receptor was assumed to receive doses from all biosphere pathways in accord with site-specific dose conversion factors and the water use and life style habits assumed for the receptor. For the TSPA-VA, DOE assumed the dose receptor is a current-day average adult living in Amargosa Valley. A survey was conducted to obtain lifestyle and dietary data for the dose evaluations.

7.3.3 TSPA-VA Results

DOE produced the following categories of TSPA-VA results:

- Deterministic results for the TSPA-VA base case
- Results of uncertainty analyses using Monte Carlo techniques
- Results of analyses to assess the sensitivity of performance to uncertainties in parameter values
- Assessments of the effect of disruptive events on performance
- Assessment of the effect of design options on performance

Collectively, these assessment results address the expected performance of the repository, the role of the various performance factors in producing the expected performance, factors that could alter expected performance, and the uncertainty in expected performance. The repository

performance forecasted for the base case is discussed in Section 7.3.3.1. Uncertainties in the TSPA-VA result are discussed in Section 7.3.3.2.

7.3.3.1 Base Case Expected Repository Performance

The deterministic results for the TSPA-VA base case are responsive to the Congressional mandate for assessment of "...the probable behavior of the repository in the Yucca Mountain geological setting...". These results were a forecast of the dose rate to the average individual located 20 km from the repository, for time periods up to one million years. Graphs showing forecasts of peak doses throughout the million-year time period were produced, and specific dose-rate values were identified and discussed for time periods of 10,000, 100,000 and one million years.

DOE described the results for the deterministic evaluation in which values for all uncertain parameters were set at their expected values as follows (DOE98, Volume 3, p. 4-21):

- “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 50,000 years, with the total dose rate reaching about 5 mrem/year.*
- 3. Within the first million years, neptunium 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. The radionuclide plutonium- 242 is also important during the one million-year time frame and has two peaks, at about 320,000 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 Tc-99 and I-129) corresponding to the assumed climate model for the expected value base-case simulation...these spikes are a result of assumed abrupt changes in water table elevation and seepage through the packages.”*

As shown in Figure 7-37, doses to the receptor 20 km from the repository, as a result of the mobile Tc-99 and I-129 radionuclides, first occur about 3,500 years after disposal. These fission products are dominant because of substantial inventory in CSNF, high solubility in seepage water, relatively low decay rate relative to 10,000 years, and negligible sorption on tuff rocks. The scenario presented in Figure 7-38 results from the assumption that a single juvenile waste-

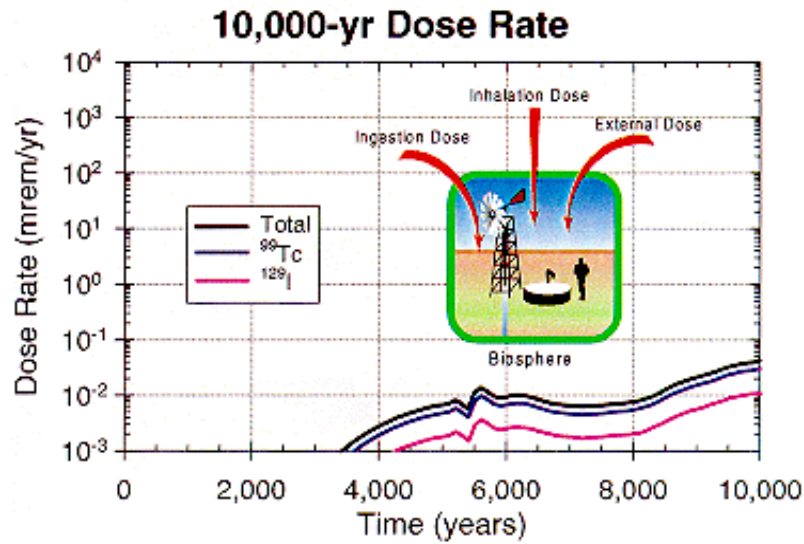


Figure 7-37. TSPA-VA Base Case Dose Rates for Periods Up to 10,000 Years (DOE98).

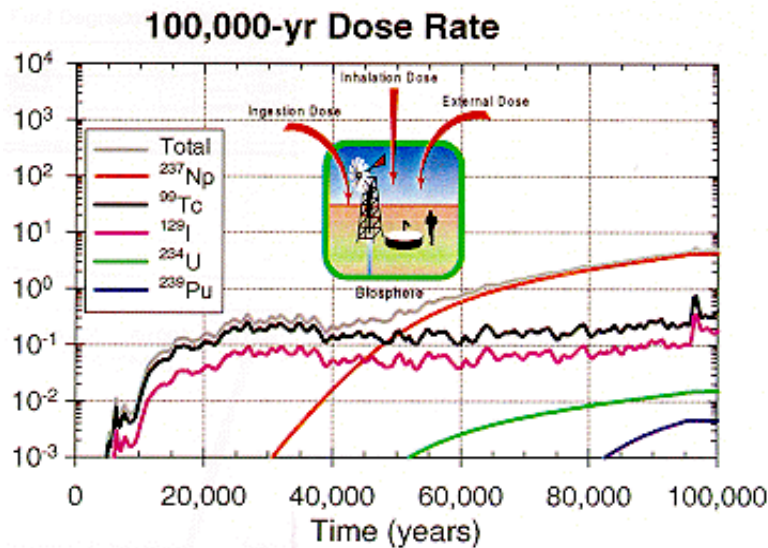


Figure 7-38 TSPA-VA Base Case Dose Rates for Periods Up to 100,000 Years (DOE98) package failure occurs at 1,000 years; the “blip” in the curve at about 5,500 years is the result of the change of climate conditions from dry to long-term average at 5,000 years, which causes a major rise in the water table. During the 10,000-year period, 17 additional packages are modeled to fail at various times, beginning at about 4,200 years. These failures contribute to the dose at

10,000 years in accord with the TSPA-VA model assumptions concerning package failure times and conditions.

Dose rate histories for times up to 100,000 years are shown in Figure 7-38. Tc-99 continues to dominate the dose rate up to about 50,000 years, after which the Np-237 dominates the dose rate out to 100,000 years. There is a relatively large inventory of Np-237 in CSNF resulting from the decay of Am-241. The Np-237 does not begin to appear at the dose location until after about 30,000 years, because its release from the waste form is solubility limited and it exhibits some sorption on the rock surfaces along the transport pathway. The Pu-239 does not begin to appear at the dose location until more than 80,000 years have elapsed because it is more strongly sorbed than the Np-237. A small fraction of the Pu-239 is assumed, however, to be attached to colloids that are not sorbed onto the rock surfaces.

As with the 10,000-year results, the dose rate forecasts for periods to 100,000 years are dominated by climate change assumptions and waste package failure history. The jagged appearance of the Tc-99 curve is the result of individual package failures; each small peak corresponds to a failure. This illustrates one of the key features of the TSPA-VA modeling scheme: because features such as slow drip entry to the package interiors and in-package dilution, which provide storage capacity along the transport path, were not included in the models, the nonsorbing species such as Tc-99 directly track release behavior, and concentrations are simply attenuated by dilution along the pathway. The sorbing and solubility-limited species, such as Np-237 and Pu-239, have the capacity for storage along the transport path because of these properties, but the effects would have been more exaggerated if factors such as in-package dilution had been included in the TSPA models.

As shown in Figure 7-39, Np-237 continues to dominate the dose rate from 100,000 years all the way to the end of the million-year dose evaluation period. At about 300,000 years, Pu-242 becomes the second most important contributor to dose and remains in this role, at a level about a factor of ten less than that of the Np-237, to the end of the dose evaluation period. The contribution of other radionuclides to dose during the long-range time frame is insignificant.

The dose rate after about 300,000 years is seen in Figure 7-39 to be essentially constant. This is because, in the TSPA-VA modeling scheme, the repository as a source term for radionuclides released to the environment goes into essentially steady state. All of the packages that are modeled to fail have failed, the seepage fluxes into the repository and into the packages have

become virtually the same and constant, and the rate of change in exposure of waste form has become constant.

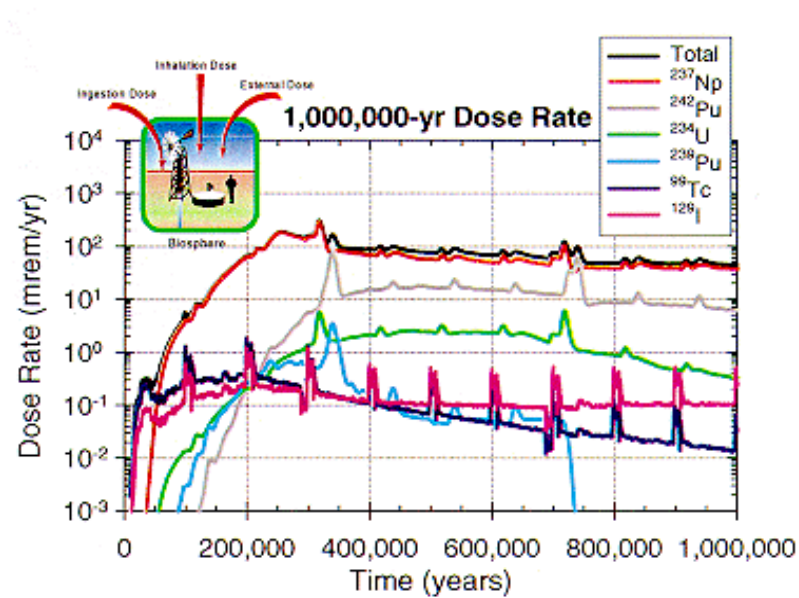


Figure 7-39.
Base Case Dose
Periods Up to
Years (DOE98)

TSPA-VA
Rates for
One Million

The dominant effect of waste package failure history and climate conditions on dose rates continues to the end of the million-year dose evaluation period. At about 200,000 years, cladding degradation begins to contribute to the exposed waste form area, and at times greater than about 700,000 years, waste packages that are never dripped on, which total about 55 percent of the package inventory, begin to fail as a result of low corrosion rates in a non-wetted condition over a very long time frame.

The base case TSPA results for the VA repository show that the performance of the highly complex and multi-element system is strongly dominated by very few factors. In brief:

- Performance is dominated by assumptions concerning waste package failure history and climate, and the effect of these factors on predicted doses is primarily a consequence of the assumptions concerning juvenile package failures and climate change.

- Three nuclides dominate the forecast doses: Tc-99 and I-129 in the shorter time frames and Np-237 in the longer time frames. The dose levels associated with the Np-237 are higher than those associated with the technetium and the iodine, in large measure because the health consequences of a unit quantity of Np-237 are much greater than those for the technetium and iodine.
- The fact that the dose results clearly reflect the occurrence of climate changes and individual package failures shows that the TSPA-VA modeling system is fundamentally simple. Factors in performance that would serve to smooth and smear the consequences of phenomena that change system conditions were omitted from the models.

7.3.3.2 Uncertainty in the TSPA-VA Results

The Monte Carlo type of analyses that were done to assess the uncertainty in the TSPA-VA deterministic base-case results showed an uncertainty range spanning about four to five orders of magnitude throughout the million-year period, as shown in Figure 7-40. These results were obtained by using statistical methods to select values from the distributions for the uncertain parameters used in the TSPA-VA models. For each of the three time frames (i.e., 10,000, 100,000, and one million years) one hundred such runs were done, and a few 1,000-run studies were done to demonstrate that the uncertainty ranges found for the 100-run studies were representative.

The large uncertainty range, i.e., spanning four to five orders of magnitude, is in part due to the many uncertain parameters in the TSPA-VA computer codes. The RIP code alone, for example, contains 177 uncertain parameters, and there are many more in the codes that have inputs to RIP. Another possible cause of the wide uncertainty range is that many of the uncertain parameters themselves have wide uncertainty ranges, either as a result of use of a broad range of possible values because the actual value of the parameter is poorly known, or because the parameter is inherently highly variable. It would be difficult, if not impossible, to sort out the sources and principal causes of the uncertainty range. The uncertainty range for the TSPA-VA results is therefore a consequence of the specific way uncertainty was used in assigning numerical value distributions to parameters in the TSPA-VA models and codes.

Another source of uncertainty, not reflected in the results of the TSPA-VA studies, is the possibility that some of the models used in the codes may not be correct, e.g., because of a sparse

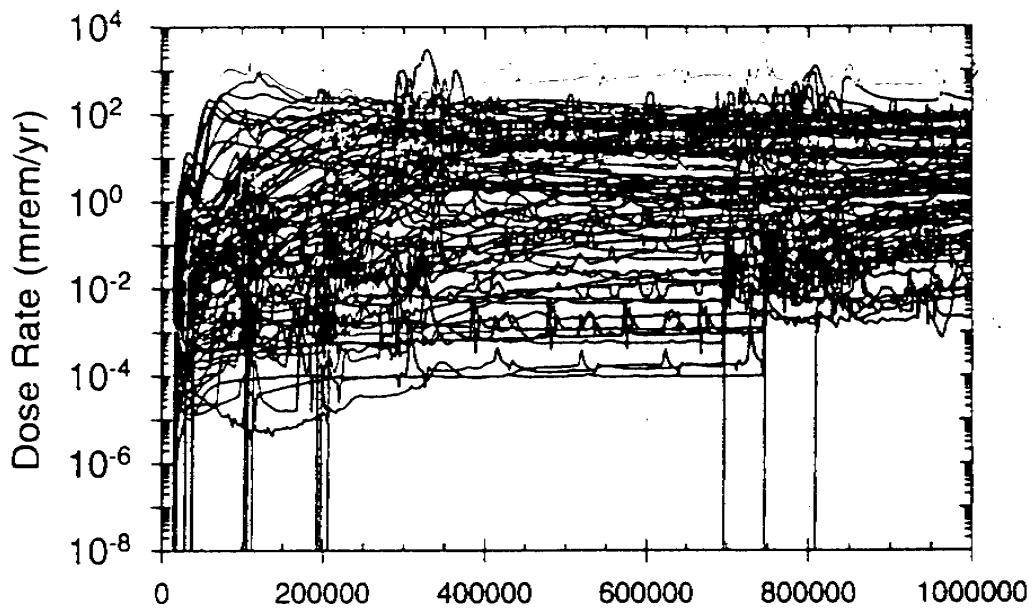
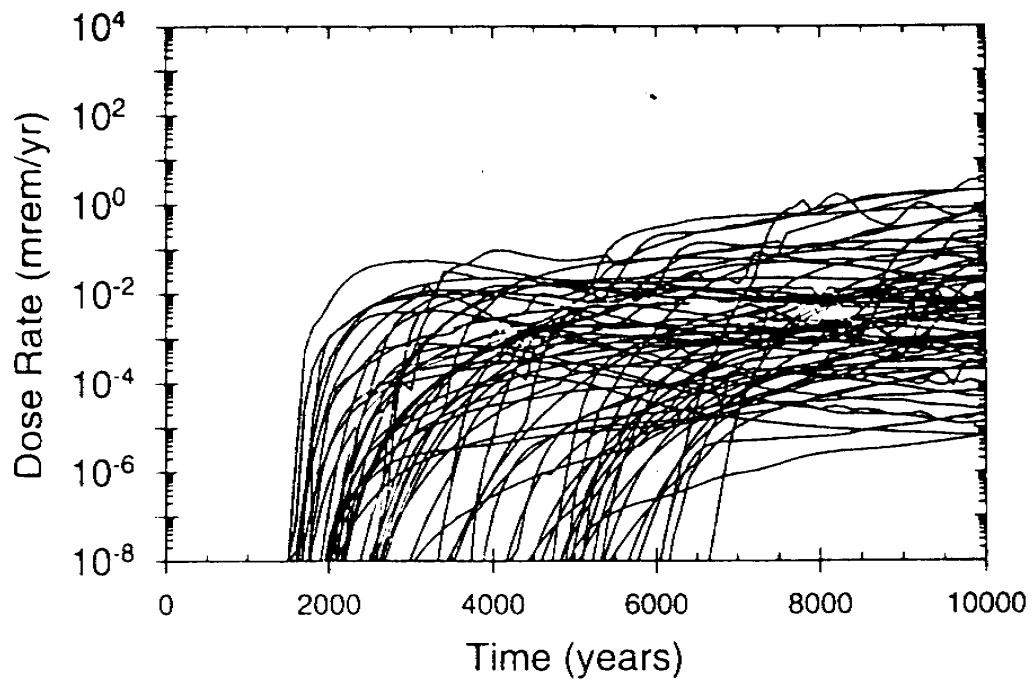


Figure 7-40. Uncertainties in the TSPA-VA Base Case Results (DOE98)

data base, or, as in the case of modeling of the near-field geochemical environment, because coupled phenomena were uncoupled to simplify modeling. This type of uncertainty should be regarded as uncertainty in the conceptual models for the waste containment and isolation systems. In translating conceptual models into calculational models, conservative assumptions are typically made about processes which should be included and how the processes would operate. This is done, in part, for modeling convenience, and, in part, because the level of process complexity cannot be handled manageably. These assumptions can have significant implications for interpreting the results of performance assessments, and should be understood when interpreting the results. (See Section 7.3.3.5 for additional discussion of conservatism in the TSPA-VA modeling.)

In evaluating the status of knowledge and uncertainty as a prelude to selecting further work to improve the TSPA methodology for a License Application (DOE98, Volume 4), DOE often noted that the models used in the TSPA-VA might not adequately capture the full range of possibilities. If this is indeed the case, and the uncertainty in parameters or models has to be expanded in order to embrace the full range of possibilities (as opposed to simply revising the model in response to better information), the uncertainty ranges for future TSPA results might actually be broader.

DOE used a technique known as Stepwise Regression Analysis to determine which of the performance factors were most important to the uncertainty results. These evaluations showed, for the 10,000-year time period, that the fraction of the packages contacted by seepage, the mean Alloy 22 corrosion rate, the number of juvenile failures, and the saturated zone dilution factor are the most important performance parameters. For the 100,000-year period, the most important parameters were the seepage fraction, the mean Alloy 22 corrosion rate, and the variability in the Alloy 22 corrosion rate. For one million years, the most important factors were found to be the seepage fraction, the saturated zone dilution factor, the mean Alloy 22 corrosion rate, and the biosphere dose conversion factors. The fraction of waste packages contacted by seepage water was the dominant performance factor for all three time periods. It is the dominant factor for TSPA modeling of repository system performance because it has a direct effect on the number of waste packages that fail, and it has a very large uncertainty.

Additional sensitivity studies were done to determine the performance factors of secondary importance to the TSPA-VA results. In these analyses, the performance factor of primary importance were held constant, and Monte Carlo runs were done for the other uncertain parameters. The performance factors that were held constant at their baseline values were the infiltration rate and mountain-scale saturated zone flow rates, the fraction of the waste packages

contacted by seepage, the seepage flow rate, the Alloy 22 mean corrosion rate, and the Alloy 22 corrosion rate variability.

With the above parameters held constant, the parameters of principal secondary importance at 10,000 years were found to be the saturated zone dilution factor, the biosphere dose conversion factors, the solubility of technetium, and the fraction of seepage contacting a package that enters a failed package. The factors that were important for 10,000 years were found to be also important for 100,000 years, except that the solubility of neptunium replaced the solubility of technetium as an important factor, and the fraction of saturated zone flow in alluvium was added to the list. At one million years, the most important factors were the saturated zone dilution factor, the cladding failures by corrosion and by mechanical disruption, the biosphere dose conversion factors, the saturated zone longitudinal dispersivity, and the saturated zone alluvium fraction. In all time frames, the most important of these secondary factors was the saturated zone dilution factor.

All of these sensitivity findings reflect the fundamentals of repository system performance: the potential doses depend primarily on the fraction of waste packages intercepted by seepage, the amount of waste form available to be a source of radionuclides, the amount of water available to pick up the radionuclides and to transport them to the environment, the amount of water available to dilute radionuclide concentrations, and the extent and means of interaction of the dose receptor with the contaminated water.

7.3.3.3 Effects of Disruptive Events on Performance

The TSPA-VA evaluated the effects of four types of disruptive events on repository performance: basaltic igneous activity, seismic activity, nuclear criticality, and inadvertent human intrusion. The basis for inclusion of evaluations of the effects of disruptive events on repository performance includes the probability of occurrence of the event, the consequences of occurrence, and any regulatory requirements that mandate or exclude consideration of disturbances.

The igneous activity evaluations considered events in which molten igneous material is cooled within the earth or on the surface. In the case where magma reaches the surface, explosive releases may carry radioactive materials directly into the atmosphere. Cooling of magma within the earth may involve destruction of waste packages so that radionuclides in the waste form are more accessible for release and transport.

Results of the direct-release igneous activity evaluations showed that the maximum dose rate from this volcanism would be about two million times less than for the base case ground water contamination scenario. The underground-cooling scenarios showed that dose rate peaks would occur tens of thousands of years after the actual magma intrusion event.

The seismic activity studies considered phenomena such as rockfall onto waste packages as a result of earthquakes, and the effects of seismicity on the hydrologic regime in the near field and in the saturated zone. These studies showed that rockfalls could not contribute significantly to waste package degradation until after at least 100,000 years and that changes to the hydrologic regimes would be negligible. Overall results of the analyses showed that seismic events would have almost no effect on repository performance over one million years.

The potential for nuclear criticality within waste packages and external to the packages after transport of fissionable material from the package was investigated within the TSPA-VA. The evaluations were done assuming that criticality occurs 15,000 years after emplacement, which is when the commercial spent fuel is most reactive. The analyses determined that criticalities external to the waste packages are not a credible event, and that criticality within a package is extremely unlikely and would have insignificant consequences. Criticality within a waste package is extremely unlikely because only 8 percent of the commercial fuel waste packages contain sufficient fissile material to achieve a critical mass and only 10 percent of the waste packages are expected to be breached in 40,000 years. Breached waste packages must retain sufficient water to act as a moderator for the nuclear chain reaction to be sustained and DOE has estimated that only 25 percent of the breached waste packages will hold water for a period sufficient to flush out boron which is included in the waste package as a neutron absorber. Even if criticality did occur within the waste package, the incremental radioactivity is less than the normal radioactivity from most waste.

In keeping with the recommendations of the NAS panel that developed the technical basis for the Yucca Mountain standards, a stylized human intrusion scenario was characterized and evaluated. Intrusion of a waste package by an 8-inch drill bit, as a result of search for water, was assumed. The bit was assumed to penetrate the package and the mountain stratigraphy to the water table, with large quantities of pulverized fuel being transported to the bottom of the bore hole, which was never sealed. Water would then dissolve the fuel inventory at the bottom of the bore hole and transport radioactive material to the dose receptor location. The intrusion was assumed to occur at 10,000 years, which is the first time at which it is estimated the drill bit could penetrate the package wall. The NAS panel did not feel it would be useful to assess hazards to drillers or

to the public from radioactive materials transported directly to the surface since these risks would be the same for all geologic repositories.

The total amount of fuel deposited at the bottom of the bore hole was assumed to range between 550 and 2,700 kilograms (1,200 and 6,000 pounds), which corresponds to about 5 to 22 percent of the total spent fuel inventory in the package. The actual mass of fuel that would be intercepted by the 8-inch drill would be about 160 kg, so the analyses assumed that large quantities of fuel would be entrained by the bit as it passed through the package.

The analyses for this intrusion scenario showed that the consequent radionuclide releases for the 2,700 kg release would produce a blip in the dose rate curve, in comparison with the base case, that starts at about 11,000 years, peaks at 12,000 years, at levels about 145 times higher than the base case dose rate at that time (i.e., 1 mrem/yr), and returns to base case levels at about 14,000 years. The 550-kg spent fuel release from intrusion produces a dose rate at 12,000 years that is 3.7 times the base case dose rate. All effects of the intrusion on dose rate are gone by 150,000 years. The TSPA-VA observed that the effects of the intrusion on dose rates are significant only for times near the occurrence of the intrusion, and that the maximum resulting dose is 1 mrem/yr.

7.3.3.4 Effects of Design Options on Performance

The TSPA-VA included evaluation of the effect, on repository performance, of design features that were not included in the VA reference design. The three features considered were emplaced drift backfill, drip shields, and ceramic coating of the disposal containers, with backfill. The objective for use of these design options would be to reduce and defer liquid water contact with the waste package.

7.3.3.4.1 Effects of Backfill

The backfill was assumed to be crushed tuff, emplaced 100 years after the end of emplacement operations. The backfill will initially perform as a thermal blanket for the waste packages, and cause a temperature spike of as much as 80-90°C. The temperature spike might cause a slight increase in the waste package corrosion rate, but it would also delay the rate of increase of relative humidity as the heat emissions from the waste packages decrease and the repository system cools. A potentially major effect of backfill would be to change the potential for, and patterns of, seepage water contacting the waste packages. This effect was not modeled in the TSPA-VA analyses.

The analyses for the assumed backfill effects showed that the use of backfill would defer corrosion of the Corrosion Allowance Material, but corrosion of the Corrosion Resistant Material would be virtually unaffected based on the modeling assumption that corrosion of this material is driven by whether or not dripping occurs and the same dripping conditions are assumed for the case of backfill and no backfill. Use of backfill would therefore have little effect on repository performance if the backfill does not reduce or defer contact of seepage water with the waste packages. The backfill might actually have effects such as diverting the seepage water around the waste packages or reducing the amount of seepage that gets to the package as a result of evaporation, but a basis for modeling such effects was not available for the TSPA-VA.

7.3.3.4.2 Effects of Drip Shields

The drip shields were assumed to be made of Alloy 22 and to be 2 cm (0.8 in.) thick. The shields would be shaped like a Quonset hut, shrouding the waste packages but not touching them. The dripshields would be covered with backfill, emplaced 100 years after emplacement of the waste packages was completed. The shields upper surfaces were assumed to be totally wet in dripping regions of the repository, and they were assumed to fail only by general corrosion. After drip shield failure, 10 percent of the waste package area under the failed shield was assumed to be wetted (in contrast, the base case analyses assumed 100 percent of the package surface area would be wetted) because only a small fraction of the drip shield surface area was modeled to fail.

TSPA-VA results based on the above assumptions showed that the drip shields enhanced the overall waste package lifetime by more than 100,000 years. Dose rates for the first 300,000 years are reduced by one to two orders of magnitude in comparison with base case results. After 500,000 years, the drip shield dose projections become the same as those for the base case. The results were interpreted to indicate that the life span if the drip shield is the key determinant of improved performance.

As a result of these findings, drip shields are included as a design feature for the repository design expected to be selected as the reference design for the Site Recommendation and the License Application (see Section 7.2.2.5).

7.3.3.4.3 Use of Ceramic Coating of Disposal Containers with Backfill

This design option involves coating the waste packages with a ceramic material in order to delay corrosion of the outer wall of the packages (in the VA design, A 516 carbon steel). Backfill is added to the repository to protect the ceramic coatings.

Performance of this design concept was modeled assuming that the ceramic coating functions as a barrier to oxygen transport to the carbon steel package wall. For the assumed conditions, the analyses determined that the ceramic coatings would not be breached for more than 300,000 years. Dose rates would not begin until about 500,000 years, and at one million years the dose rates would be nearly two orders of magnitude less than those for the TSPA-VA base case.

If ceramic coatings perform as modeled for the TSPA-VA, they would have a profound effect on repository system performance. At this time, however, there are uncertainties and concerns associated with potential for defects and flaws in the coatings, differential thermal expansion between the coating and the substrate that could result in cracks in the coating, and dissolution of the coating over long time periods. Analysis of these effects is needed before the potential benefits of use of ceramic coatings can be verified.

7.3.3.5 Conservatism In The TSPA-VA Base Case Results

The TSPA-VA base-case results (an expected (average) value dose rate of 0.04 mrem/yr 10,000 years after disposal, to a reference person 20 km downstream) are a consequence of choices that were made concerning performance parameter values, performance models, and assumptions. This section discusses conservatism that was exercised in making the TSPA-VA choices, and the effects of conservatism on the base case results. Similar discussions are provided for the NRC performance assessments (Section 7.3.5.3) and the EPRI assessments (Section 7.3.6.4).

Performance Parameters

The TSPA-VA base-case evaluations used expected values of performance parameters, based on available information. Expected values for some of the parameters, such as the dilution factor for the saturated zone and corrosion rates of Alloy 22, were based primarily on results of expert elicitations because of limited availability of data at the time that the TSPA-VA analyses were performed. The parameter values developed by the expert elicitations may be conservative because the experts are, themselves, working with limited information. Expected values of parameters, and the uncertainty ranges for parameters that are inherently variable, may change in the future as a result of data additions, but the TSPA-VA analyses sought to be as realistic as possible, rather than conservative, in their choices of performance parameter values.

Performance Models

Conservatism in the suite of performance models and computer codes used for the TSPA-VA analyses was introduced by using simplified models and by omitting from the suite of models some performance factors that could have significant impact on predicted doses. Examples of this type of conservatism include:

- Dilution and transport delay for radionuclides released from the waste form but in water still within the failed package were not considered. Under realistic package failure conditions during the first 10,000 years, when disruptive failure scenarios are insignificant, water will fill the package interior very slowly from a penetration in the top. By the time that radionuclide release and in-package transport occurs, temperature gradients will be too low to drive advective transport processes, and temperature levels will be too low for inside-to-outside corrosion of the Alloy 22 to occur and create an exit at the bottom of the package. Radionuclide transport rates within the package will therefore be low, the package interior will have to fill with water in order to enable radionuclides to exit through the same penetration that provides water ingress, and the volume of water to fill the package interior will be available to provide dilution. Radionuclide releases to the exit of the package may therefore be greatly delayed, and concentrations at the package exit would be much lower than for the no-dilution assumption.
- Release of radionuclides from a breached waste package was assumed in the VA models to begin immediately after the waste package was breached, i.e., an exit hole in the metal container was assumed to be created as soon as the container wall was breached by corrosion. In reality there would be a time delay before an exit hole at another location on the container was developed. This time delay could be relatively short if exterior corrosion was taking place concurrently at opposite sides of the container, or it could be very long if, as indicated above, the exit pathway had to develop from inside the container. By delaying the exit of radionuclides the actual containment time of the waste containers would be significantly increased and doses during the regulatory time frame would be consequently decreased.
- Dilution of radionuclide concentrations during transit of the unsaturated zone from the repository to the water table was not considered. When few packages are failed and releasing radionuclides (in the TSPA-VA, only 18 of 10,000 packages are failed at 10,000 years), uncontaminated percolation water adjacent to contaminated streams emanating from the failed waste packages could provide extensive dilution as a result of mixing of contaminated and uncontaminated water in the fracture and matrix flow paths. This mixing would lower the radionuclide concentrations at the start of saturated zone transport and result in lower predicted doses to the receptor.

- A simple, one-dimensional model of radionuclide transport along the saturated zone flow paths from the repository to the dose receptor location was used, and dilution of initial SZ radionuclide concentrations under the repository was assumed to occur at the end of the path, in accord with dilution factors recommended by experts (for the base case, a dilution factor of 10 was used). Processes that could delay and disperse radionuclide transport along the pathway, and therefore would reduce the predicted dose rates to the receptor, were not included in the modeling.
- Dilution during well pumping by the dose receptor was assumed not to occur. This expected dilution process, which is included in NRC modeling of repository performance, would reduce predicted doses to the receptor.

These processes and phenomena were omitted from TSPA-VA modeling of repository performance because at the time the data base for characterizing the relevant performance parameters and their uncertainties was limited or non-existent. Also, the magnitude of these effects is difficult to quantify with high confidence even with site characterization and laboratory work focused on them. However, these processes would be expected to function in the actual repository environment, and reasonable but cautious estimates could be made to support assessments, through a combination of data collection and expert judgment.

Rather than choosing to incorporate models for these processes in the TSPA-VA assessments, with estimated values of the parameters used in the calculations, they were omitted from the suite of TSPA-VA models. This approach had the consequence of producing a spectrum of performance results that are an assessment of a potentially very conservative performance scenario, incorporating some unrealistic modeling assumptions. Omission of these modeling features introduces a significant level of conservatism in the assessment results whereas better performance would reasonably be expected.

Additional data (e.g., additional characterization of the SZ geology and hydrology), may enable inclusion of at least some of these performance factors in the TSPA for the LA. Their omission introduces conservatism to the TSPA results, but also avoids licensing issues that may be difficult to resolve unless a data base adequate to support their use is available.

Conservative Assumptions

The TSPA-VA evaluations included conservative assumptions for some of the key performance factors, as follows:

- In the base case, early failure of a waste package was assumed to occur at 1,000 years as a result of an imperfection such as a poor weld. Performance parameters selected in association with this assumption (e.g., the size of the hole on the package wall) were such that nuclide releases from this single package were a dominant factor in the predicted base case dose rate at 10,000 years.
- The Corrosion Resistant Material for the waste package wall, Alloy 22, was assumed to be penetrated rapidly by crevice corrosion as a result of being under carbon steel in the VA waste package design. This assumption was derived from the waste package expert elicitation, which conservatively interpreted the highly limited data base for the corrosion performance of Alloy 22.
- In characterizing corrosion processes, the TSPA-VA assumed that all ground water seeping into the emplacement drifts contacts the waste packages, even though the package width is only one-third the width of the drift, thereby overstating the amount of water available to cause corrosion. In addition, the entire surface of a waste package wetted by seepage water dripping onto the package was assumed to be wetted, and all seepage water contacting the package was assumed to enter the package wall penetration(s) when they occur. The TSPA-VA support analyses (DOE98a) recognized that only a small fraction of the waste package surface would be wetted (the total amount of water contacting the package each year is estimated to be on the order of 20 liters), and that only a fraction of the seepage water contacting the package would enter the wall penetration (e.g., because corrosion products would block entry). Because of uncertainties in placing values on the relevant performance parameters, these factors, which could greatly defer and diminish radionuclide release from the waste form, were omitted from the TSPA-VA evaluations and the bounding conservative assumptions were used.
- The TSPA-VA assumed that 0.1 percent of the Zircaloy-clad commercial spent fuel rods emplaced in the repository will be “failed” at the time of emplacement, that the spent fuel contents of each penetrated waste package will include 1.15 percent stainless-steel-clad fuel rods, all of which fail completely and immediately when the package wall is penetrated, and that all waste form area in failed fuel rods is exposed and contacted by water that enters the package. Overall, therefore, 1.25 percent of the waste form area in a failed package was assumed to be exposed and wetted. In the context of the TSPA-VA evaluations this was considered by some (i.e., NRC staff and the NWTRB) to constitute cladding “credit” because only a small fraction of the waste form in a failed package was assumed to be exposed and wetted. The TSPA-VA assumptions

may in fact greatly overstate the extent of exposed waste form area. An extensive data base shows that “failures” of spent fuel cladding are predominantly hairline cracks which would expose only a small waste form area. In addition, the Zircaloy cladding is not susceptible to significant degradation after disposal, and there are only about 2,100 stainless-steel-clad subassemblies, which could be packaged together in less than 100 of the 10,000 waste packages. These segregated packages could be made more failure resistant by using some of the design options assessed in the VA, such as drip shields. With a greatly prolonged waste package lifetime the level of assumed cladding “failure” at emplacement would be lowered by an order of magnitude with consequent lowering of the dose to the receptor. In summary, if only the penetrations of Zircaloy cladding that exist at emplacement allow water to contact the waste form, and if extreme assumptions concerning stainless-steel-clad spent fuel are avoided, the DOE assumptions could overstate the waste form area available for radionuclide release by as much as three orders of magnitude.

7.3.4 Reviews of the TSPA-VA

Formal reviews of the DOE Viability Assessment and the TSPA-VA were documented by the Nuclear Regulatory Commission, the TSPA-VA Peer Review Panel, and the Nuclear Waste Technical Review Board. Their comments are summarized below.

7.3.4.1 NRC Review of the TSPA-VA

In a March 1999 letter to the NRC Commissioners, the NRC Staff provided comments on the TSPA-VA (NRC99c). In addition, the NRC provided some informal feedback to DOE during the May 25-27, 1999 DOE/NRC Technical Exchange (NRC99b). The NRC's feedback was based primarily on a comparison of the TSPA-VA with NRC's TPA 3.2 performance assessment. Details of TPA 3.2 are presented in Section 7.3.5. As discussed in that section, there are substantive differences in the models and parameters used by the two agencies. The purpose of this section is not describe the differences between the TSPA-VA and TPA 3.2 but rather to summarize some of the key NRC comments on the TSPA-VA.

The NRC Staff review covered: (1) the preliminary design concept for the critical elements of the repository and the waste packages; (2) the TSPA based on this design concept and data available as of June 1998; and (3) the license application (LA) plan. The Staff did not review the DOE cost estimates to construct and operate the Yucca Mountain repository. The review focused on those issues that needed to be addressed before the LA is issued (scheduled for 2002) to insure that the application will be complete and minimize the need for a protracted license

review. The NRC agreed with DOE's position that work should proceed toward a decision on recommending the Yucca Mountain site as a repository for high-level waste.

There were a number of areas where the NRC Staff did not have major comments at the time of its review based on general agreement with DOE on the particular issues. These included: mechanical disruption of the waste packages; radionuclide release rates and solubility limits; spatial and temporal distribution of flow in the unsaturated zone (UZ); distribution of mass flux between fractures and matrix in the unsaturated zone; retardation in the UZ fractures; retardation in the water-production zones and alluvium; dilution of radionuclides in the ground water from well pumping; airborne transport of radionuclides; dilution of radionuclides in the soil; and location and lifestyle of the critical group. This is not to say that these processes are insignificant; rather, there were no significant issues in these areas at the time of the reviews.

Areas where the Staff had significant comments included:

- Repository design
- Waste package corrosion
- Quantity and chemistry of water contacting waste packages and waste forms
- Saturated zone flow and transport
- Volcanic disruption of the waste packages
- Quality assurance

With regard to repository design, NRC expressed concern as to whether adequate time was available before the LA is scheduled for submittal to address all the design options under consideration, select a reference design, develop data and models, and conduct the analyses required to produce an LA which is complete and of high quality.

Doses received by down gradient receptors are highly sensitive to the corrosion performance of the waste packages. The DOE is exploring several alternatives to the waste package design used in the TSPA-VA, which was a 10-cm outer layer of carbon steel corrosion allowance material and a 2-cm inner layer of Alloy 22 corrosion resistant material. It was not clear to the NRC that the DOE would be able to gather adequate long-term corrosion data in time to definitively support the LA. The TSPA-VA relied heavily on expert elicitation rather than long-term test data and this is a significant weakness.

The amount and chemistry of the water which contacts the waste packages is of critical importance not only to waste package lifetime but also to release of radionuclides once the waste package is breached. The NRC concluded that "...the range of activities outlined in the LA Plan

are unlikely to provide an adequate licensing basis for assessing the quantity and chemistry of water contacting waste packages and waste forms. Additional data and analysis of seepage under both isothermal and thermal conditions will be required for a complete LA.”

The NRC was not satisfied that flow and transport in the saturated zone from beneath the repository to a receptor 20 km down gradient had been adequately characterized. Additionally, the NRC did not concur with the DOE’s view that saturated zone uncertainties were a “moderate” contributor to receptor dose uncertainties. This descriptor was inappropriately optimistic based on sensitivity studies conducted by both organizations. The Staff expressed concerns that the location where ground water enters the alluvium (which delays radionuclide migration) was not well documented. High permeability features between the repository and the receptor could alter the flow direction away from the alluvium and confine the flow to the fractured tuffs.

Based on Staff review, the NRC concluded that the consequences of volcanism were understated in the TSPA-VA. The DOE assumptions on physical conditions were not representative of basaltic volcanism at Yucca Mountain. In addition, the DOE’s models did not consider the impact of the dynamic forces produced by the volcanism on waste packages in a volcanic conduit.

Implementation of an appropriate Quality Assurance (QA) program has been an on-going problem. The NRC has reviewed and accepted the DOE’s QA program on procedural basis. However, audits and surveillances have identified deficiencies in implementing the program. Some data in the technical data bases are not traceable. The NRC is concerned that the LA Plan did not recognize these implementation deficiencies and provide for remedies.

The NRC staff provided some additional reactions to the TSPA-VA in the May 1999 Technical Exchange (NRC99b). The TSPA-VA documentation included several features which facilitated the NRC’s understanding of the DOE performance assessment. These included extensive use of plots of intermediate outputs such as time-dependent Tc-99 release from a waste package. Plots of the performance of sub-systems such as the number of waste packages which failed as a function of time were also valuable as were dose rate plots which showed the mean, median, 5th, and 95th percentiles over time. The DOE’s presentation of the results of sensitivity analyses and the dose rates expected with alternative conceptual models also enhanced the NRC’s understanding of the TSPA-VA. On the other hand, the NRC felt that there were areas where transparency and traceability could be improved. The NRC staff noted that the flow of key information between the RIP computer code and external process models was difficult to trace.

The NRC also concluded that there was inadequate sampling of parameters potentially important to repository performance and they could not determine whether correlations between sampled parameter had been properly addressed. The Staff suggested that a table listing all important parameters and their assigned distributions would significantly facilitate review.

The NRC felt that both agencies needed to have a better technical basis for establishing the initial waste package failure levels. Improved linkage was required between initial defects and waste package failure rates. This would involve consideration of the detectability of initial defects and consideration of the expected performance of the defective waste packages. Further, with regard to long-lived waste packages, the NRC averred that there were potential failure processes such as stress corrosion, microbial activity and exposure to alternating wet/dry cycles which could accelerate failure. These processes were not considered by either organization.

The NRC concluded that there were no major performance-affecting differences in the approaches taken by the two organizations with regard to ground water infiltration and deep percolation. However, the modeling approaches taken for unsaturated zone flow and transport differed markedly.

In past near-field modeling the DOE did not consider that penetration of the boiling isotherm in the drift wall could occur by water flowing down a fracture. The NRC concluded that the DOE's assumption that water will not contact a waste package until the waste temperature drops below the boiling point was not conservative. In the TSPA-VA, the drift seepage model was based on ambient conditions and was not coupled to a thermal model. DOE assumed the first waste package fails after 1,000 years and is under a drip.

The NRC observed that the TSPA-VA methods for calculating biosphere dose conversion factors (DCF) were consistent with the NRC approach, but the Commission raised some questions as to whether the procedures used for sampling the DCF distributions created modeling inconsistencies. The NRC also felt that the documentation on dose parameters used in the TSPA-VA needed to be improved.

The NRC concluded that the model for igneous activity used in the TSPA-VA was inadequate. Additional work would be required to develop acceptable models. However, based on discussions between the DOE and the NRC subsequent to publication of the TSPA-VA, the NRC was of the opinion that acceptable modeling approaches can be developed before the License Application is submitted.

7.3.4.2 Review by the TSPA Peer Review Panel

DOE created the TSPA-VA Peer Review Panel to provide the Civilian Radioactive Waste System Management and Operating Contractor with a formal, independent review and critique of the TSPA-VA (PRP99). In its review of the Viability Assessment, the Panel was charged with considering both the analytical approach used and its traceability and transparency in assessing the probable behavior of the repository. Factors evaluated in assessing the analytical approach included:

- Physical events and processes included in the assessment
- Use of appropriate and relevant data
- Assumptions made
- Abstraction of process models used in total system models
- Application of accepted analytical methods
- Treatment of uncertainties

The Panel concluded that, due to the complexity of the system and the nature of the current or reasonably obtainable data, it may be impossible for any technical team to develop the analytical capabilities to prepare a credible assessment of the probable future behavior of the repository. The long time scales which must be considered, coupled with the complexity of the geologic setting, compound the analytical problems. The Panel suggested that dealing with these complex coupled processes can best be handled through bounding analyses or by incorporation of engineered features which minimize the effects of these processes.

In the Panel's words, a credible assessment "*would have needed to include:*

- *Component subsystem models that capture important and relevant phenomena;*
- *Adequate databases;*
- *Proper coupling between the subsystem models; and*
- *Tests of modeled behavior*".

Although the TSPA-VA offers many examples of partial, even substantial, success in each of these four areas, the Panel has also observed examples of important deficiencies in each.

- Concerning subsystem models, the final dose estimates within the TSPA-VA rest in large part on potentially optimistic, or at least undemonstrated, assumptions about the behavior of certain barriers in the system (for example, performance of the cladding and the waste package).

- Concerning databases, some of the important analyses are not supported by an adequate database, (for example, databases for corrosion of spent fuel alteration products and the saturated zone analysis).
- Concerning coupled processes (that is, thermohydrological, thermomechanical, and thermochemical effects) and the data and models that support them, the Panel believes that it may be beyond the capabilities of current analytical methodologies to analyze systems of such scale and complexity. For this reason, the effects of coupled processes can probably best be dealt with through a combination of bounding analyses and engineered features designed to minimize the effects of such processes.
- Concerning tests of modeled behavior, the TSPA-VA does not contain the convincing direct measurements or confirmation of the modeled behavior of components or subsystems for which testing is feasible. This testing should be part of the analyses of such a complicated system.”

The Panel concluded that the sensitivity analyses in the TSPA-VA did not provide sufficient insights to overcome these deficiencies and uncertainties.

The Panel expressed concern over the lack of data relating to the performance of the waste packages and reliance on instead on expert elicitation. The Panel stated that DOE must define the environmental extremes to which the Alloy C-22 corrosion resistant liner will be exposed and establish experimentally the critical temperature for crevice corrosion in these aggressive environments. The need to obtain more and better data to enhance performance assessment credibility was a repeated theme throughout the Panel’s report.

The behavior of the waste packages is strongly dependent on the extent to which contact with infiltration water seeping into the drifts is minimized. The Panel was not convinced that the TSPA-VA base case correctly captured seepage into the drifts over long periods of time. The Panel concluded that “Better characterization of the hydrologic properties near the drifts, improved modeling, consideration of coupled effects, and additional experimentation at the drift scale would add confidence to the approach taken.”

The Panel reviewed the impacts of five potentially disruptive processes on the Yucca Mountain repository. The Panel concurred with DOE findings in the TSPA-VA that impacts of earthquakes would be minor as would the impacts of volcanism on offsite groups. The Panel also agreed with DOE's analysis that nuclear criticality was highly improbable and, if it occurred, only modest increases in offsite doses would be expected. However, the Panel was not satisfied with DOE's analysis of human intrusion. They stated that the scenario in which the

waste generated from an intruding borehole was driven downward into the SZ was not realistic and analytical treatment of transport within the saturated zone was potentially non-conservative. The particular concern with the transport model was the assumption that radioactive material was distributed over a wide area at the top of the SZ. This would not be the case with the selected drilling intrusion scenario. The Panel noted that a regulatory basis for analyzing human intrusion had not been established by either NRC or EPA at the time when the TSPA-VA calculations were made. The approach taken on the climate change in the TSPA-VA was judged to be reasonable, in-so-far as temporal variations in precipitation are concerned. The Panel noted that the U.S. Geological Survey disputed the manner in which the variation in precipitation was translated into infiltration rates into the repository but the Panel took no position on that issue.

Two potentially non-conservative approaches used in the TSPA-VA were identified by the Peer Review Panel, namely:

- Long-term performance of Zircaloy cladding on spent fuel
- Buildup of radionuclides in soil irrigated with contaminated groundwater

With regard to cladding performance, the Panel stated that additional failure mechanisms including (1) pitting and crevice corrosion, (2) hydride-induced embrittlement and cracking, and (3) unzipping of the cladding due to secondary phase formation when the UO_2 fuel is converted to various alteration products in a moist, oxidizing environment all need to be experimentally investigated. Until such work is completed and the expected cladding longevity can be substantiated, the TSPA-VA assumptions about the ability of the cladding to act as a significant barrier are not defensible.

The Panel observed that irrigation water was assumed in the TSPA-VA to be deposited on the soil for only one year prior to intake by the receptor via various soil-related pathways. In reality irrigation can continue for thousands of years and an equilibrium concentration for each nuclide will be established in the soil which is higher than that based on only a one-year exposure period. In addition, the assumption that iodine is rapidly washed through a soil column is not supported by field observations which show considerable holdup in the surface layers.

The Panel also identified three factors which were believed to be treated with significant conservatism in the TSPA-VA including:

- Transport through penetrations in the waste package
- Retention of radionuclides in spent fuel alteration products
- Potential sorption of technetium and iodine in the UZ and SZ

The Panel felt that the modeling of the transport of radionuclides from failed waste packages through pits, cracks or crevices was not realistic since no significant retardation was included. Since this assumption is not consistent with expected physical reality, better methods are required to analyze the movement of radionuclides within and from the failed waste packages.

Any UO_2 in spent fuel packages which is exposed to moist air is expected to be converted to secondary uranium minerals such as schoepite within a few hundred years after waste package and cladding failure. It has been experimentally established that neptunium would be incorporated into the alteration products and, consequently, Np release would be controlled by the dissolution rate of these alteration products. While this process was not included in the TSPA-VA base case, it was cursorily examined in a sensitivity analysis (DOE98, Volume 3, Section 5.5.3). No impact was shown over the first 10,000 years or after about 700,000 years because releases are dominated by other nuclides for those time periods. However, at 100,000 years, the dose rate is reduced by about a factor of 10 when solubility of Np from the alteration products is considered.

No sorption of technicium or iodine (the major contributors to dose over the first 10,000 years) on geologic materials was considered in the TSPA-VA. However, the Peer Review Panel cited field observations, such as those of Straume et al. (STR96), taken near the site of the Chernobyl nuclear power plant accident suggesting that radioiodine may be retarded in soil surface layers. The Panel did not cite any instances where technetium was retarded but suggested that the issue should be reviewed on the basis, for example, of measurements near the Chernobyl site.

In addition to these general conclusions, the Panel provided detailed comments on all of the component models used in the TSPA-VA including the UZ flow, thermohydrology, near-field geochemical environment, waste package degradation, fuel cladding as a barrier, waste form degradation, radionuclide mobilization, UZ transport, SZ flow and transport, biosphere, and disruptive events. Recurring themes were the need for additional data and improved models to produce a credible and defensible LA.

7.3.4.3 Review by the U.S. Nuclear Waste Technical Review Board

The Nuclear Waste Technical Review Board (NWTRB; see Section 4.4 of this BID) also critiqued the TSPA-VA (TRB99). The Board stated that they had identified no features or processes which would disqualify the Yucca Mountain site but felt that DOE should give serious attention to replacing the high-temperature design evaluated in the TSPA-VA with a ventilated low-temperature design where waste package surface temperatures were maintained below the

boiling point of water. Such a change should significantly reduce the uncertainties involved in attempting to analyze complex coupled thermal-hydraulic and thermal-mechanical, and thermal-geochemical interactions within the repository.

The NWTRB also expressed concerns as to whether the amount of work required to support a technically defensible decision on Yucca Mountain could be completed on DOE's proposed schedule, which calls for a site recommendation decision by 2001. This is a matter of particular concern, since the Board stated that expert elicitation should not be used as substitute for data gathering at the site or in the laboratory. Areas where additional factual input is required include waste package performance (e.g., resistance to stress-corrosion cracking), and the magnitude and distribution of seepage into the repository.

The Board also stressed the need for long-term scientific studies assuming the site is ultimately found to be suitable and construction is approved. These scientific studies should include selected aspects of both natural and engineered barriers.

In summary, the Board agreed with DOE "that Yucca Mountain continues to merit study as the candidate site for a permanent geologic repository and that work should proceed to support a decision on whether to recommend the site to the President for development. ... The Board supports continuing focused studies of both natural and engineered barriers at Yucca Mountain to attain a defense-in-depth repository design and to increase confidence in predictions of repository performance."

7.3.5 NRC Total System Performance Assessments

7.3.5.1 Background

To support its licensing responsibilities, the NRC is developing the capability to review DOE's TSPA in support of a License Application, if the Yucca Mountain site is found to be suitable for disposal. The Commission staff, like DOE, is iteratively developing TSPA modeling capability based on evolving information and insights concerning factors that affect repository system performance. Development of the TSPA methodology is independent of DOE's effort, and the DOE and NRC TSPA models and codes differ in detail.

The NRC's strategic planning calls for early identification and resolution, at the staff level, of TSPA issues before receipt of an LA, if the Yucca Mountain site is found to be suitable for disposal. The principal means for achieving this goal is on-going, informal, pre-licensing consultation in which performance issues are identified and discussed, and issue resolution is sought. Resolution of issues is sought at the staff level before formal licensing reviews, but issues may be raised and considered again in the licensing process.

To implement its goals, the NRC has focused its pre-licensing work on issues most critical to the post-closure performance of the proposed repository; these have been designated as Key Technical Issues (KTI). To facilitate dialog with DOE concerning resolution of the KTIs, the NRC has established Issue Resolution Status Reports (IRSR) to serve as the primary mechanism through which feedback to DOE concerning KTIs and KTI subissues will be expressed and documented. The IRSRs address acceptance criteria for issue resolution and the status of resolution. Updating revisions of the IRSRs will be issued periodically as progress is made in resolution of the KTIs and their subissues.

One of the Key Technical Issues identified and discussed in an IRSR is Total System Performance Assessment and Integration (TSPAI). The NRC has, to date, issued the original version of the IRSR on this topic in April 1998 and Revision 1 in November 1998 (NRC98). As basis for its review of the DOE TSPA and development of its own TSPA methodology, the staff has adopted the hierarchical structure of performance assessment factors shown in Figure 7-41. This performance factor structure was used to develop the NRC TSPA code structure (e.g., TSP 3.x.y) illustrated in Figure 7-42. This code structure can be compared to DOE's TSPA-VA code structure shown in Figure 7-36.

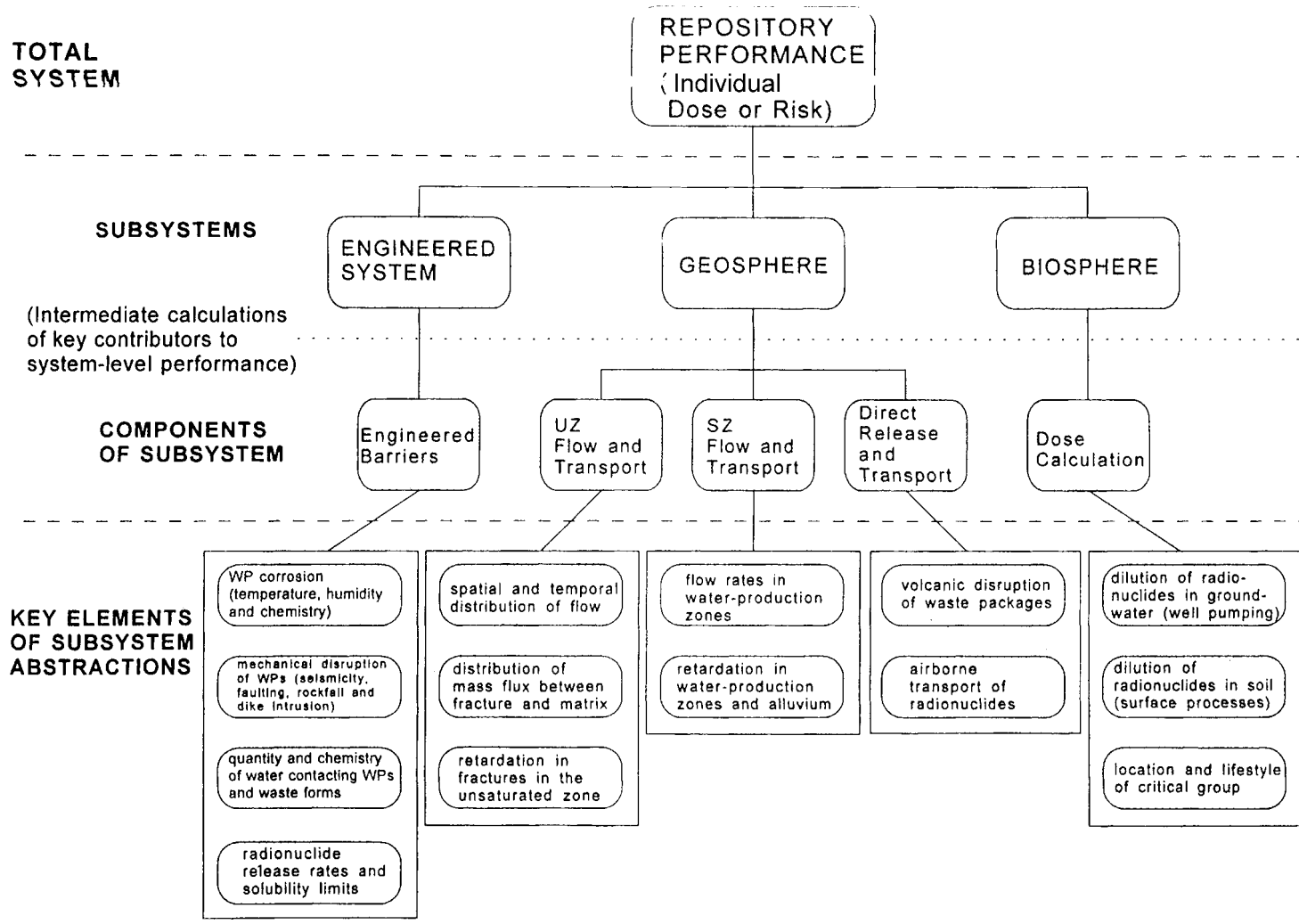


Figure 7-41. Structure of Performance Factors for NRC Performance Assessments (NRC98)

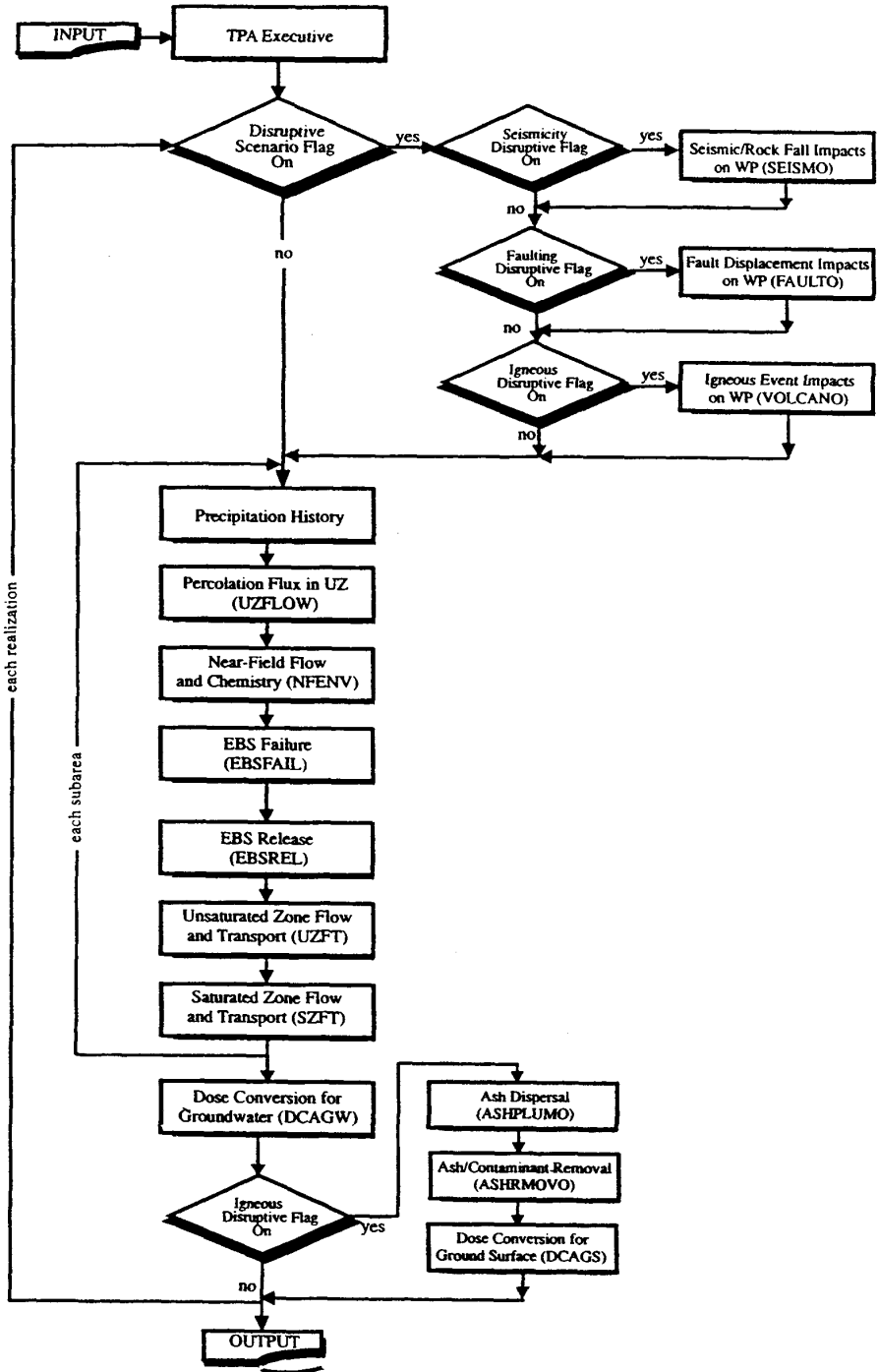


Figure 7-42. Structure of NRC Computer Codes for Performance Assessments (NRC98)

The IRSR on total system performance assessment and integration identifies and describes the key subissues for this topic as follows:

- “Demonstration of the Overall Performance Objective. This subissue focuses on the role of the performance assessment to demonstrate that the overall performance objectives have been met with reasonable assurance. This subissue includes issues related to the calculation of the expected annual dose to the average member of the critical group and the consideration of parameter uncertainty, alternate conceptual models, and the results of scenario analysis.
- Demonstration of Multiple Barriers. This subissue focuses on the demonstration of multiple barriers and includes: (1) identification of design features of the engineered barrier system and natural features of the geologic setting that are considered barriers important to waste isolation; (2) description of the capability of barriers to isolate waste; and (3) identification of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect the performance of natural barriers.
- Model Abstraction. This subissue focuses on the information and technical needs related to the development of abstracted models for TSPA. Specifically, the following aspects of model abstraction are addressed under this subissue: (i) data used in development of conceptual approaches or process-level models that are the basis for abstraction in a TSPA, (ii) resulting abstracted models used to perform the TSPA, and (iii) overall performance of the repository system as estimated in the TSPA. In particular, this subissue addresses the need to incorporate numerous features, events, and processes into the performance assessment and the integration of those factors to ensure a comprehensive analysis of the total system.
- Scenario Analysis. This subissue considers the process of identifying possible processes and events that could affect repository performance; assigning probabilities to categories of events and processes; and the exclusion of processes and events from the performance assessment. This is a key factor in assuring the completeness of a TSPA.
- Transparency and Traceability of the Analysis. This subissue emphasizes staff expectation of the contents of DOE’s TSPA to support an LA. Specifically, it

focuses on those aspects of the TSPA that will allow for an independent analysis of the results.”

Details of acceptance criteria and review methods for the subissues related to demonstration of overall performance and demonstration of multiple barriers will be provided in the next revision of the IRSR for TSPAI. Details of criteria and review methods for model abstraction, scenario analysis, and transparency are included in NRC98.

7.3.5.2 NRC Development and Use of TSPA Models

The content and characteristics of NRC’s TSPA models have, like DOE’s, evolved over time as information and insights as basis for the models have developed. Current models, also like DOE’s, are considered to be a snapshot in time from an on-going model-development process.

Under its Iterative Performance Assessment (IPA) program, NRC has adopted a phased approach to its TSPA modeling capability. Phase 1 used relatively simplistic models and was designed primarily to demonstrate capability to perform TSPA reviews as part of the licensing reviews. Phase 2 used significantly enhanced modeling methods to identify and assess factors of primary importance to repository system performance. Phase 3, which is still underway, uses more general and versatile computer codes to perform TSPA evaluations analogous to those performed by DOE.

Three versions of the Total-system Performance Assessment (TPA) code have been developed in Phase 3 of the IPA program. TPA 3.1.3 has been used to calculate mean doses for alternate conceptual models, and TPA 3.1.4 has been used for system-level sensitivity and uncertainty studies. The most recent version of the TPA code, 3.2, was used to provide feedback to DOE on the results of NRC’s review of the TSPA (see Sections 7.3.2. and 7.3.3).

The most recently documented description of the NRC TPA codes is provided in NUREG 1668, which describes the characteristics and use of the 3.1.3 and 3.1.4 codes to perform sensitivity and uncertainty analyses for a proposed repository at Yucca Mountain (NRC99a). Characteristics of the TPA 3.2 code have not yet been documented, but results of its use were presented and discussed at the May 1999 DOE/NRC Technical Exchange (NRC99b) in which NRC staff provided feedback to DOE concerning results of their review of the TSPA-VA.

The TPA 3.1 and 3.2 codes have capability and flexibility comparable to those of the DOE codes for the TSPA-VA. As previously noted, the DOE and NRC codes differ significantly in detail,

but both have capability to evaluate performance for alternative repository design features, natural system features, and disruptive scenarios, at a level of detail and characterization of uncertainty commensurate with the available information base. At present, the principal difference between the NRC and DOE performance assessment codes is that the NRC codes give considerable attention to disruptive events associated with seismicity and volcanism, while the DOE approach considers these phenomena to be unlikely to occur in ways that could affect repository performance. These differences are expected to be resolved as part of the issue resolution process.

Principal features of the NRC's Phase 3 performance assessment codes include the following:

- Water infiltration into the subsurface. Calculation of percolation flux takes into account the time history of climate change, variation of shallow infiltration with climate change, and the areal-average percolation flux at the repository horizon.
- Near-field environment. The near-field environment, which affects the waste package corrosion rate, is characterized in terms of drift wall and waste package surface temperatures, relative humidity, water chemistry, and water reflux during the thermal pulse phase.
- Waste package degradation and EBS release. Waste package failures depend on near-field conditions, corrosion mechanisms and rates, and mechanical effects such as rockfall. Radionuclide release from the EBS is calculated in terms of rate of release from the waste form, solubility limits, and transport mechanisms out of the EBS. No credit is taken in the base case for cladding performance as a barrier.
- Transport in the UZ and SZ. Time-dependent flow velocities in the UZ are calculated using the hydrologic properties of the major hydrostratigraphic units. Matrix and fracture flow are modeled. Radionuclide retardation on fracture surfaces is assumed not to occur, but sorption in the rock matrix is modeled. The conceptual hydrologic model for flow in the SZ assumes fracture flow in the tuff aquifer and matrix flow in the alluvial aquifer.
- Airborne transport for direct releases. NRC performance assessments include consideration of airborne releases from low-probability intrusive igneous events which cause direct release of waste package materials into the air. Factors considered include number of packages failed and quantities of radionuclides released, ash deposition patterns, and degradation of deposited, contaminated ash.
- Biosphere dose exposure scenarios. Dose evaluations are done for the average person in a designated receptor group. Two types of groups are considered: a farming community 20 km downgradient from the repository, and a residential

community. The farming community is assumed to use contaminated ground water for drinking and agriculture; the residential community uses it only for drinking. Dilution of radionuclide concentrations in the ground water as a result of pumping is considered.

NUREG-1668 (NRC99a) reports the results of dose evaluations in which the base case TSP 3.1.4 model and 11 alternative conceptual models (such as including cladding credit) were used to calculate doses at 10,000 and 50,000 years for a receptor 20 km from the repository. The repository system conceptual design was similar to that used by the DOE in the TSPA-VA, but the corrosion-resistant inner package barrier was assumed to be Alloy 625. The annual base case mean peak total effective dose equivalent (TEDE) was projected to be 2.3 mrem at 10,000 years. Annual results for the alternative conceptual models ranged from a low of 0.012 mrem when cladding credit was taken to a high of 12.5 mrem when no radionuclide retardation was assumed. The range of results is shown as a bar chart in Figure 7-43.

As previously noted, the NRC presented its more recent TSP 3.2 results evaluations at the DOE/NRC Technical Exchange in May 1999 (NRC99b). Results presented for the ground water dose using the NRC's mean-values data set are shown in Figure 7-44, for 10,000 and 100,000-year dose rates. As can be seen, the 10,000-year dose rate is forecasted to be about 0.002 mrem/yr, and the 100,000-year dose is about 0.2 mrem/yr. These results can be compared to DOE's TSPA-VA results, which indicated a 10,000-year dose rate of 0.04 mrem/yr and a 100,000-year dose rate of about 5 mrem/yr (see Figures 7-37 and 7-38). Reasons for differences in the NRC and DOE results are not readily apparent because parameter values and modeling approaches used by the two agencies differed markedly. For example, the DOE assumed cladding credit while the NRC did not; the NRC assumed an average of 32 juvenile waste package failures while the DOE assumed one; the DOE used three-dimensional modeling of UZ below the repository which suggested significant lateral diversion while the NRC used one dimensional modeling with seven stream tubes and no lateral diversion. In addition, the NRC assumed dilution during pumping of contaminated ground water by the dose receptor, while DOE assumed this dilution did not occur.

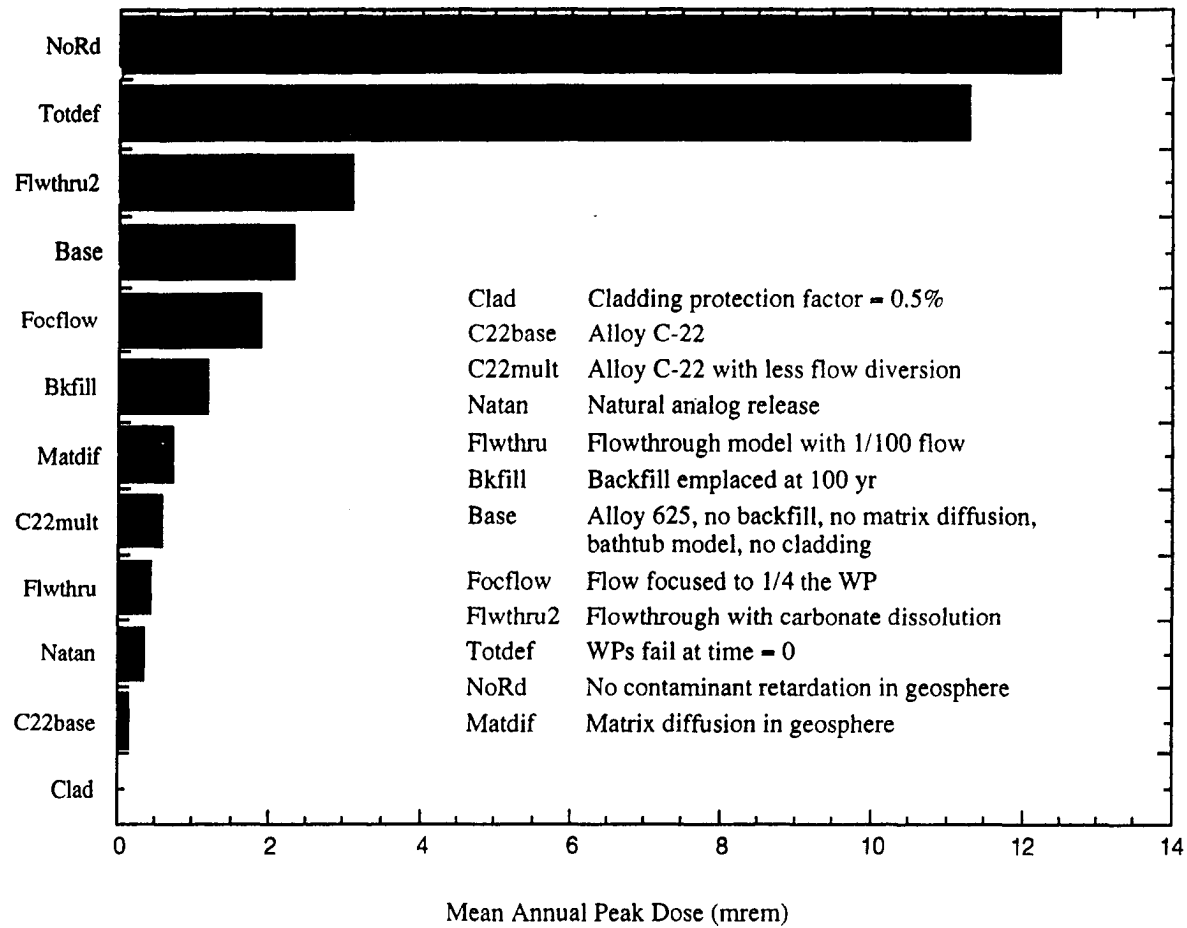


Figure 7-43 NRC TSPA Results for Alternative Conceptual Models (NRC99a)

7.3.5.3 Conservatism In The NRC Performance Assessments

As noted in Section 7.3.5.2, the NRC staff are independently developing performance assessment capability in order to be able to perform comprehensive reviews of DOE's TSPA in the License Application. The NRC performance assessment capabilities and methods are, like DOE's, continuing to evolve. Documentation of NRC's parameter values, models and assumptions are not yet as comprehensive as DOE's; the most recent description of the NRC models and the results of their use was provided in the NRC/DOE Technical Exchange of May 27-29, 1999 (NRC99b). As reported during the Exchange, NRC's base-case performance evaluations using VA design parameters projected a 10,000-year dose rate of about 0.003 mrem/yr; DOE's base-case 10,000-year dose rate projection was 0.04 mrem/yr. Conservatism in NRC's performance parameters, models, and assumptions, as indicated by information provided at the Technical Exchange, are summarized below.

Performance Parameters

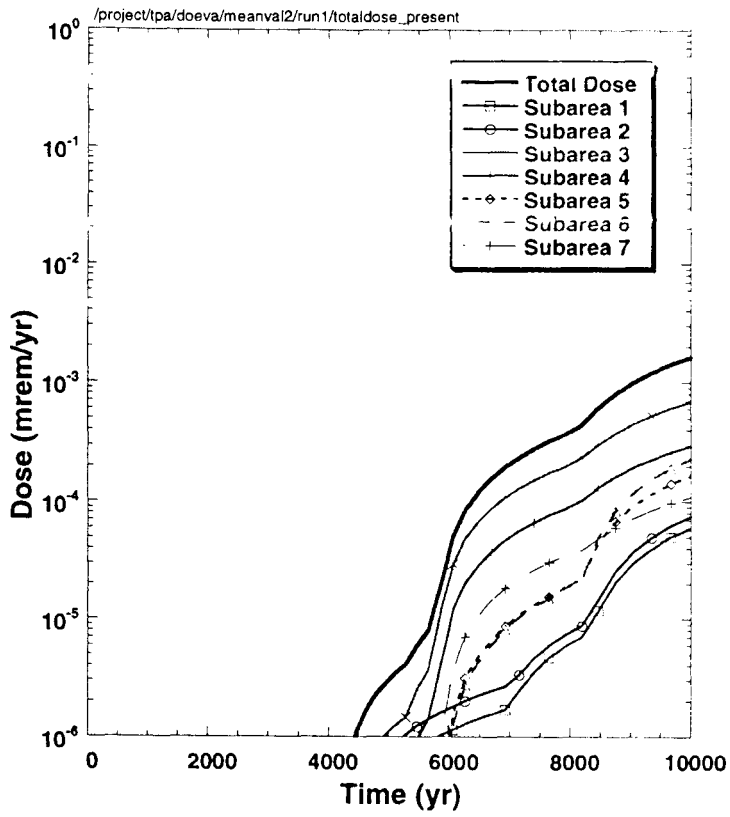
NRC presentations at the May 1999 Technical Exchange indicated that "mean values" of the performance parameters were used for the base case performance assessments. Values of some of the parameters were presented, but comparisons with DOE are difficult because of differences in modeling approaches and parameters used. In general, NRC's use of "mean values" appears to correspond in concept to DOE's use of "expected values." Values of parameters used by NRC for precipitation and infiltration were, for example, similar to those used by DOE.

Performance Models

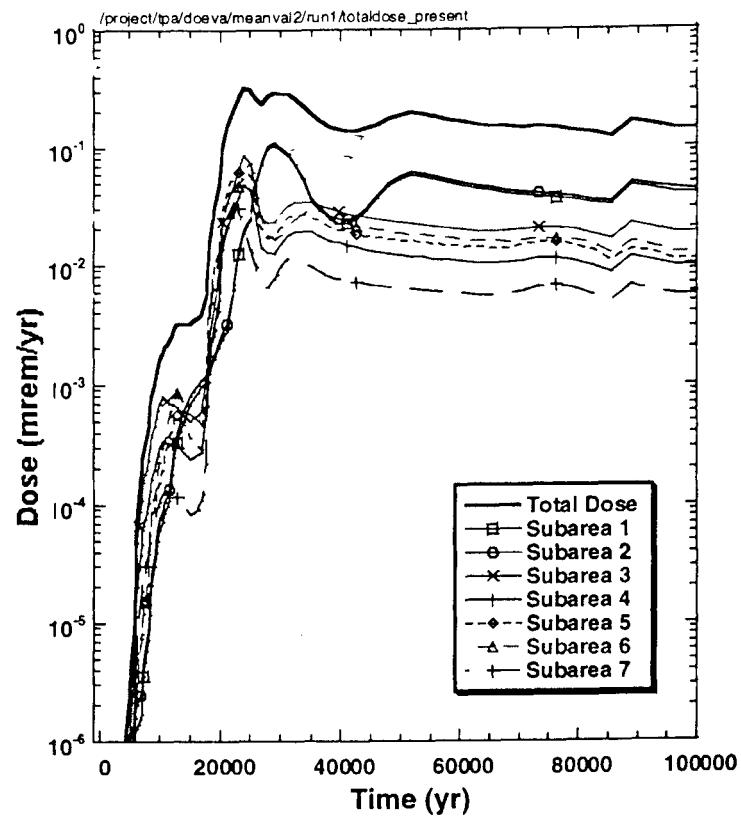
Key features of NRC's performance assessment modeling approach that are indicative of conservatism include the following:

- Impacts of igneous events, seismic rock falls, and fault displacements on waste packages were included in the models. Seismicity impacts were included in the base case evaluations; volcanism and faulting impacts were treated separately.
- No credit was taken for spent fuel cladding as an engineered barrier. Half of the spent fuel in a failed waste package was assumed to be exposed, wetted, and a source for release of radionuclides.

Outputs Using Mean-values Data Set



10 kyr



100 kyr

Figure 7-44. NRC TSPA Results for Mean-Values Data Set (NRC00b)

- Transport of radionuclides in the unsaturated zone from the repository to the water table was assumed to occur vertically, with no effect of matrix diffusion or sorption on fracture surfaces. This assumption is similar to that made by DOE in the TSPA-VA.
- Radionuclide transport in the saturated zone was assumed to occur via four pathways through fractured tuff and alluvium. Transport in the tuff occurred only via fractures, with flow rates between 50 and 500 m/yr. Flow velocities in the alluvium were assumed to be between 3 and 5 m/yr, and radionuclide retardation was assumed to occur.
- Dilution of radionuclide concentrations in ground water as a result of pumping by the dose receptor was assumed to occur (the dilution factor was not stated). This is a non-conservative modeling feature in contrast with DOE's assumption that such dilution does not occur.

Conservative Assumptions

Conservative assumptions in the NRC performance assessments described at the May 1999 Technical Exchange (NRC99b) included the following:

- Thirty-two waste packages were assumed to be defective at the time of emplacement. Rates and mechanisms of degradation and radionuclide release for these and other packages that fail were not described, however.
- The mean value of the localized corrosion rate for the Alloy 22 corrosion resistant material in the waste package was stated to be 2.5 E-4 m/yr. This is a factor of 100 higher than experimental values cited in EPRI's IMARC-4 report (EPR98) and in DOE's VA Technical Support Document (DOE98a).

Detailed comparison of NRC and DOE performance assessment conservatism is not possible because the modeling approaches and parameters used differ significantly. In general, it appears that, in comparison with DOE, NRC's approach produces a larger radionuclide source term (e.g., as a result of assuming no cladding credit), but compensates for it by assuming that dilution occurs during pumping. The net result is that the results of NRC's performance assessments reported at the May 1999 NRC/DOE Technical Exchange agree with DOE's TSPA-VA results within an order of magnitude.

7.3.6 EPRI Total System Performance Assessments

7.3.6.1 Background

The nuclear power utilities have for many years maintained oversight of the OCRWM program in DOE because of their contracts with the Department concerning its responsibilities for receipt and disposal of commercial spent fuel. Technical contributions to the oversight are provided by EPRI in programs that are selected and guided by the utilities. EPRI maintains peer capability to review and comment on DOE's program activities and to independently perform performance assessments and other analyses of the type done by the Department within the OCRWM program.

EPRI has performed independent total system performance assessments in parallel with DOE's efforts. A report on EPRI's TSPA concepts and methods was first issued in 1990 (EPR90), and TSPA reports were subsequently issued in 1992, 1996, and 1998 (EPR92, 96, 98). A more recent EPRI report (EPR00) represents more of a critique of the supporting documentation for the TSPA-SR than an independent assessment. The EPRI studies have kept pace with the DOE efforts, making use of the evolving repository design concepts, data bases, and modeling methods. The EPRI Phase 4 report, issued in November 1998 (EPR98) parallels the DOE's TSPA-VA report (DOE98) and uses the VA design. The EPRI Phase 5 report, issued in November 2000, provides a critique of the models and assumptions in the DOE's TSPA-SR.

The overall goal of the EPRI assessments is to provide an "...independent assessment of the performance of the potential repository site, identifying fatal flaws in the site itself, in the engineering design, or in the licensing program, so that the decision makers in the utility industry can judge the likelihood of potential outcomes of the licensing process and take appropriate action" (EPR96).

7.3.6.2 EPRI's TSPA Technical Approach

EPRI uses a logic tree approach to performance assessment modeling. The EPRI TSPA code is termed the Integrated Multiple Assumptions and Release Calculations code (IMARC). The logic tree approach, illustrated in Figure 7-45, represents uncertain inputs to the TSPA calculations as nodes in a tree, with branches from a node indicating alternative models or parameter values for that input and the weight associated with that model or parameter value. In contrast, the DOE TSPA code structure (Section 7.3.2.2) and the NRC approach (Section 7.3.4) use a central processor (e.g., the RIP code for DOE), which is fed information from codes for the various repository performance factors.

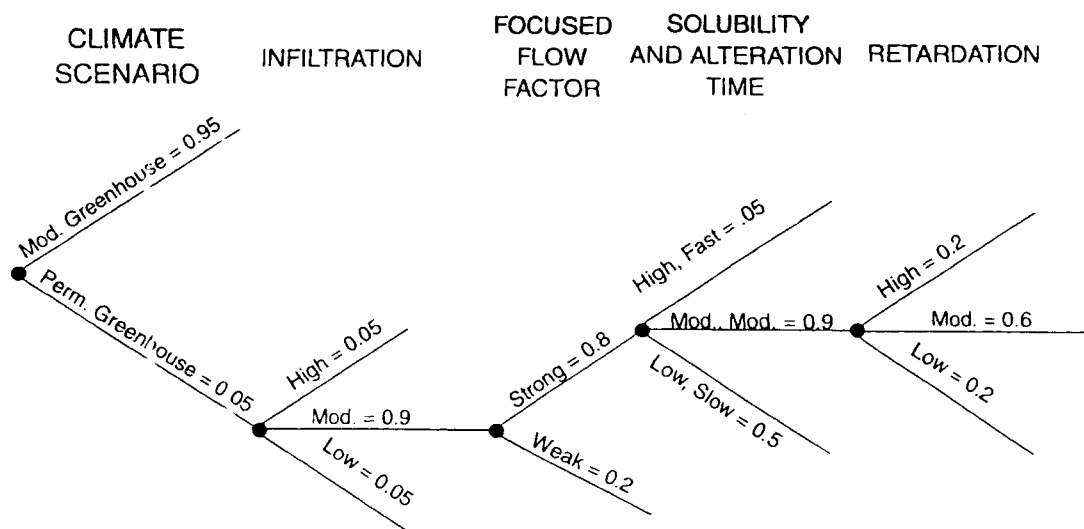


Figure 7-45. EPRI's IMARC Logic Tree (EPR98)

All TSPA methods include models for essentially the same performance factors, e.g., climate, infiltration, waste package performance, etc. They differ, however, in the details of how they model the performance factors and in their assignment of values for uncertain performance parameters. For example, the DOE assumed three climate conditions for the TSPA-VA, with precipitation spanning the range 170 to 540 mm/yr; in contrast, the EPRI interpreted the historic climate data to indicate two future climate conditions, with precipitation spanning the range 150 to 220 mm/yr.

Other key features of the EPRI Phase 4 TSPA modeling approach are outlined below. As for DOE, details of models and parameter characterization have evolved in accord with evolution of the data bases for performance assessment. Because the modeling approaches used in IMARC-4 were similar to those used in IMARC-3, the IMARC-4 report (EPR98) did not repeat technical details of modeling that were discussed in EPR96.

Climate

As indicated above, EPRI's interpretation of available data concerning past and possible future climate conditions led to an estimate that the long-term average precipitation should be between 150 and 220 mm/yr, a much narrower range than used by DOE in the TSPA-VA. EPRI believes

the DOE precipitation values are based on ostracode species assemblages found in Minnesota and Washington, rather than on specific plant taxa calibrated near Yucca Mountain.

Infiltration

The basic IMARC net infiltration model is a one-dimensional finite difference code that incorporates source and sink terms for surface infiltration, uptake of water by plants, and drainage from the root zone to the deep subsurface (which is net infiltration). For Phase 4, the runoff features of the model were revised as a result of recent data. As a result, the net infiltration for current climate conditions increased from the Phase 3 (1996) value of 1.2 mm/yr to 7.2 mm/yr. The full glacial climate value increased from 2.9 to 19.6 mm/yr. (DOE's TSPA-VA values showed similar increases in comparison with TSPA-95 values.) The TSPA-VA results are higher than the Phase 4 results because the DOE assumed a precipitation rate of 300 mm/yr as compared to EPRI's assumption of 195 mm/yr for a full glacial climate.

Near Field Conditions

For IMARC-4, EPRI developed a model and analytic solution which describes heat transfer and fluid flow in the near field in terms of a uniform disk-shaped heat source located in a moist, unsaturated, porous medium. Large-scale convective gas flow and countercurrent flow of water and vapor were assumed to occur. Heterogeneity of the repository's geohydrologic regime was represented by what was termed "focused flow, and "hot" and "cool" zones of the repository were characterized. The objective of the modeling was to estimate that fraction of the waste package inventory that is wetted; results indicated that the maximum fraction of the waste packages that are wetted is 0.24. In contrast, DOE's expected values in the TSPA-VA for waste packages with seeps were about 0.5 during superpluvial conditions and about 0.33 during the extended periods associated with long-term average climate (DOE98, Volume 3, Figure 4-3).

Waste Package Performance

The waste package performance model used in IMARC-4 differed significantly from that used in IMARC-3 because of improved understanding of the repository environment and corrosion processes, and because the reference corrosion resistant material (CRM) was changed from Alloy 825 to Alloy 22. The basis for characterizing corrosion rates was changed from Weibull

distributions²⁵ to recently-obtained corrosion data and the results of DOE's expert elicitation on waste package performance. Corrosion rates were characterized for various environmental conditions, e.g., humid air or water dripping onto the package, and for various corrosion mechanisms, including crevice corrosion of the Alloy 22, which is anticipated to represent the mechanism for most-rapid penetration of the CRM. Results for the VA waste package design (see Section 7.2) show that, in the absence of drips onto the package, penetration would not occur for more than one million years. When drips do contact the packages, penetration by general corrosion is predicted not to occur for about 30,000 years. Under adverse conditions, the carbon steel outer wall could be penetrated in only 300 years, and the Alloy 22 inner wall could be penetrated by crevice corrosion, which is conservatively assumed to occur during the time period during which the waste package temperatures are greater than about 80°C. The EPRI estimates that "hot" waste packages would remain above the 80°C threshold for crevice corrosion for about 3,000 years. For "cold" waste packages this period would be reduced to about 200 years. The EPRI notes in IMARC-4, as did the DOE in the TSPA-VA, that the data base for estimating Alloy 22 corrosion rates is currently quite limited.

Source Term Parameters

Source term parameters discussed in IMARC-4 include radionuclide sorption, solubility, release from the waste form, and waste form alteration. Values for these parameters were changed in IMARC-4 in comparison with IMARC-3 because of recent data additions. The computer code COMPASS, Version 2.0, which is a compartment model for predicting radionuclide release rates from the engineered barrier system (EBS) into the near-field rock, was used in IMARC-4. The Compass 2.0 code models EBS features, such as waste form, canister corrosion products, backfill, and rock fractures, as compartments. It accounts for time-dependent cladding degradation, modes of water contact with the waste package, and modes of water transport through the waste package interior (overflow or through-flow).

Discussions of source term parameters in IMARC-4 addressed the following:

- New values of sorption coefficients for sorption of radionuclides on corrosion products (principally iron oxides) were presented for cases where recent data differ from results of a prior expert elicitation by more than a factor of five. Median values for the actinides are in the range 5-10 m³/kg; the median value for Np is 0.1 m³/kg.

²⁵ A Weibull distribution is a function used to describe the fraction of waste packages which have failed as a function of time based on mean container lifetime, threshold failure time and failure rate at the mean lifetime.

- Extensive discussion was presented on the validity of the two-orders-of-magnitude reduction in the solubility of Np in the TSPA-VA in comparison with TSPA-95. The EPRI analyses basically concurred with the action, which was based on re-assessment of prior data and additions to the data base for solubility values. The solubility of neptunium is important to prediction of doses after 10,000 years, when neptunium is the principal contributor to dose.
- Extensive discussion was provided concerning thin films surrounding spent fuel undergoing dissolution. The EPRI concluded that the TSPA-VA approach was a “sensible, but non-unique first step in attempting to derive more realistic radioelement solubility constraints from laboratory tests.” The EPRI recommended additional modeling and laboratory tests to establish lower, more realistic solubility constraints.

Flow and Transport in the Unsaturated and Saturated Zones

The flow and transport models used in IMARC-4 were the same as those used in Phase 3. Values used for parameters were revised, however, as a result of recent insights concerning conceptual modeling of the UZ and SZ and continuing integration of field and theoretical studies.

The IMARC-4 UZ hydrology model accounts for transient, variably-saturated flow and advective-dispersive transport in a coupled dual-porosity-dual-permeability regime, from the base of the repository to the water table. Radionuclide sorption can occur both in the fractures and in the rock matrix. In the SZ, the model takes into account three-dimensional advective-dispersive transport of the radionuclides during down-gradient migration. The SZ model can handle matrix diffusion, radionuclide sorption and daughter-product ingrowth.

The repository footprint can be divided into subregions, each of which constitutes the top of a UZ hydrologic column. Input variables such as infiltration rates can therefore be varied over the area of the repository. The model assumes that there is no lateral coupling between the columns and that the system is isothermal, so that no coupling to the energy equations is needed.

Once the radionuclides reach the water table, they can advect, disperse, sorb, diffuse into or out of the matrix, and decay within the three-dimensional SZ. Ground water flow in the SZ is assumed to be representative of long-term steady-state conditions. The bulk hydraulic conductivity of the fractured rock mass is assumed to be representative of an equivalent porous medium, which may be anisotropic.

IMARC-4 discusses the impact of recent determinations that the net infiltration rate is much higher than originally believed, and the discovery of bomb-pulse Cl-36 at repository depths, on conceptual modeling of the UZ. It also discusses the impact of current lack of data for the SZ on uncertainty in the flow paths and dilution factors for the SZ. It notes that IMARC-3 asserted that overall dilution for the SZ was about a factor of ten, and that this value is retained in IMARC-4 and corresponds to the base case value used by DOE in the TSPA-VA. It also discusses dilution for a small radionuclide plume, such as would result from a single package failure, and asserts that the dilution factor for this situation would be on the order of 100,000.

Biosphere

The EPRI's IMARC analyses use a probabilistic model to estimate radiation doses. The model has three basic parts: probabilistic modeling of releases from the repository, characterization of dose conversion factors for the biosphere pathways and the nuclides of interest, and characterization of the dose receptor. In IMARC-4, EPRI used a farming critical group and the water-only pathway for their base case. Other possible dose circumstances (e.g., all pathways) were also evaluated. The critical group was assumed to be located 5 km from the boundary of the repository, i.e., at the boundary for release to the accessible environment as defined by 40 CFR Part 191.

The hypothetical critical group was assumed to extract ground water from the point of highest contamination in the contaminant plume, and to use this contaminated water for all of their food and water needs for their entire lifetime. Dose conversion factors were based on ICRP definitions of dose established in 1991 and on IAEA recommendations for metabolism of the elements established in 1994.

7.3.6.3 Results of IMARC-4 Dose Evaluations

The EPRI's IMARC-4 analyses produced base case results for conditions and assumptions outlined above, and also produced results for a wide range of sensitivity analyses. The EPRI base case results are shown in Figure 7-46. These results were obtained assuming that 0.01 percent of the waste packages had failed at emplacement (i.e., one package) and that 0.1 percent failed soon after emplacement (i.e., 1,000 yr). These early failures may be caused by manufacturing defects, construction errors, or emplacement mishandling. The EPRI modeling assumes no corrosion failures during the initial 10,000 years while the DOE modeling assumes that 17 waste packages will fail by corrosion during this period. Thus, the EPRI assumption for

total waste package failures (juvenile plus corrosion) is 11 while the equivalent DOE assumption is 18.

The dose receptor was assumed to be an average member of a farming community located 5 km from the repository, and the doses are the result of exposure only via the ground water pathway. When all exposure pathways were included, the dose rate variations as a function of time were similar to those shown in Figure 7-46, but about a factor of ten higher. This indicates that, for the EPRI modeling approach for the critical group, the drinking water contribution to dose is minor in comparison with the agricultural and other pathways.

Comparison of Figure 7-46 with the results of the DOE TSPA-VA analyses, Figure 7-39, shows that the dose rates at various times are generally similar (e.g., DOE projects a dose rate at 10,000 years of 0.04 mrem/yr; EPRI projects 0.08 mrem/yr), and the sources of dose are similar, i.e., Tc-99 and I-129 are dominant in the near term and Np-237 is dominant in the long term. In the EPRI results, Figure 7-46, the decrease in dose rate over the interval 60,000 to 100,000 years is the result of depletion of the Tc-99 and I-129 inventories for release from the repository.

EPRI IMARC-4 results are compared to DOE's TSPA-VA results and NRC's TSP 3.2 results in Section 7.3.7.

7.3.6.4 Conservatism In The EPRI Performance Assessments

As indicated in Section 7.3.6.2, the EPRI approach to total system performance assessments differs markedly from those used by DOE and NRC. As a result, direct comparison of EPRI conservatism with that of DOE and NRC is neither possible nor appropriate. In general, the IMARC-4 report (EPR98) suggests that EPRI seeks to be as realistic as possible in all aspects of its assessment efforts. For example, EPR98 criticizes the DOE interpretation of data concerning past climates as being too conservative, observes that the assumption of an early package failure is arbitrary, and notes that the EPRI and TSPA-VA approaches to modeling of fracture /matrix interactions in the saturated zone differ markedly.

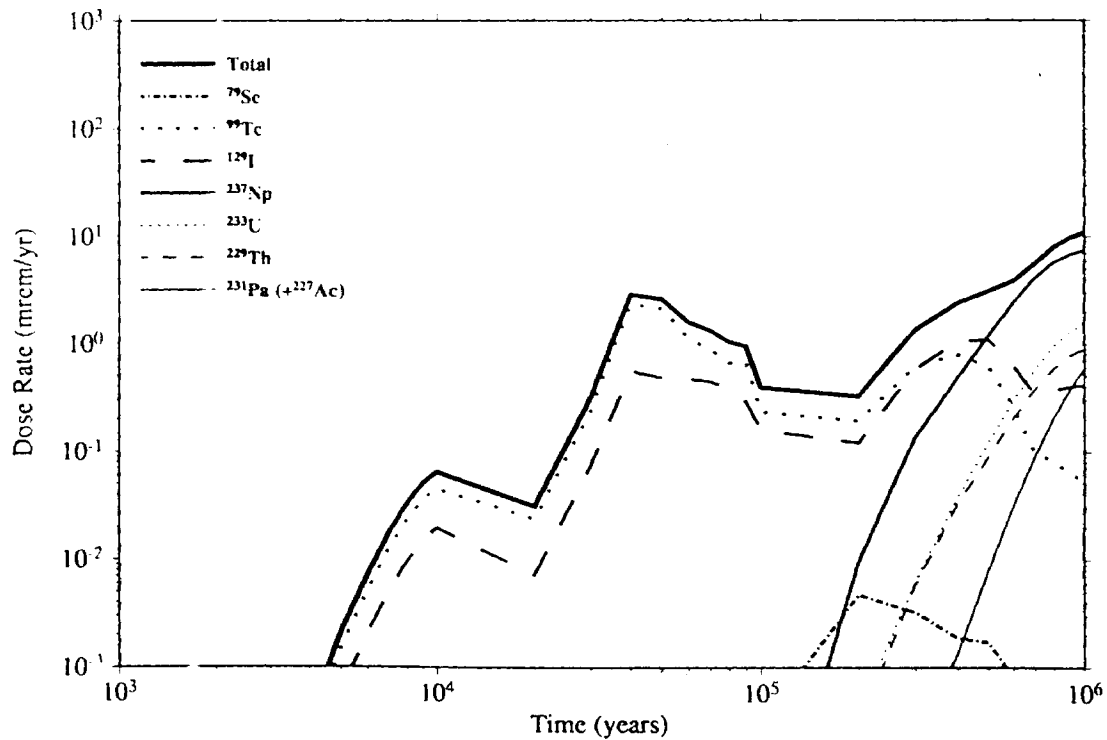


Figure 7-46. Results of EPRI's IMARC-4 Dose Evaluations (EPR98)

In contrast to DOE's adoption of expert opinion as the basis for waste package material corrosion rates, EPR98 includes a comprehensive effort to develop parametric models of corrosion behavior on the basis of available data. Like NRC, the EPRI IMARC-4 analyses take no credit for spent fuel cladding as a barrier. However, in contrast to NRC's bathtub model, EPRI uses a flow-through model for water entry to and exit from the interior of a failed waste package. This is similar in concept to DOE's approach, which assumed that radionuclides are instantaneously released to the EBS from the wetted waste form.

The IMARC-4 report, EPR98, includes a discussion which compares the IMARC-4 and TSPA-VA results. The report states:

"We observe that the magnitude of the doses estimated by IMARC Phase 4 are in general agreement with those in the TSPA-VA (within an order of magnitude for all time periods). This agreement can be considered quite close, given that the models, level of abstraction, and input parameters for particular FEPs [features, events, and processes] are considerably different between the two analyses. Whether this is simply fortuitous or speaks to the robustness of the combined analyses is not altogether clear. It may be that one particular combination of

conservatisms (and potential non-conservatisms) in one TSPA effort were, on the whole, balanced by a different combination of conservatisms/nonconservatisms in the other TSPA analysis. There is certainly some evidence for this.

In the end, this independent comparison of TSPA approaches for the proposed Yucca Mountain repository provides further confidence that the major FEPs controlling the overall safety of the facility have been identified.”

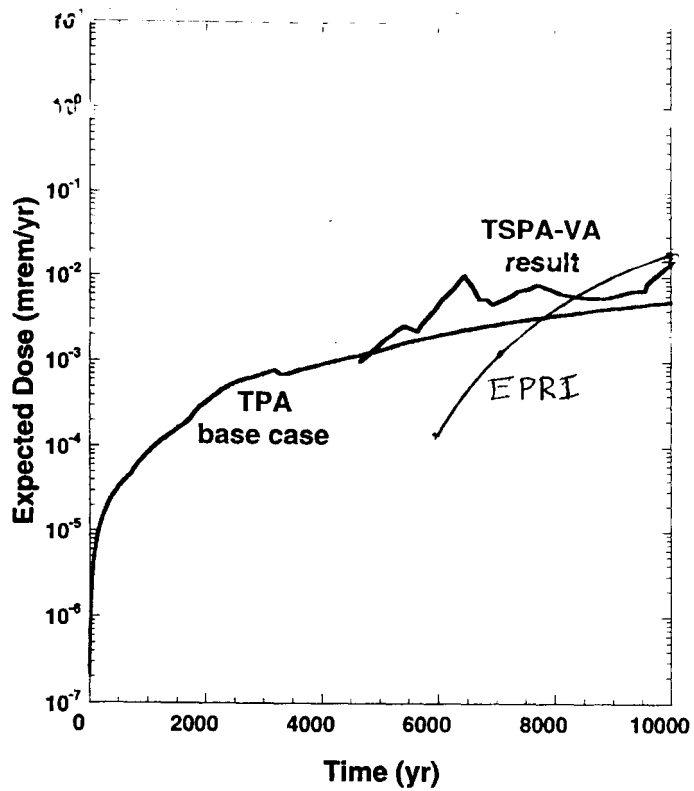
7.3.7 Comparison of DOE, NRC, and EPRI TSPA Results for the VA Repository

Although the TSPA models, assumptions, and parameter values used by DOE, NRC and EPRI differed greatly, each of the TSPA evaluations discussed above (DOE’s TSPA-VA, NRC’s TSP 3.2, and EPRI’s IMARC-4) has as its basis the VA repository design concept, key features of which are the waste package design (an outer wall of carbon steel and an inner wall of Alloy 22), and an areal heat loading of 85 MTU/acre. Despite widely different modeling concepts, and with only the principal design features of the repository and the existing data base as the basis for commonality of the analyses, the results of the three TSPA efforts are quite similar, as shown in Figure 7-47.

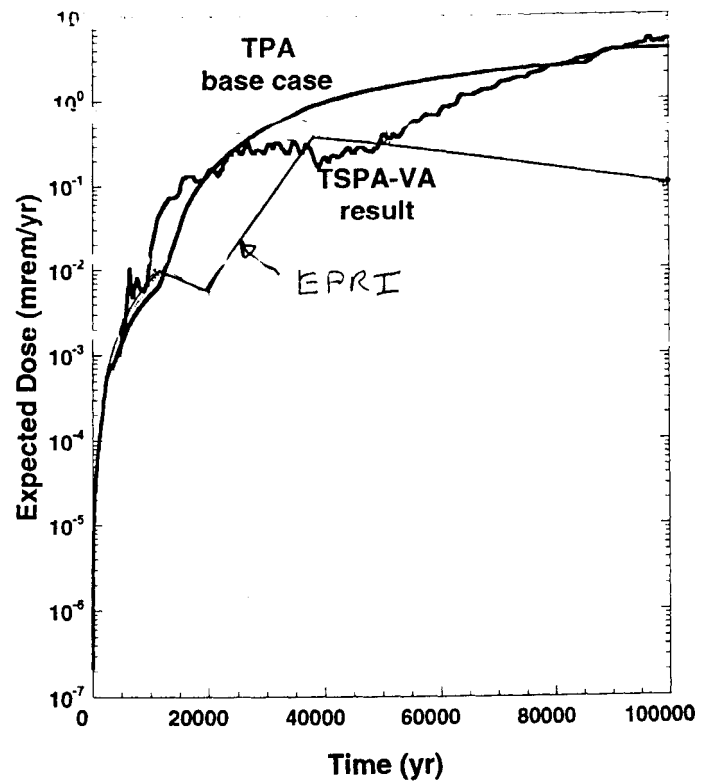
In Figure 7-47 the EPRI results are decreased by a factor of ten in comparison with the actual results because the EPRI dose receptor was assumed to be located only 5 km from the repository. This location, in comparison with the 20 km distance assumed by DOE and NRC, would not have achieved the SZ radionuclide concentration reduction as a result of dilution that was assumed for the DOE and NRC analyses. Decreasing the EPRI results by a factor of 10 therefore puts all results on essentially the same basis with respect to the SZ dilution factor.

The similarity of the three sets of TSPA results may be the fortuitous consequence of offsetting assumptions. For example, DOE’s TSPA-VA took credit for cladding performance as a barrier but took no credit for dilution during pumping; NRC’s assumptions were the opposite of these.

Conversely, the similarity may be due to the dominant influence on results of performance factors for which the three analyses made similar assumptions, e.g., those concerning future climate conditions and early waste package failures. For all analyses, the dose rate results at 10,000 years are dominated by radionuclide releases from packages that were assumed to fail



10,000 yr



100,000 yr

Figure 7-47. Comparison of DOE, NRC, and EPRI Performance Assessment Results (derived from NRC99b)

relatively soon after repository closure, and by the highly mobile Tc-99 and I-129 isotopes whose arrival at the dose receptor location is not significantly affected by assumptions concerning phenomena along the UZ and SZ pathway.

After EPA and NRC post-closure radiation protection standards for a possible repository at Yucca Mountain are established, opportunities for differences in assumptions concerning the dose receptor and biosphere pathways will be narrowed. Similarly, the need for assumptions concerning performance parameter values will be reduced by future additions to the data base. However, alternative TSPA modeling approaches can and will be maintained.

7.3.8 Performance Assessments in the Yucca Mountain DEIS

DOE issued its Draft Environmental Impact Statement (DEIS) for disposal of highly radioactive wastes at Yucca Mountain in August 1999 (DOE99a). The DEIS used, as the basis for the Proposed Action, the VA repository design and basic TSPA methodology. Details of TSPA methodologies used for the DEIS and the VA differ, however, because of the differences in the scopes and purposes of the DEIS and the VA.

The VA and the TSPA-VA focused on a single radiation-dose receptor location and a reference design for engineered features of the repository; variations in performance were evaluated as a result of variations in the values of performance parameters for this single, fixed system. In contrast, because of the scope of options considered in the DEIS, the TSPA-DEIS considered alternatives for dose receptor locations, waste quantities and types, and repository designs. Consideration of these options made it necessary to modify some of the details of the TSPA-VA methodology for the TSPA-DEIS evaluations.

7.3.8.1 Comparison of Bases for the DEIS and VA TSPA Evaluations

A principal cause of the differences in TSPA methodologies for the VA and the DEIS is the difference in waste quantities and types considered. As mandated by the Nuclear Waste Policy Act of 1982, the VA considered 70,000 MTU of emplaced wastes, assumed to comprise 63,000 MTU of commercial spent fuel and 7,000 MTU-equivalent of DOE spent fuel and high-level wastes from defense production operations. The DEIS used these waste quantities as the basis for the Proposed Action, but also considered other amounts and types of wastes.

The DEIS alternatives to the quantities and types of wastes considered in the VA and the Proposed Action were designated as Modules 1 and 2. Module 1 increases the quantity of

commercial spent fuel from 63,000 to 105,000 MTU. The latter quantity is based on the Energy Information Administration’s assumption that all currently operating reactors would extend their licenses for a period of 10 years; the quantity represents the expected maximum discharge of commercial spent fuel through 2046. Module 1 also includes all DOE spent fuel and high-level wastes not included in the Proposed Action. This addition increases the mass of such wastes in the repository from 7,000 MTU-equivalent, for the VA reference design and the DEIS Proposed Action, to about 14,000 MTU-equivalent for the Module 1 option.

Module 2 includes the waste inventories of Module 1 plus inventories of other types of wastes that are candidates for disposal at Yucca Mountain. These are designated as Greater-Than-Class-C (GTCC) wastes and DOE Special-Performance-Assessment-Required wastes. The quantities of these wastes are characterized in terms of their volume, which is estimated in the DEIS to total about 6,000 cubic meters. This would correspond to 8 percent of the total volume of Module 2 wastes. The incremental effect of the additional wastes under Module 2 is small in terms of repository radionuclide inventories and their long-term releases from the repository.

Consideration of Module 1 and Module 2 waste inventories and alternative thermal loadings of 85, 60, and 25 MTU/acre produces great variation in the potential size of the repository footprint. Figure 7-48 shows the emplacement blocks identified to accommodate the DEIS disposal inventory options, with the emplacement area for the VA reference design (63,000 MTU commercial spent fuel; 7,000 MTU-equivalent DOE wastes; 85 MTU/acre) shown as the shaded area in the so-called Upper Block. For emplacement of all wastes considered in the DEIS (Modules 1 and 2), at the lowest thermal loading considered (25 MTU/acre), all of the area of all of the blocks shown in Figure 7-48 would be used for waste emplacement. All blocks shown in Figure 7-48 are in essentially the same stratigraphic horizon, i.e., that used for the TSPA-VA. The repository emplacement areas associated with the options considered are summarized in Table 7-11, reproduced from Table I-25 in Appendix I of the DEIS.

Table 7-11. DEIS Estimates of Waste Emplacement Areas

Thermal Load, MTU/acre	Drift Spacing, meters	Emplacement Area, Module 1, acres	Emplacement Area, Modules 1 & 2, acres
85	28	740	1,240
60	40	1,050	1,750
25	38	2,520	4,200

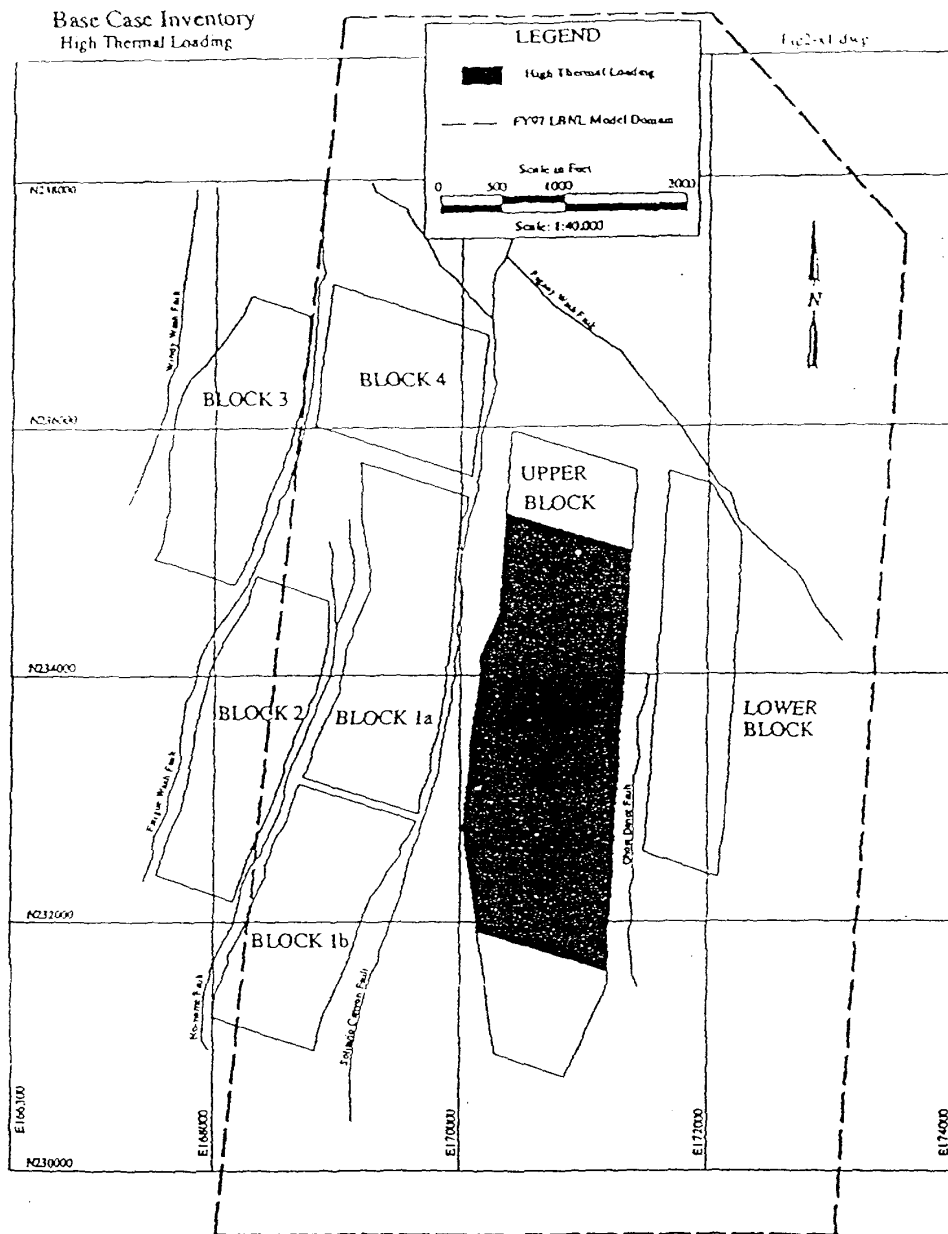


Figure 7-48. Emplacement Block Layout for DEIS Disposal Option

As indicated by Table 7-11, under the options considered by the DEIS, the repository emplacement area could be, at the highest inventory and lowest thermal loading, six times greater than for the VA reference design (i.e., 4,200 acres vs. 740 acres). This potential range of repository conditions necessitated modifications to the TSPA models that were used in the TSPA-VA for the reference VA repository.

The performance assessment codes and their configuration used for the TSPA-DEIS assessments were basically the same as those used for the TSPA-VA (see Section 7.3.2 and Figure 7-36 of this BID). Details of some of the codes were, however, modified for the DEIS evaluations. Basic performance assumptions (e.g., those concerning juvenile failure and wetting of waste packages, amount of spent fuel failed at emplacement, etc.) were the same for both sets of evaluations.

To accommodate the DEIS options for waste inventories, thermal loading, and dose receptor location, the TSPA-VA codes were modified in four areas: thermal hydrology, waste package degradation, waste form dissolution, and elements of the RIP code. Modifications to the RIP code were concerned with the repository environment, the FEHM model, stream tubes, and radionuclide transport paths. Details of the modifications are provided in the DEIS support document, “EIS Performance-Assessment Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain” (DOE99b).

7.3.8.2 Results of the TSPA-DEIS Evaluations

The TSPA-DEIS calculated mean dose rates at 10,000 and one million years for the “average individual” as defined in the VA (Section 7.3.2 of this BID). Biosphere dose conversion factors used in the TSPA-DEIS evaluations were the same as those used in the TSPA-VA. The biosphere dose conversion factors were assumed to have the same value at all dose receptor locations considered (5, 20, 30, and 80 km) even though the 5 km distance, which is not suitable for irrigation or farming, would be a drinking-water-only pathway, and the 80 km distance is a lake playa, where discharge and evaporation of contaminated water would produce deposits of contaminated dust. The assumption of constant dose conversion factors for all distances is stated in the DEIS to be conservative because development of location-specific factors for the 5- and 80-km distances would result in values for the factors that are lower than those actually used.

Results of the TSPA-DEIS dose evaluations analogous to those obtained for the TSPA-VA evaluations are presented in Chapter 5 of the DEIS in tables showing mean peak dose rates for the Proposed Action inventory and the alternative dose receptor distances and areal mass

loadings considered. Results for the Proposed Action inventory and 10,000-year doses, which are presented in Tables 5-4, 5-8, and 5-12 of the DEIS, are summarized in Table 7-12.

Table 7-12. Peak Dose Rates at 10,000 Years for the Proposed Action Inventory and Alternative Distances and Thermal Loads

Dose Receptor Distance, km	Peak Dose Rates at 10,000 Years, mrem/yr		
	85 MTU/acre	60 MTU/acre	25 MTU/acre
5	0.32	0.14	0.13
20	0.22*	0.13	0.06
30	0.12	0.05	0.04
80	0.03	0.003	0.0005

* Corresponds to TSPA-VA base case distance and thermal loading conditions

As noted in the Table 7-12 footnote, the result for the 20-km distance and the thermal loading of 85 MTU/acre (0.22 mrem/yr) corresponds to the base case evaluation in the TSPA-VA, which showed a mean 10,000-year dose rate of 0.1 mrem/yr for 100 probabilistic realizations and a 10,000-year mean dose rate of 0.04 mrem/yr for the deterministic base case evaluation in which expected values of all parameters were used.

The time variations of peak dose rates for times up to 10,000 years in the TSPA-DEIS evaluations are shown, and compared to the TSPA-VA results, in Figure 7-49. Since all basic modeling conditions and assumptions were the same for the two sets of evaluations, differences in the details of the curves can be ascribed to the modifications made to the TSPA-VA codes for the TSPA-DEIS evaluations. The similarity of the curves supports the DOE contention that the model modifications for the DEIS evaluations had only a minor impact on results.

The TSPA-DEIS results for time history of peak doses over a period of one million years are shown in Figure 7-50, which was included in the DEIS documentation as Figure 4.1-3 of the support document, DOE99b. The curves for distances of 5, 20, and 30 km, which are virtually indistinguishable in Figure 7-50, are also virtually identical to the results obtained in the TSPA-VA evaluations, presented in the VA documentation as Figure 4-13 of Volume 3 of DOE98. The DEIS support document, DOE99b, speculates that the low dose rates for the 80 km distance are the result of greater dispersion and radionuclide holdup in the alluvium which, in the performance assessment models, becomes the flow and transport medium at distances beyond about 25 km.

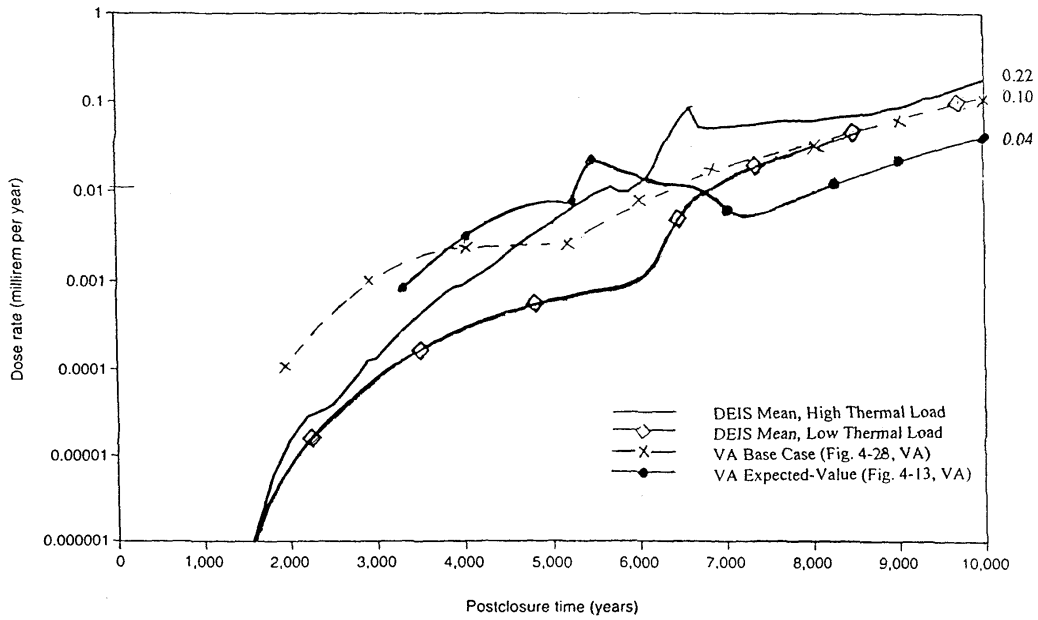


Figure 7-49. Time History of Projected Dose to 10,000 Years, VA and DEIS Evaluations (DOE99a)

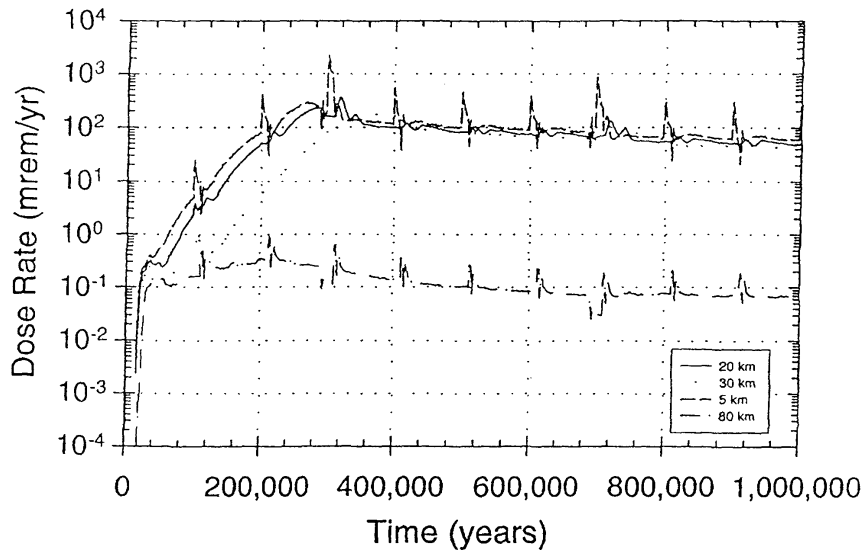


Figure 7-50. DEIS Dose Rate Time Histories for Periods Up to One Million Years (COE99a)

As shown by Figures 7-49 and 7-50, the TSPA-VA and TSPA-DEIS dose rate results are consistent with each other because the performance assessment models, parameters, and assumptions used in the two sets of evaluations were essentially the same. Both sets of evaluations therefore used the conservative assumptions discussed in Section 7.3.3.5 of this BID, and both therefore substantially understate the expected performance of a repository with the design features (i.e., the VA design) that were the basis for the evaluations and the results obtained. If the performance evaluations for the Final EIS are revised to reflect the design features expected to be the basis for the Site Recommendation, i.e., the EDA II design discussed in Section 7.2.2.5 of this BID, the projected performance of the repository would be greatly improved in comparison with the results obtained for the DEIS. Preliminary performance assessment results for the EDA II repository are discussed in Section 7.3.9, and these evolved into the full TSPA for Site Recommendation (TSPA-SR), discussed in Section 7.3.10. The results of the TSPA-SR clearly show dramatic improvement in performance in the 10,000-year time period.

7.3.8.3 DEIS Evaluations of Radionuclide Concentrations in Ground Water

The DEIS used the VA design and modeling methods to calculate ground water concentrations of radionuclides released from the repository. Results were obtained for the various waste inventory, thermal loading, and dose-receptor distance options within the DEIS scope.

The results of the DEIS concentration evaluations for the radionuclides released during periods up to 10,000 years and transported to locations 5, 20, and 30 km downstream from the repository are summarized and compared to the current Maximum Concentration Limits (established for the Safe Drinking Water Act in 1976) in Table 7-13. The predicted ground water concentration values are strongly influenced by the assumed waste package failure at 1,000 years and by assumptions of limited dilution during transport. As a result of the assumptions that maximize the release from the repository and minimize dilution during transport, the radionuclide concentrations shown in Table 7-13 are much higher than would be predicted for more realistic performance assumptions.

Table 7-13. Comparison of DEIS Ground Water Concentrations With MCLs -
All Concentrations in picoCuries/liter.

Radionuclide Contributors to 10K-Yr Dose	Current MCL, in pCi/L	Mean Conc.	95 Percentile	Mean Conc.	95 Percentile
		For 85 MTU/acre, 5 km 20 km 30 km	Conc. For 85 MTU/acre, 5 km 20 km 30 km	For 25 MTU/acre, 5 km 20 km 30 km	Conc. For 25 MTU/acre, 5 km 20 km 30 km
Tc-99	900	45	390	17	1.9
		30	84	7.3	14
		10	130*	4.5	6.3
I-129	1	0.13	0.57	0.10	0.40
		0.07	0.12	0.50	0.15
		0.04	0.20	0.02	0.0
C-14	2,000	2.1	8.2	1.6	5.6
		1.1	1.8	0.79	5.9
		0.64	3.1	0.40	0.21

* The apparent inversions of concentrations with distance are a consequence of the modeling methods used for the DEIS performance evaluations.

As can be seen in Table 7-13, the concentrations reported in the DEIS for the VA repository are well below the current MCL values despite the conservative assumptions that are the basis for the concentration evaluations. These assessments (both the TSPA-VA and DEIS) have subsequently been shown to be conservative compared to current design and TSPA-SR analyses. Consequently, the primary conclusion that the results are below the MCLs remains current and appropriate. TSPA-SR analyses have shown that there are no anticipated releases that can be compared with the MCLs during 10,000 years (see Section 7.3.10.)

7.3.9 Preliminary TSPA Results for the EDA II Design

The current design concept for a repository at Yucca Mountain, EDA II, is discussed in Section 7.2.2.5. DOE intended that the engineered design features used for the VA were a step in repository design evolution, which would culminate in the design that would be used for the License Application if the Yucca Mountain site is approved for disposal. During 1999, the Department therefore defined and assessed alternative improved designs and selected the EDA II concept, discussed in Section 7.2.2.5, as the concept to be used for the Site Recommendation. This design concept was subsequently used as the basis for the TSPA-SR, described in Section 7.3.10.

7.3.9.1 Performance Factors Basis for the EDA II Design

The basis for selection of the EDA II design was, in large measure, the findings and recommendations that emerged from reviews of the VA design concept by parties such as the Nuclear Waste Technical Review Board (TRB99), the NRC staff (NRC99c), and the Total System Performance Assessment Peer Review Panel (PRP99). The reviewers determined that some of the engineered features of the VA repository contributed significantly to uncertainty in the TSPA-VA results and would raise technical issues that would be difficult to resolve during licensing reviews.

Major design features of the VA repository that contributed to TSPA-VA performance uncertainty included:

- The high thermal loading, 85 MTU/acre, and resulting high temperatures in the rocks surrounding the repository, caused significant uncertainties concerning thermal, hydrological, chemical, and mechanical coupling effects. It also caused uncertainties concerning the behavior of rock structure and ground water surrounding the drifts during repository temperature variations with time.
- The use of concrete lining in the drifts caused concerns about the effect of materials in the concrete on the chemical constituents in ground water that contacts waste packages and the effect of those constituents on the corrosiveness of the water.
- The use of carbon steel as the Corrosion Allowance Material and the outer wall of the waste packages and use of Alloy 22 as the Corrosion Resistant Material and the inner wall of the waste packages, caused concern that the carbon steel could create potential for crevice corrosion of the Alloy 22, thereby increasing the rate of penetration of the Alloy 22 and consequently greatly reducing the waste package lifetime.
- The waste packages were not protected from the potential that ground water at the repository horizon could, at times relatively soon after emplacement, drip onto the packages and thereby produce aqueous corrosion, enter the package interior, contact the waste form, mobilize the radionuclides, and transport the radionuclides to the environment.

As a result of these concerns, DOE adopted the EDA II design described in Section 7.2.2.5, i.e., a design with major features consisting of use of drip shields, use of Alloy 22 as the outer wall of the waste packages, increased spacing between drifts, and a thermal loading reduced from 85 MTU/acre to 60 MTU/acre. A comparison of principal design features of the VA and EDA II designs, and the impact of the EDA II design on the uncertainties associated with the VA design,

is shown in Table 7-14. As can be seen from this table, the design changes are responsive to the concerns cited above.

Table 7-14. Impact of EDA II Design Features on VA Performance Uncertainties Identified by Reviewers of the TSPA-VA

Design Feature	VA Repository	EDA II Repository	EDA II Impact
Areal thermal loading	85 MTU/acre	60 MTU/acre	Reduce thermal coupling issues
Drift Liner and Invert Material	Concrete	Steel	Eliminate effect of concrete materials on water chemistry; reduce corrosion rates and radionuclide release rates; increase package lifetime
Drift Spacing	28 meters	81 meters	No temperature rise above boiling point in rock between drifts; reduces overall performance uncertainty
Waste Package Materials	10 cm. Carbon steel over 2 cm. Alloy 22	2 cm. Alloy 22 over 5 cm. 316L stainless	Eliminate crevice corrosion potential; reduce Alloy 22 penetration rate by a factor of 25 or more; increase package life
Peak Waste Package Power	95% above average	20% above average by blending assemblies	Reduce thermal gradients; less driving force for water movement and degradation processes
Drip Shield	None	2 cm. Titanium 7	Protect waste packages; defer contact by water and eliminate juvenile failure potential
Backfill	None	Yes	Divert water from waste packages; protect against rockfall

7.3.9.2 Evolution of the Repository Safety Strategy

In conjunction with adoption of the EDA II design, DOE has also revised its repository safety strategy. As stated in the Rev 4 Repository Safety Strategy document (TRW00), the strategy continues to rely on multiple natural and engineered barriers to achieve safety performance, but

reflects the EDA II enhanced engineered design and improved understanding of the principal performance factors. The revised strategy also is responsive to modifications in the regulatory framework for the potential repository system. It reflects the NRC's change from use of prescriptive subsystem performance objectives to use of a risk-informed, performance based approach, with a total system performance assessment built on defense-in-depth and safety margins.

Implementation of the revised strategy has been accomplished in part by refining and increasing the number of performance factors considered in modeling and assessment of performance. In contrast to the 19 performance factors considered in the TSPA-VA, the Rev 3 strategy considers 27 performance factors, with seven of them identified as factors of principal importance, as shown in Table 7-15. The principal factors are: seepage into drifts; performance of the drip shield; performance of the waste package barriers; solubility limits of dissolved radionuclides; retardation of radionuclide migration in the unsaturated zone; retardation of radionuclide migration in the saturated zone; and dilution of radionuclide concentrations during migration.

A technique defined as "neutralization analysis" was used to help identify the principal factors. In using this technique, specific performance factors are removed from the performance modeling system to determine its impact on overall performance. An example of the use of neutralization analysis, reproduced from TRW00, is shown in Figure 7-51.

7.3.9.3 Results of early TSPA Evaluations for the EDA II Design

Interim results of some TSPA evaluations for the EDA II repository were published leading up to the completion of the TSPA for Site Recommendation (TSPA-SR). These interim results clearly illustrate some of the advantages of the EDA II design, and are reproduced in this section for the sake of completeness and clarity. However, some of the issues raised in these interim analyses were justifiably eliminated in the subsequent TSPA-SR, and the reader is referred to Section 7.3.10 for the latest information on TSPA.

Figures 7-52 and 7-53 show results presented at the NWTRB meeting of June 1999 (TRB99a). Results presented in the TSPA-VA (DOE98) have been added to the figures to show the comparison with TSPA-VA results. From the similarity of the shape of the curves in Figure 7-52, it is evident that the basis for the TSPA-VA and TSPA-EDA II analyses was at least similar, if not the same, e.g., assumption of juvenile waste package failure at 1,000 years, accompanied by the other performance assumptions detailed in the TSPA-VA (DOE98, Vol.3).

In contrast to the results shown in Figures 7-52 and 7-53, The Rev 3 repository safety strategy document, TRW00, states that the base case TSPA analysis for the EDA repository shows that radionuclide releases are not expected for more than 100,000 years. These results take into account the expected performance of the titanium drip shields and the waste packages with Alloy 22 as the corrosion resistant wall material on the outside.

A scenario which assumed common-mode failure of a drip shield and the underlying waste package at 9,000 years produced the radionuclide release and dose results shown in Figure 7-54, which is reproduced from Figure 2-2 of TRW00. This dose/time-history curve is the same as the base case for the neutralization analysis shown in Figure 7-51.

The early-failure scenario represented by the dose/time curve shown in Figure 7-54 is extremely improbable. Its probability of occurrence is on the order of 10^{-9} , estimated as follows:

- One drip shield out of 10,000 has to fail by being penetrated by water.
- For the common-mode failure to occur, the failed drip shield has to be over a waste package that is vulnerable to early failure as a result of a phenomenon such as a poor weld. About 0.1 percent, or one out of 1,000 packages, might have such vulnerability.
- The failure in the drip shield has to be over the vulnerable part of the waste package surface, e.g., the weld bead. If the hole in the drip shield is on the order of 2 cm (one inch) in diameter, and the waste package is on the order of 600 cm (six meters) in length, the chance of the drip shield hole being over the vulnerable part of the package surface is on the order of one in 100.

Combination of these factors produces the result that the probability of the early common-mode failure leading to nuclide releases after about 10,000 years, as shown in Figure 7-54, is $(10^{-4})(10^{-3})(10^{-2}) = (10^{-9})$. This is the same order of likelihood of occurrence as that for volcanic intrusion of the repository. This result, in combination with the base case results showing no radionuclide releases for 100,000 years, suggests that the basic paradigm for TSPA methodology and assumptions for EDA II repository performance evaluations should be significantly different from that used for the TSPA-VA analyses.

Table 7-15. Factors Potentially Important to Postclosure Safety (TRW00)

Key Attributes of the Repository System	Principal Factors in the Viability Assessment	Factors for the Enhanced Repository System
Limited Water Contacting Waste Package	Precipitation and Infiltration into the Mountain	Climate Infiltration
	Percolation to Depth	Unsaturated Zone (UZ) Flow above Repository
	Seepage into Drifts	Seepage into Drifts*
	Effects of Heat and Excavation on Flow	Coupled Processes—Effects on UZ Flow
		Couples Processes—Effects on Seepage
	N/A	Environments on Drip Shield
	N/A	Performance of Drip Shield*
Long Waste Package Lifetime	Dripping onto the Waste Package	Environments on Waste Package
	Humidity and Temperature at the Waste Package	
	Chemistry on the Waste Package	
	Integrity of Outer Waste Package Barrier	Performance of Waste Package Barriers*
	Integrity of Inner Waste Package Barrier	
Low Rate of Radionuclide Release from the EBS	Seepage into Waste Package	Environments within Waste Package
	Integrity of Commercial Spent Nuclear Fuel (CSNF) Cladding	Commercial Spent Nuclear Fuel (CSNF) Waste Form Performance
	Dissolution of UO ₂ and Glass Waste Forms	DOE-Owned Spent Nuclear Fuel (DSNF), Navy Fuel, and Plutonium Disposition Waste Form Performance
		Defense High-Level Waste (DHLW) Waste Form Performance
	Solubility of Neptunium-237	Solubility Limits of Dissolved Radionuclides*
	Formation of Radionuclide Bearing Colloids	Colloid Associated Radionuclide Concentrations
	Transport within and out of the Waste Package	In-Package Radionuclide Transport
	EBS Radionuclide Migration—Transport Through Invert	Transport through invert
Delay and Dilution of Radionuclide Concentrations During Transport Away from the EBS	Transport through Unsaturated Zone (UZ)	Advection Pathways in the UZ
		Retardation of Radionuclide Migration in the UZ*
		Colloid Facilitated Transport in the UZ
		Coupled Processes—Effects on UZ Transport
	Transport in the Saturated Zone (SZ)	Advection Pathways in the Saturated Zone (SZ)
		Retardation of Radionuclide Migration in the SZ*
		Colloid Facilitated Transport in the SZ
	Dilution from Pumping	Dilution of Radionuclide Concentrations during Migration*
Biosphere Transport and Uptake	Biosphere Transport and Uptake	

*Principal factors of the postclosure safety case.

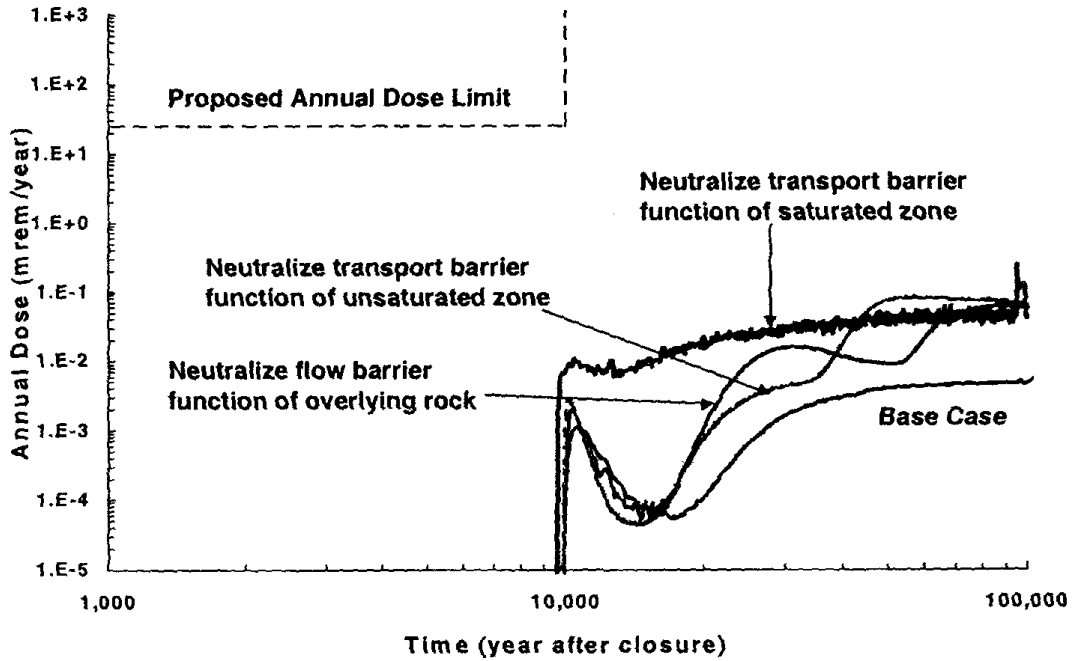


Figure 7-51. Barriers Importance Analysis to Assess Natural Barriers of the Repository System-Early Waste Package Failure Scenario (TRW00)

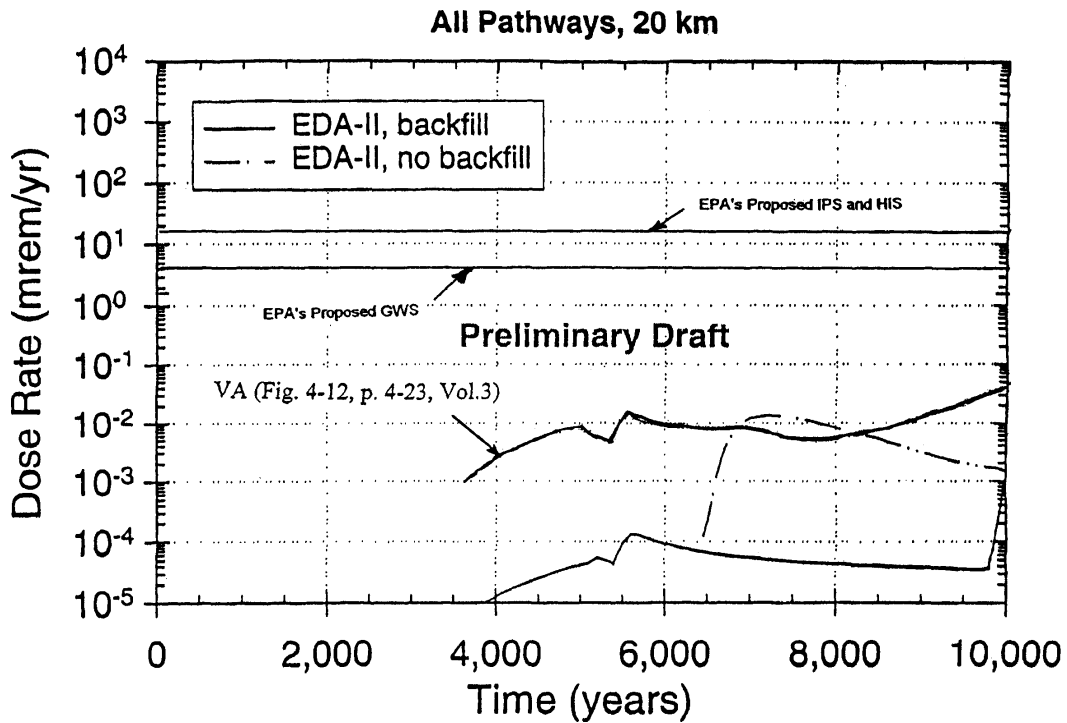


Figure 7-52. Comparison of VA and EDA 10,000-Year Doses (TRB99a)

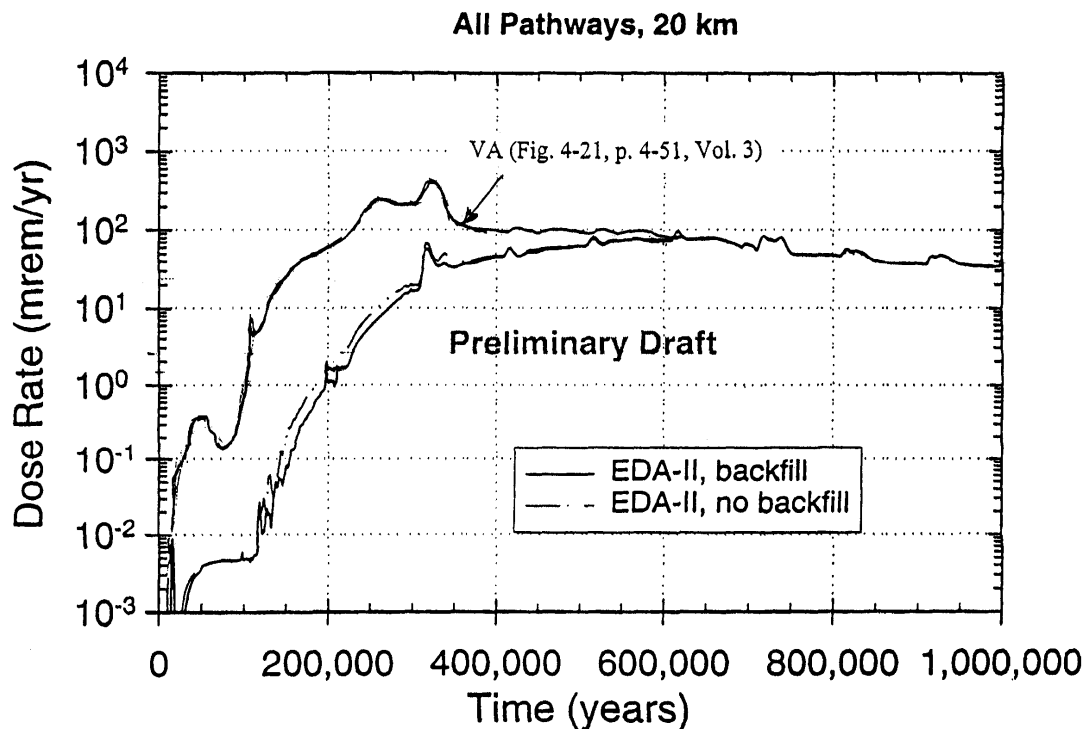


Figure 7-53. Comparison of VA and EDA Million-Year Doses (TRB99a)

The “Performance Factor” results presented in Table 7-8 of this BID and the TSPA-VA results presented in DOE98 Vol. 3 can be combined to produce the comparison of major radiation dose milestones for the VA and EDA II repositories shown in Table 7-16. As can be seen from the values in this table, the expected performance of the EDA II repository is significantly better than that of the VA repository. This is the result of the design features selected specifically to improve expected performance and to reduce uncertainties in expected performance and in the results of TSPA evaluations. As previously noted (TRW00), the EDA II repository would not be expected to release radionuclides and to cause radiation doses for more than 100,000 years.

The peak dose values in Table 7-16 are the result of use of highly conservative assumptions used in the TSPA-VA and TSPA-EDA II evaluations, such as the assumption that all of the area of the waste form in a commercial fuel rod with breached cladding is exposed to, and contacted by, water that enters the interior of a waste package whose wall is penetrated by water (see Section 7.3.3.5 of this BID). Use of more realistic assumptions would reduce the estimated peak dose rates by several orders of magnitude.

Table 7-16. Comparison of Major Dose Milestones for the VA and EDA II Repositories

Dose Parameter	VA Repository	EDA II Repository
Dose at 10,000 Years, mrem/yr	0.04	0.001
Time to 15 mrem/yr, years	70,000	290,000
Time to 25 mrem/yr, years	90,000	310,000
Peak Dose During One Million Years	350 mrem/yr	85 mrem/yr
Time of Peak Dose, years	320,000	650,000

7.3.10 Performance Evaluation for the Site Recommendation

Since selection in 1999 of the EDA II design (Section 7.2.2.5) to be the basis for the Site Recommendation (SR), and publication of Revision 3 of the Repository Safety Strategy in January 2000 (TRW00), DOE continued to evolve and refine the design and strategy concepts for use in performance evaluation for the SR. This evolution has led to the recent publication of Rev 4 of the Repository Safety Strategy in November 2000, and the publication of the Total System Performance Assessment for Site Recommendation (TSPA-SR). This last document has now replaced the TSPA-VA as the current iteration of the TSPA for Yucca Mountain. In this section, the key features of the Repository Safety Strategy (Section 7.3.10.1) and the TSPA-SR (Section 7.3.10.2) are described. Very recently, the EIS for Yucca Mountain has been supplemented

7.3.10.1 Evolution of the Repository Safety Strategy

Key features of Revision 3 of the Repository Safety Strategy (TRW00) are summarized in Section 7.3.9.2 above. That version of the strategy was developed to be responsive to comments on the VA design and TSPA models, and to conform to anticipated regulatory requirements. The Rev 3 strategy is reflected in the EDA II design, described in Section 7.2.2.5.

Revision 4 of the Repository Safety Strategy was described by DOE in the June 6-7 Technical Exchange and compared to the Revision 3 strategy (DOE00a). The comparison of important factors in the Rev 3 and Rev 4 strategies is summarized in Table 7-17. This comparison indicates that Revision 4, in comparison with Revision 3, is based on more extensive analyses, improved performance assessment models, a broader data base, and an integrated and extended evaluation of factors important to performance.

Table 7-17. Comparison of Rev 3 and Rev 4 Repository Safety Strategies

Strategy Element	Revision 3	Revision 4
Principal Factors	<p>Subjective judgments about performance factors</p> <p>Judgments supported by barrier neutralization analyses</p> <p>No consideration of disruptive factors</p>	<p>Based on extensive TSPA analyses and barrier importance analyses</p> <p>TSPA includes both nominal and igneous activity scenarios</p>
Performance Assessment	Use of VA models	<p>Updated, fully-documented models</p> <p>Probabilistic analyses to address uncertainties</p>
Measures to Increase Assurance of Safety	<p>Preliminary consideration of safety margins and defense in depth</p> <p>Initial plans for safety assurance, performance confirmation</p>	<p>Full evaluation of safety margins and defense in depth</p> <p>Full documentation of potentially disruptive features, events, and processes</p> <p>Rev 1 of Performance Confirmation Plan</p>

Rev 4 of the Repository Safety Strategy revises the Principal Performance Factors established in Rev 3, which are discussed in Section 7.3.9.2 and shown in Table 7-15. Rev 4 retains all of the Rev 3 principal factors except wellhead dilution, which is no longer considered to be a principal factor because dilution factors are expected to be determined by regulation. Rev 4 adds four new principal factors to the remaining Rev 3 list, which includes waste package performance, seepage, drip shield performance, dissolved radionuclide concentrations, and UZ and SZ travel times. The principal performance factors added by Rev 4 are:

- Colloid-associated radionuclide concentrations, added for defense-in-depth.
- Probability of igneous activity. Radionuclide release is possible in less than 10,000 years; risk depends on probability of occurrence.
- Effects of igneous activity on the repository. Risk depends on damage to waste packages and drip shields.
- Biosphere dose conversion factors. Radionuclide release is possible in less than 10,000 years; risk depends in part on biosphere transport and uptake.

Overall, the barriers seen to be potentially most important to waste isolation include the overlying rock, drip shield and waste package performance, and the UZ and SZ radionuclide transport barriers. Other important barriers are the commercial spent fuel cladding, HLW canisters within the HLW waste packages, the drift invert, and the inner waste package barrier, which is planned to be 100 mm of stainless steel (DOE00b).

The DOE00a presentation also identified and addressed potential performance assessment vulnerabilities under the Rev 4 strategy. These vulnerabilities and the envisioned means to mitigate them are shown in Table 7-18.

Table 7-18. Potential Performance Assessment Vulnerabilities and Mitigation Measures

Potential Vulnerabilities	Mitigation Measures
Adequacy of treatment of model uncertainty	Mitigate through defense in-depth, including analysis of rockfall effects
Over-conservatism in some models	Studies to assess appropriateness of less conservatism in key models
Thermal loading issues	Improve basis for thermal design for LA; Use flexible design that can be modified after performance confirmation testing; maintain options until final selection
Potential for igneous activity	Demonstrate low probability, low risk, and high margin
Reliability of complex metal barriers	Use defense-in-depth and alternative EBS concepts
Possibility of peak dose rate exceeding 100 mrem/yr	Reduce conservatism in key models, i.e., solubilities of Np and Pu, UZ and SZ flow and transport

DOE completed documentation of Rev 4 of the Repository Safety Strategy in November 2000.

7.3.10.2 TSPA-SR

The methodology used for the SR (TSPA-SR) was described by TRW00b. The methodology is a step in iterative development of TSPA methodology, in that it includes improvements in comparison with VA models and methodology that are responsive to comments on the VA, and that traceability has been improved in comparison with the VA. The TSPA approach has also been modified for the TSPA-SR to meet anticipated EPA and NRC regulatory requirements. Documentation of the TSPA-SR methodology, models, and bases is included in Analysis/Model Reports (AMRs) and Process Model Reports (PMRs) that are derived from the AMRs. There are more than 100 AMRs and 9 PMRs.

The reference repository design for the TSPA-SR is basically the EDA II design discussed in Section 7.2.2.5 of this BID. Key design features include:

- Average thermal load of 62 MTHM/acre
- 50 years of ventilation
- Pre-emplacement blending of fuel to level the thermal load
- Titanium drip shields over the waste packages
- No backfill
- End-to-end loading of waste packages, with close spacing
- All waste packages have 20 mm outer layer of Alloy 22 and 100 mm inner layer of stainless steel; weld stresses mitigated by laser peening

An important feature of the TSPA-SR in comparison with the TSPA-VA is that the TSPA-SR starts with a comprehensive identification and screening of features, events, and processes (FEPs) that can affect repository system performance (the VA used assumed conditions and omitted some FEPs). Scenarios for the retained FEPs are constructed and screened, and implementation of the scenarios in the TSPA is specified. The TSPA-SR includes scenarios for expected (“nominal”) conditions, and scenarios for low-probability disruptive events: volcanism, and human intrusion.

TSPA dose projection results were produced for the nominal conditions, the volcanic scenarios, the human intrusion scenario, and ground water concentrations in comparison with the EPA ground water protection standards. Unlike the earlier TSPA-VA, the nominal and igneous disruptive event are combined via their probabilities into a single probability-weighted dose consequence curve that is compared with performance objectives. Evaluation results include characterization of uncertainties and their significance.

The TSPA-SR performance assessment model is an upgraded version of the TSPA-VA model. The core dose calculation model, which receives input from models of performance factors such as UZ and SZ flow and transport, is now termed GoldSim and is an improved version of the RIP code used for the TSPA-VA. The basic code configuration for the TSPA-SR evaluations is similar to that for the TSPA-VA, which is shown in Figure 7-36.

The TSPA-SR methodology differs significantly from the TSPA-VA methodology in that it includes consideration of disruptive events retained after screening of all FEPs. As described in the DOE/NRC June 6-7, 2000 Technical Exchange (DOE00c), Igneous activity and seismic damage to cladding were screened into the performance scenarios to be considered. Seismic effects on cladding were included in the evaluations for nominal performance of the repository.

All other potentially disruptive events were screened out, i.e., faulting, ground motion damage to the EBS, rockfall, seismically-induced water table rise, and nuclear criticality were eliminated from consideration.

DOE developed the human intrusion scenario for the TSPA-SR to be consistent with existing guidance in the draft 40 CFR 197 (EPA99), the proposed version of 10 CFR 63 (NRC99), and the proposed version of 10 CFR 963 (DOE99a). The implementation of the regulatory requirements was conducted in the TSPA-SR as shown in Table 7-19 (TRW00b). The central feature for treatment of these requirements was to be consistent with the more conservative of the proposed requirements from the draft regulations. Most notably, the intrusion is assumed to occur at 100 years, consistent with the proposed NRC requirement (NRC99). Intrusion at later times, when (consistent with EPA99) a waste package might more reasonably be degraded enough to be unrecognizable as an intrusion event, was treated as a sensitivity case study.

Table 7-19 Implementation of regulatory requirements in the TSPA-SR for regulatory requirements (Table excerpted from TRW00b).

NRC Base Assumptions (from Proposed 10 CFR Part 63)	EPA Additional and/or Conflicting Assumptions (from Proposed 40 CFR Part 197)	Conceptualization for TSPA-SR
Assumed intrusion is a drilling event.	Assumed intrusion is acute and inadvertent.	Inadvertent drilling event.
Drilling result is a single, nearly vertical borehole that penetrates a waste package and extends down to the SZ.	Borehole penetrates a degraded waste package.	Single vertical borehole from surface through a single waste package to the SZ.
Intrusion occurs 100 years after closure	Intrusion time should take into account the earliest time after disposal that a waste package could degrade sufficiently that current drilling techniques could lead to waste package penetration without recognition by the drillers.	Intrusion occurs at 100 years (a 10,000 year intrusion time is examined in a sensitivity simulation).
Borehole properties (diameter, drilling fluids) are based on current practices for resource exploration.	Borehole results from exploratory drilling for groundwater.	Borehole diameter consistent with an exploration groundwater well.

NRC Base Assumptions (from Proposed 10 CFR Part 63)	EPA Additional and/or Conflicting Assumptions (from Proposed 40 CFR Part 197)	Conceptualization for TSPA-SR
Borehole is not adequately sealed to prevent infiltrating water.	Natural degradation processes gradually modify the borehole, the result is no more severe than the creation of a groundwater flow path from the crest of Yucca Mountain through the potential repository and to the water table.	Infiltration and transport through the borehole assumes a degraded, uncased borehole, with properties similar to a fault pathway.
Hazards to the drillers or to the public from material brought to the surface by the assumed intrusion should not be considered.	Only consider releases through the borehole to the SZ; consider releases occur gradually through air and water pathways, not suddenly as with direct removal.	Groundwater is only pathway considered.
A separate consequence analysis is required, identical to the performance assessment, except for the occurrence of the specified human intrusion scenario.	Unlikely natural processes and events are not included, but analysis could include disturbances by other processes or events that are likely to occur.	Intrusion borehole is applied to nominal case; effects of volcanism are not included.
Peak dose is not to exceed 25 mrem/yr. in the first 10,000 years.	Peak dose is not to exceed 15 mrem/yr. In the first 10,000 years.	Does not affect simulations.

The approach used in TSPA-SR for evaluating these conditions are shown in Table 7-20. The analyses are based on a representation of an exploratory drilling intrusion, which leads to disruption of a waste package and an enhanced pathway through the unsaturated zone. The saturated zone and biosphere analysis are the same as in the nominal scenario.

Table 7-20 Technical Assumptions Implemented in the Human Intrusion Scenario in TSPA-SR (Table excerpted from TRW00a).

Issue	Key Component Affected	TSPA-SR Implementation
Borehole diameter	Infiltration Borehole Transport	Typical water well borehole has a diameter of 20.3 cm (8 in.)
Infiltration into borehole	Infiltration	Assumed infiltration rate distribution is based on modeled infiltration in the Yucca Mountain region for the glacial transition climate. Values at the high end of the distribution inherently include the possibility of surface water collection basin focusing.
Seepage into penetrated waste package	Infiltration Waste Mobilization	Volumetric flux is equivalent to infiltration rate times borehole area. Volume of drilling fluid is ignored.
Type of waste package penetrated	Waste Mobilization	Sampled from CSNF and co-disposed waste packages. Co-disposed packages contain both DSNF and HLW glass.
Thermal and geochemical conditions in waste package	Waste Mobilization	Assume temperature and in-package chemistry as calculated in nominal scenario. This assumes Well J-13 water and ignores any chemical effects of the drilling fluid.
Waste form degradation	Waste Mobilization	Waste in penetrated package is assumed to have perforated cladding from drilling disturbance.
Solubilization of radionuclides in water	Waste Mobilization	Infiltrating water can mix with waste in entire waste package. Solubility is based on temperature and in-package chemistry as in nominal scenario.
Borehole flow and transport properties	Infiltration Borehole Transport	Volumetric flux consistent with seepage into the waste package. Transport properties consistent with a UZ fault pathway.
Borehole location	Infiltration SZ Transport	Random over the footprint of the potential repository. Uncertainty in location is captured in infiltration rate and location that radionuclides enter the SZ.
Borehole length	Borehole Transport	Borehole length from the potential repository to SZ conservatively assumes water level consistent with glacial transition climate.
SZ	SZ Transport	Assume SZ flow and transport properties identical to nominal scenario.
Biosphere processes	Biosphere	Assume exposure pathways and receptor characteristics identical to nominal scenario.

In the igneous disruption scenario, a dike is assumed to intersect the repository as a result of volcanic activity. Probabilities of an intrusion event resulting in a dike intersecting the repository, and of volcanic eruption as a result of dike intrusion, are established by expert elicitation. Repository response to the dike intrusion, and dose consequences of volcanic eruption, were evaluated on the basis of repository design features, site data, and use of data from the 1995 Cerro Negro eruption as an analog.

The eruption was assumed to be a violent strombolian type eruption. The igneous scenario is subdivided into two scenarios: eruption and intrusion. The eruption scenario refers to penetration of the repository, leading to total disruption of waste packages and drip shields encountered by the magma, bringing waste to the surface. Doses result from ash eruption, with downwind transport, redistribution of ash at the surface, and subsequent human exposures. The intrusion scenario refers to penetration of the repository by magma, leading to total disruption of waste packages and drip shields encountered by the magma, but without further movement of radionuclides. However, since the engineered barriers are assumed to be totally destroyed, this scenario functions as equivalent to assessing juvenile failures of waste packages. Releases for the magma intrusion scenarios are via releases to groundwater from the disrupted waste packages. Biosphere dose conversion factors were modified to account for the effect of ash discharge and fallout on dose potential as a result of the eruption on biosphere pathways.

The TSPA-SR methodology also differs significantly from that of the TSPA-VA in its treatment of saturated zone flow and radionuclide transport. The presentation at the June 6-7, 2000 DOE/NRC Technical Exchange (DOE00d) described the changes as follows:

- The 3-D SZ site-scale flow and transport model is used to simulate radionuclide transport in the TSPA-SR (vs. the streamtube approach in the TSPA-VA)
- Radionuclide concentrations are calculated in the water supply of the hypothetical farming community in TSPA-SR (vs. concentration in the SZ, as in TSPA-VA)
- Matrix diffusion is explicitly simulated in the SZ site-scale model for TSPA-SR (vs. use of the effective porosity approach for transport in fractured media used in TSPA-VA)
- Particle tracking method used for radionuclide transport in the SZ site-scale model for TSPA-SR (vs. finite element transport method used in streamtubes for TSPA-VA)
- Minor sorption of Tc and I in alluvium for TSPA-SR (vs. no sorption in TSPA-VA)

The SZ site-scale flow and transport model was calibrated using data from ongoing DOE and Nye County drilling and measurement programs. At present, an “Alluvial Uncertainty Zone” has been defined as a result of limited data. The northern boundary of the alluvium varies across the entire uncertainty zone; the western boundary of the alluvium varies approximately from the Fortymile Wash channel to the tuff outcrops in the west; and the flow path length in the alluvium (to the 20-km radius) varies from about 1 up to 9 km..

The TSPA-SR approach to biosphere modeling is basically the same as that which was used in the TSPA-VA. The definition of dose receptors was based on the definitions in the draft EPA and NRC regulations, e.g., the EPA’s Reasonably Maximally Exposed Individual and the NRC’s Critical Group. An analysis of uncertainty in ground water usage by a hypothetical farming community was performed on the basis of current water usage and demographic data. An analysis of radionuclide buildup in soils from long-term irrigation with contaminated ground water was also performed for the TSPA-SR.

The TSPA-SR methodologies and information outlined above were included in the AMRs and the PMRs, which provide detailed documentation of technical approaches and justifications used in the TSPA-SR.

Results for the TSPA-SR are shown in Figure 7-54 for the igneous and nominal scenarios. There are no doses during the first 10,000 years from the nominal scenario. This is a substantial change for the TSPA-VA results, in which the design led to a significant potential for juvenile releases. These juvenile releases have been eliminated by the EDA II design, with the results shown in the figure. Doses during the period up to 2000 years are dominated by the igneous eruption scenario (TRW00b). From the period between 2000 years to after 10,000 years, the igneous intrusion scenario becomes the most important scenario (TRW00b). Subsequent to that time, all three scenarios (nominal, eruption, and igneous intrusion) play a role in establishing the dose curve. TRW00b also reported the results of uncertainty analyses that led to the mean dose rates shown in Figure 7-54.

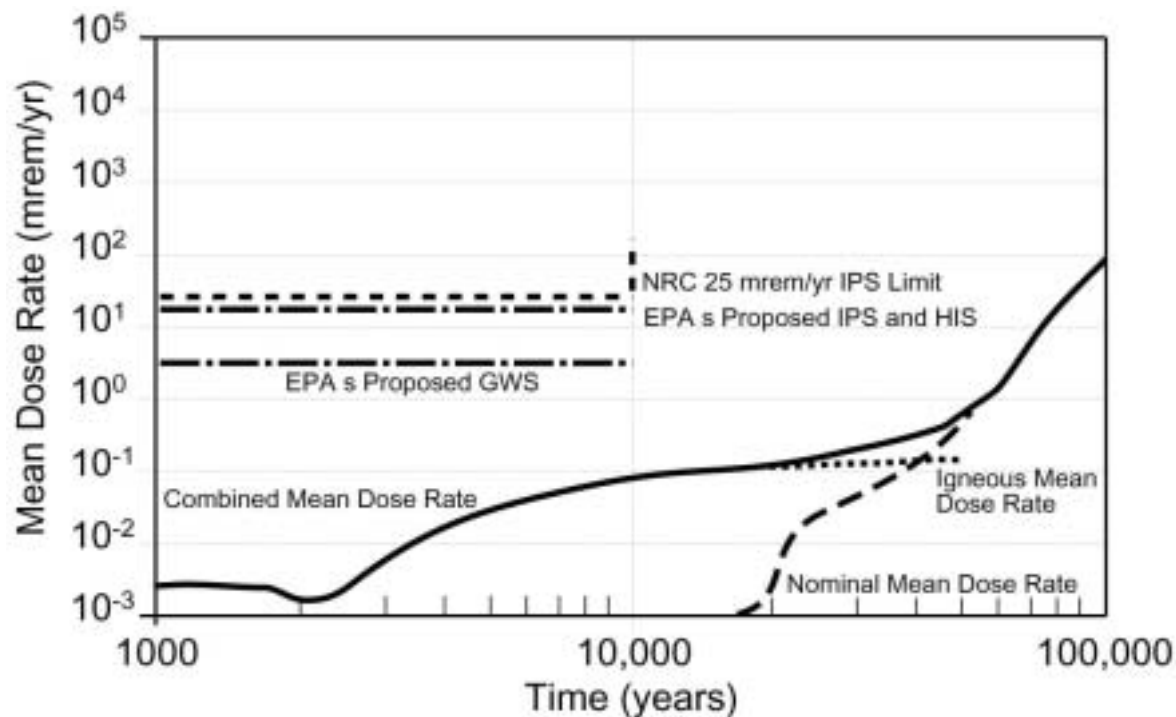


Figure 7-54 Comparison of Proposed Radiation Protection Standards with Expected Values of TSPA-SR Calculations for a Repository at Yucca Mountain for Nominal and Igneous Scenarios (Figure adapted from TRW00b).

Results from the TSPA-SR human intrusion analysis are shown in Figure 7-55. The base case analysis is the result of an assumed intrusion event at 100 years after closure. Also shown on the figure is the result of an inadvertent intrusion even at 10,000 years after closure.

Additional results were presented by TRW00b for comparison with proposed groundwater criteria. The TSPA-SR indicates that no radionuclide releases from the EDA II repository would be expected during 10,000 years unless it is violently disrupted by volcanic activity. The results for the EDA II design from the TSPA-SR for comparison with the ground water protection MCLs are shown in Figures 7-56 and 7-57. The groundwater protection analyses assumed a representative water volume of 1285 acre-feet/yr centered on the highest concentration in the plume in the saturated zone. It was recognized in the TSPA-SR (TRW00a) that the regulatory time period for groundwater protection is 10,000 years. However, the analyses were carried out to 100,000 years to ensure that no significant degradation of the performance occurs after 10,000 years.

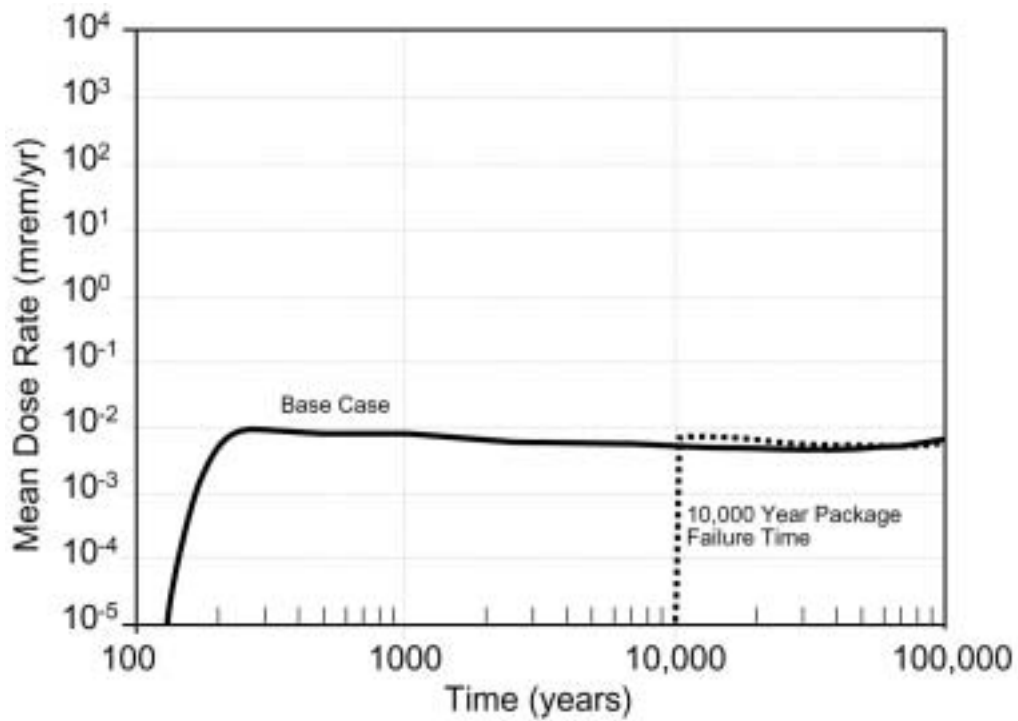


Figure 7-55. Expected Values of TSPA-SR Calculations for a Repository at Yucca Mountain for the Inadvertent Human Intrusion Scenario (Figure adapted from TRW00b).

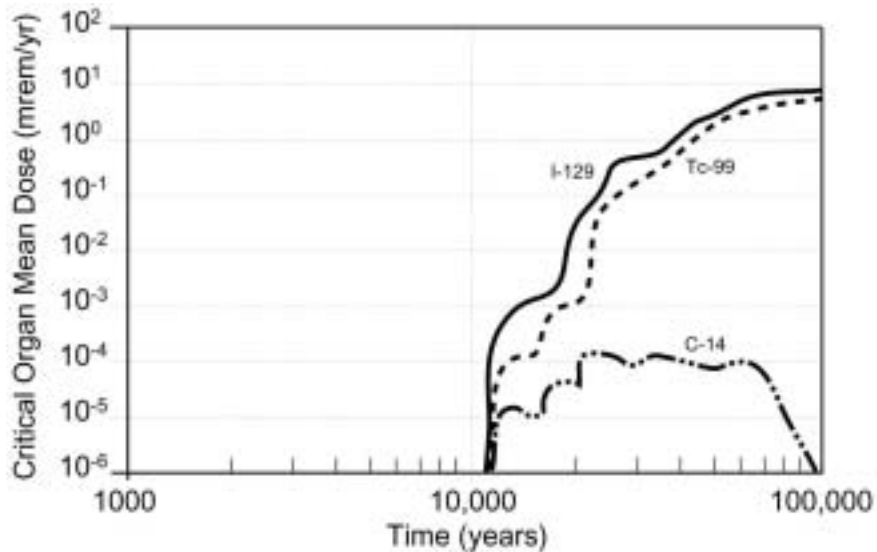


Figure 7-56. Summary of Groundwater Protection Performance Results of the TSPA-SR: Combined Beta and Photon-Emitting Radionuclides (Figure adapted from TRW00b).

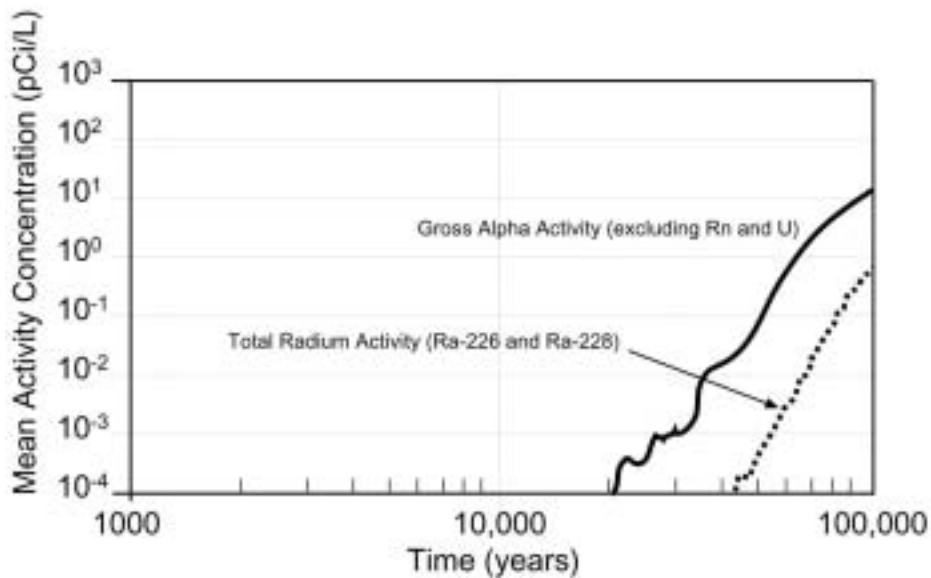


Figure 7-57. Summary of Groundwater protection Results for TSPA-SR for Gross Alpha Activity (Figure adapted from TRW00b).

Another facet of the Safety Strategy has been an extensive evaluation of parameter uncertainty and sensitivity. The TSPA-SR (TRW00a) reported three kinds of evaluations of parameter uncertainty and sensitivity: Uncertainty Importance Analysis, Sensitivity Analysis, and Robustness Analysis. *Uncertainty Importance Analysis* refers to the use of regression analyses to determine the most important parameter contributors to the spread of output results, and classification-tree analyses to determine the parameters leading to extreme outcomes in the distributions. *Sensitivity Analysis* refers to single-parameter sensitivity analyses, in which one parameter is varied while the others are held at particular values. *Robustness Analysis* (also referred to as Degraded Barrier Analysis in the TSPA-SR) refers to a focused approach to examining parameters associated with extreme degradation behavior of individual barriers, keeping intact the remaining analysis of the system.

Uncertainty importance analyses were performed beginning with stepwise linear rank regression analysis. The results of this analysis were evaluated using classification and regression tree analysis to determine decision rules that determine whether a particular realization would

produce doses at the upper or lower end of the output distribution. These approaches were used to evaluate the spread in doses at a particular time and the spread of times needed to produce a particular dose. Particular attention was also focused on the extreme high end of the output distribution, to determine which parameters lead to the extremes of the output.

The uncertainty importance analyses showed that the waste package and saturate zone processes are the most important factors in the nominal scenario, whereas the probability of the occurrence of igneous disruption of the repository is the most important factor for igneous scenarios. As discussed in the TSPA-SR, the assessment that these are the “most important” in this uncertainty importance analysis reflects two factors: the change in variance of dose rate with variance of the parameter, and the change of the dose rate itself with changes in the parameter. If either of these two derivatives is small, the techniques used in the TSPA-SR will tend to show the parameter to be unimportant.

Sensitivity analysis, as used in the TSPA-SR, refers to a single parameter variation method. This is considered to be a complementary technique to the uncertainty importance analysis. In this approach, a single parameter was ranged between its 5th and 95th percentiles, and other parameters were fixed at particular values.

Robustness analysis was conducted by setting a suite of parameters associated with a particular barrier at their 5th or 95th percentile, whichever tends to maximize the dose rate over the time period of interest. For the sake of completeness, the results are also shown compared to results from the same suite of parameters set at the opposite end of the behavior (i.e., values which tend to minimize dose consequences). The intent of these robustness analyses is to present the behavior of the system as a whole if any part of the system degrades quickly, and functions according to its extreme behavior. Robustness analyses were conducted on nine facets of system behavior (TRW00a):

- UZ. This barrier represents the function of the UZ above the potential repository in limiting the amount of water that reaches the potential repository. This barrier includes the climatic conditions at Yucca Mountain, the processes at and near the surface that lead to infiltration, and flow through the UZ above the potential repository. Parameters treated in the robustness analysis were the seepage-uncertainty factor and the flow-focusing factor. Degraded conditions for these parameters resulted in a small increase in dose rate over the base case.
- Seepage into emplacement drifts. This barrier represents the function of the drifts themselves as a capillary barrier that limits the amount of water that enters the drifts.

Both infiltration and seepage parameters were set to their degraded behavior for this analysis. Degraded conditions for these parameters resulted only in about a factor of 5 increase in dose rate over the base case.

- Drip shield. The first of the engineered barriers, the drip shield limits the amount of water that reaches the waste package. In the robustness analysis, the general corrosion rate parameters were set to their extreme values. While the drip-shield lifetime is significantly degraded in this analysis, there is almost no change in the dose rate. This results reflects the fact that the waste package degradation model is independent of the drip shield function. This appears to be an example where the high degree of conservatism in one model masks the importance of a different function, as discussed in TRW00a.
- Waste package. The primary engineered barrier, the waste package limits the amount of water that reaches the waste form and limits radionuclide transport out of the EBS. Degradation parameters considered in the robustness analysis were: residual hoop-stress state and stress intensity factor at the closure-lid welds, Number of manufacturing defects at the closure-lid welds per waste package, Alloy-22 general corrosion rate, microbially-induced corrosion enhancement factor for general corrosion, and enhancement factor for Alloy-22 general corrosion from aging and phase stability. The enhanced case (optimistic parameters) led to no releases from the waste package for the first 100,000 years. The degraded parameters show a somewhat earlier failure profile, with first failure occurring at 7,000 years compared to 12,000 years for the base case. For the degraded case there is 50 percent probability that 1 percent of waste packages fail at about 10,000 years and 10 percent of waste packages fail at about 12,000 years. For the base case it is about 25,000 years for the 1 percent failure and about 50,000 years for the 10 percent failure. Accordingly, the predicted mean dose starts earlier (about 8,200 years versus about 15,000 for the base case), and the predicted mean dose rates are much higher.
- CSNF cladding. The Zircaloy cladding is an engineered barrier that is part of the waste form. It limits the amount of water that reaches the CSNF portion of the waste and limits radionuclide transport out of the CSNF waste form. (CSNF is planned to be approximately 90 percent of the mass of waste in the potential repository.) Four of the five parameters in the cladding degradation model were evaluated in the robustness analysis: the number of rods initially perforated in a CSNF waste package, the uncertainty in localized corrosion rate, the uncertainty of the CSNF degradation rate, and the uncertainty in the unzipping velocity of the cladding. It was concluded that these parameters are unimportant for performance in the first 100,000 years, but that they contribute to the spread of doses during the period 100,000-1,000,000 years. The effect of these parameters on dose rate in the robustness analysis is not reported by TRW00a.

- Concentration limits. This barrier represents the function of environmental conditions and radionuclide solubility limits in limiting radionuclide transport out of the EBS. The primary dose contributor in the first 30,000 years is technetium-99. The solubility of Tc-99 is assumed to be large (1 M), and is not treated as uncertain. The primary radioelements for the period after 30,000 years are neptunium, americium, and uranium. The solubilities of each of these is controlled by pH in the TSPA-SR model. The pH, in turn, is assumed to not vary widely in the invert. This limits the variability of the dose rate as a function of any other factors in the near-field model. In particular, TRW00a notes that most of the releases are by a diffusive mechanism, hence controlled by diffusion-related parameters. This too appears to be an area in which a strong structural conservatism of the model (in this case the assumed diffusional releases) tend to mask the importance of other effects.
- EBS transport. This barrier represents the function of environmental conditions and diffusion in the drift invert in limiting radionuclide transport out of the EBS. In this case of the robustness analysis, the combined effects of degraded concentration limits and high diffusion cases. The results are reported as a decrease in the time to early-arrival doses (defined as time to 10^{-3} mrem/yr) of several thousand years, and an increase in the peak dose rate of about a factor of 5.
- UZ transport. This barrier represents the function of the UZ below the potential repository in delaying radionuclide transport to the biosphere. An extensive set of robustness analyses were presented for this function. The degraded cases showed between a factor of 5-10 higher dose rates than the base case, whereas the enhanced cases showed significantly improved behavior (many orders of magnitude) over the base case.
- SZ. This barrier represents the function of the SZ in delaying radionuclide transport to the biosphere. The robustness analysis was used to investigate parameters associated with travel time in the saturated zone: sorption, and flow rate. The difference between degraded and enhanced performance in these analyses is between one to two orders of magnitude, with the base case very close to the upper end of this variability.

The TSPA-SR explicitly acknowledges that the results of these analyses are dependent upon the scenarios and conceptual models implemented in the TSPA-SR. They note that the conservatism of parameter values and assumptions may tend to mask the importance of some of these to the results, or may mask the importance of others.

7.3.10.3 The Yucca Mountain Science and Engineering Report

A more in-depth discussion of the basis for the TSPA-SR was recently provided in the Science and Engineering Report (DOE01). This basis was used to update information in the Draft EIS, to reflect the most current information regarding design and safety, and was published recently as a Supplemental EIS (DOE01a). The information presented in these reports is generally a synopsis of information published in prior reports, but the format of the report permits a straightforward comparison between scenarios, models, and parameter assumptions implemented in the TSPA-SR and observations, elicitation, and measurements that support them. In addition, a series of discussions are presented on alternative conceptual models omitted from the TSPA-SR, and the rationale for their omission. These discussions of alternative conceptual models provide useful contrasts with the TSPA-SR models, and the alternatives are briefly summarized here for each component of the TSPA:

- For unsaturated zone flow, alternative conceptual models are discussed for lateral flow in the Ptn unit, fracture flow in the Ptn unit, episodic flow in the Tsw unit, low permeability of faults in the Ptn and Chn/CFu units, and for discrete fracture flow. Each of these conceptual models was argued to be equivalent to, or less conservative than, the conceptual model used in the TSPA-SR. The data needed to resolve between the TSPA-SR conceptual model and these alternatives were argued to be either currently unavailable or impractical to gather.
- Alternative thermal-hydrological conceptual models were stated to be in one of several categories: alternative representation of fractured rock in numerical models, alternative selection of representative property values, and the potential for permanent changes in those properties from the effects of heating. Specific attention is drawn to alternative thermal-hydrologic-chemical and thermal-hydrologic-mechanical approaches. No arguments are made as to the relative conservatism of the alternative approaches and the TSPA-SR model.
- Alternative conceptual models for the physical and chemical environment of the repository were described as alternative concepts for the thermal-hydrologic-chemical seepage model, for the approach to representing precipitates and salts contacting the drip shield and waste package, for evaluating microbial communities in the repository, for interactions of steel and titanium, and for modeling rockfall. It was argued by DOE01 that these alternative approaches would provide results comparable to those in the TSPA-SR.
- Alternative conceptual models for waste-package and drip-shield degradation were discussed for oxidation, localized corrosion thresholds, stress corrosion cracking, stress mitigation, and hydrogen-induced cracking. The TSPA-SR model for stress corrosion cracking is described as conservative compared to alternatives, as it is based on data for

stainless steel, which is more prone to such cracking than Alloy 22. Estimates of the relative conservatism of the other conceptual models are not given by DOE01.

- Alternative conceptual models associated with flow diversion are discussed by DOE01 in the representation of seepage inflow as a function of time and location, in the environment under the drip shield, and in water drainage from the drifts. It is stated that the TSPA-SR model is expected to be similar in average behavior to the alternative concepts for seepage inflow. The environment under the drip shield in the TSPA-SR is conservatively assumed to permit microbially induced corrosion, which may not occur in alternative conceptual models. The alternatives associated with drainage of water from the drifts have not been fully evaluated, but they are argued to be unimportant.
- DOE01 states that alternative conceptual models for waste form degradation and radionuclide release were considered in all aspects of the TSPA-SR model. Specific alternatives are discussed for evaluation of inventory, in-package chemistry, cladding degradation, dissolution of spent fuel (both commercial and DOE), glass degradation, solubility, and colloid generation.
- DOE01 states that alternative conceptual models for transport in the engineered barrier system are improbable or not supported by data. It is argued that the use of a multi-dimensional model would lead to small differences in performance.
- Alternative conceptual models associated with unsaturated-zone transport below the repository are argued to be similar to those for unsaturated zone flow above the repository. In addition, the potential for a drift shadow of drier conditions is mentioned, which would be less conservative than the TSPA-SR model. In addition, a discussion is provided of matrix diffusion. It is argued that neglect of matrix diffusion is not realistic, and that such a model should not be used.
- For the saturated-zone system, DOE01 identifies four key areas of uncertainty for which alternative models are possible: treatment as a porous medium or if discrete features need to be included, behavior in the vicinity of large hydraulic gradient, recharge and the time scale over which it functions, and scale dependence of data and model parameters. No arguments are made by DOE01 as to the relative conservatism of the TSPA-SR model and alternatives in this area.

In addition to the discussion of alternative conceptual models for the nominal scenario, DOE01 provided in-depth discussions of alternative scenarios that might be considered in a TSPA, but which have been discarded based on either probability of occurrence, reasonableness, or lack of consequences. Scenarios associated with water table rise and nuclear criticality are discussed in detail, with extensive presentations of the determination that neither is credible.

7.3.11 Uncertainties in Projecting Repository Performance Over Very Long Time Periods

Repository performance assessments, such as those for a potential repository at Yucca Mountain discussed in Sections 7.3.1 through 7.3.9, are the means to assess potential doses to individuals as a result of radionuclide releases from the repository. Potential doses are, in turn, the major factor in regulatory compliance evaluations.

A 10,000-year time horizon for repository dose projections is well established in regulatory approaches in the United States (40 CFR Part 191, Part 194, and Part 197; 10 CFR Part 60 and draft Part 63) and in other nations (GAO94). Performance projections have a fundamental role in implementing these regulations and in regulatory decision making, and the reliability of performance projections over such time frames for estimating potential doses to exposed individuals and groups is inextricably tied to geologic stability issues for any particular repository site. This section discusses uncertainties over the time period for assessing the performance of a repository at Yucca Mountain, and the effect of the uncertainties on regulatory decision making.

As discussed in Sections 7.3.1 through 7.3.9, a series of Total System Performance Assessments (TSPA) has been reported by DOE during the past decade, and additional TSPA evaluations will be done for the License Application if the Yucca Mountain site is recommended to be suitable for disposal (Section 7.3.10). Some TSPA evaluations have been performed for time periods up to one million years because some of the radionuclides that could be released from the repository to the environment have half lives that would enable potential doses to humans to occur for such time periods (e.g., Np-237 and Pu-242).

Results of the TSPA evaluations show that potential doses increase over long time frames, i.e., beyond 10,000 years. The TSPA-SR analyses (TRW00b) showed, for example, that, for the EDA II repository design and TSPA-SR models, peak doses would continue to increase over 100,000 years after disposal, but would not reach very high values even at that time.

In recognition of this type of performance profile, the National Academy of Sciences Committee that developed the report, *Technical Bases for Yucca Mountain Standards* (NAS95), recommended that compliance with the individual-protection standard be evaluated at the time of peak dose. The Committee's report states, at page 72:

One commonly expressed concern regarding the performance assessment modeling is that it requires simulating performance at such distant times in the

future that no confidence can be placed in the results. Of course, the level of confidence for some predictions might decrease with time. This argument has been used to support the concept of a 10,000 year cutoff. We do not believe, however, that there is a scientific basis for limiting the analysis this way.

*...We recommend calculation of the maximum risks of radiation releases whenever they occur **as long as the geologic characteristics of the repository environment do not change significantly**. The time scale for long-term geologic processes at Yucca Mountain is on the order of approximately one million years. After the geologic environment has changed, of course, the scientific basis for performance assessment is substantially eroded and little useful information can be developed. (Emphasis added)*

This statement leaves open the question of what changes in the geologic environment are “significant” to prediction of repository system performance. It also implicitly presumes that changes in the engineered barrier system that affect the source term for radionuclide releases to the environment do not occur, or occur in a predictable fashion, and are independent of any changes in the geohydrologic regime important to transport of radionuclides from the repository to and through the environment. The potential for independent and interactive changes in both the natural and engineered features of the repository system that can affect reliability and uncertainty for long-term predictions of repository system performance is discussed below.

Effects of Natural Processes and Events on Long-Term Performance Predictions

Processes and events that affect characteristics of the geohydrologic regime directly involved in radionuclide release and transport would be important to the reliability of long-term projections of repository performance. The directly-involved geohydrologic characteristics are those of the unsaturated zone above and beneath the repository and those of the saturated zone along the radionuclide transport path to the point of evaluation of compliance with the regulatory standards. At issue, for reliable long-term projections of performance and compliance, is the potential for the long-term geologic processes to produce changes in flow/transport path features, such as fracture network transmissivity and faults, which alter the basis for characterizing performance of natural system features in the repository performance models.

As noted by the NAS Committee, and discussed in detail in Section 7.1.1.8, available data suggest that the time scale for long-term geologic processes at Yucca Mountain is on the order of one million years, and that the characteristics of the processes and events, such as the frequency and magnitude of earthquakes, can be bounded. This concept addresses, however, only the macro-scale aspects of the occurrence and effects of geologic processes and events, e.g., the

frequency and magnitude of earthquakes. Of importance to the reliability of long-term performance predictions is the potential for the events and processes to produce local, small-scale impacts important to radionuclide transport, e.g., to open or close fractures and faults along the flow path that transports radionuclides released from the repository, such that ground water travel times and radionuclide interactions with the host rocks are significantly altered.

Repository system performance models based on site characterization data will not identify specific small-scale flow/transport features, but the values for parameters that represent those features in the models may be altered in uncertain ways in the long term as a result of the geologic processes and events. For example, ground water flow rates through fracture/fault pathways indicated by bomb-pulse Cl-36 data and other site characterization data that are important to repository system performance may be altered by seismic or volcanic events and changes in climate. If the fracture network along radionuclide transport paths is loosened (more fractures are created, fracture apertures increase, or fractures become more interconnected) by the accumulated effects of seismic or igneous activity, the simplest application of Darcy flow would result in decreased ground water travel times from the repository to down-gradient receptors. If the hydraulic gradient along the travel path remains the same over the time period considered, the predicted performance of the repository would be improved.

Similarly, if translation of geologic formations occurs differentially under tectonic movement (overall, the site will translate on the surface of the earth a distance of over one mile during the next million years, but the various geologic layers might translate at different rates), the flow/transport paths in the volcanic rocks and the highly heterogeneous valley-fill alluvium may be significantly, but unpredictably, altered. The evolving stress state in the fractured rocks as affected by the regional tectonic framework could cause the hydrologic behavior of the fractured rocks to change in unpredictable ways – either by opening fracture networks, as described above, or perhaps by tightening the network and thereby increasing flow rates under the same gradient. However, the tectonic movement over the long term could also change the gradients in the flow system around the repository. The net effect of gradually changing hydrologic characteristics could be to increase ground water travel times, to decrease them, or to leave them essentially the same as was the basis for the performance assessment models.

The potential for such performance-relevant changes would introduce significant uncertainty in long-term performance predictions, so that the results of the predictions would not be a reliable measure of compliance with regulatory standards or of dose potential in general. To assume that the TSPA models are appropriate and applicable without modification for predictions over the next million years is to assume that the various natural processes and events have no significant

effect on any aspects of the natural features of the site and repository system that are important to ground water flow and radionuclide transport. It would also assume that bounding estimates are a sufficient basis for evaluation of dose potential and compliance with regulatory standards.

The effect of seismic activity on repository system performance was investigated by DOE as part of the Viability Assessment (DOE98, Volume 3, Section 4.4.3). The evaluations considered the potential for, and consequences of, rockfalls onto waste packages, changes in hydrologic properties, potential for water table rise, and indirect effects such as alteration of ground water flow and transport paths from fault alteration near the repository site.

The analyses determined that the probability of rockfall causing a waste package to split open over 10,000 years is essentially zero because the package walls are essentially uncorroded and able to withstand impacts from virtually any size rock. The analyses of indirect effects recognized that changes could occur in faults and flow paths that could either impede or enhance ground water flow and radionuclide transport, but concluded, on the basis of available site characterization data, that radionuclide concentration profiles would be affected only if several faults occur.

The analyses focused on potential for performance impacts of seismic activity during the first 10,000 years after disposal, and determined that they would be negligible for this time period, a finding with which the TSPA Peer Review Panel concurred (PRP99, p. 8). The analyses also implied, however, that the long-term effects of seismic activity could be substantial. Rockfalls would contribute to waste package failures after about 500,000 years, and changes in faults would accrue over long time periods involving seismic events. Seismic activity could therefore alter the long-term engineered and natural features of the repository system in ways that are uncertain and would affect the reliability of performance assessment results for the long term.

Effects of Engineered Barrier System (EBS) Changes on Long-Term Performance Predictions

The engineered features of the proposed repository at Yucca Mountain consist basically of an array of waste packages and supporting structures emplaced in drifts excavated at a depth of about 300 m beneath the surface of the mountain. During the first 100 years after disposal, the surface temperatures of the waste packages will rise to about 150-170°C as a result of heat emissions from decay of relatively short-lived radionuclides in the wastes such as Cs-137 and Sr-90. At about 10,000 years, the waste package surface temperatures will have decreased to about 50°C, and at 100,000 years the temperatures of the waste package and repository will have returned to the pre-emplacment ambient temperature of about 28°C (DOE98).

The temperature of the geohydrologic regime surrounding the drifts will also undergo a thermal pulse in response to that associated with the waste packages. Characteristics of the thermal pulse within the near-field geohydrologic regime, and the effects of that pulse on the regime, will depend on the physical properties of the regime and the waste package emplacement configuration.

The thermal pulse described above can affect many of the performance factors important to the safety performance of the repository. Within the engineered features, for example, it will affect the corrosion rates of waste package materials as a function of time. It can also affect the properties of the near-field geohydrologic regime around and near the drifts so that, for example, the rate and location of water seepage into the drifts is altered from initial conditions (see, for example, PRP99).

The thermal pulse, and external factors such as climate change and seismic activity, can therefore affect the degradation and configuration of the EBS system and the near-field geohydrologic regime so that, over long periods of time, the system differs significantly from that which was initially established and for which the TSPA models and parameter uncertainties were established. As detailed below, these degradation and configuration changes introduce uncertainties to long-term assessments of repository system performance and add “uncertainty to the uncertainties” in performance parameters.

An important consideration with respect to increase in uncertainties with time is the fact that evaluation of critical parameters affecting engineered barrier performance, such as corrosion rates, is done in relatively short term data acquisition programs, typically on the order of a few years at most, and often over much shorter time spans. The results of such measurements are then used in performance assessments that require extrapolation of the short term data to time scales of hundreds to thousands, and tens of thousands, of years. There is an inherent uncertainty about the reliability of these extrapolations that must be accepted, since measurements for repository time scales are not possible. At best, confirmation of materials performance over time is limited to time spans of decades, for materials that have been in use for such periods.

The inherent uncertainty is especially important with respect to the long-term stability of the surface films that produce the corrosion resistance of materials such as Alloy 22, and for the non-uniform corrosion mechanisms induced by external conditions such as applied stresses (e.g., stress corrosion cracking). The gradual alteration of physical and chemical conditions in the repository over the very long term (tens of thousands of years) adds additional uncertainty to the

reliability of extrapolating short-term data. Over such long time periods, uncertainties relative to materials performance arise that cannot be addressed with laboratory measurements. For example, will the stresses introduced in waste package materials as a result of prolonged heat emissions from the wastes increase the material's susceptibility to non-uniform corrosion mechanisms, or increase the general corrosion rates as a result of film breakdown?

The inherent uncertainties in EBS performance, and potential for change in the uncertainties, are compounded by uncertainties in the performance of the natural system to the extent that these systems interact. For example, uncertainties in projections of contact of waste packages by ground water that seeps into the drifts affects estimates of waste package degradation and subsequent radionuclide releases to the natural barrier system.

Discussion of Uncertainties by the TSPA Peer Review Panel

The Final Report of the TSPA Peer Review Panel (PRP99) provides a comprehensive inventory and discussion of the changes in the repository system that can occur over the long term. The Panel report notes (p. 33) that unknown changes in factors such as climate, and locations of people and their sources of food and water, will occur over time periods of one million years. The report also notes that the time periods are long in comparison with the time available for testing the corrosion rates of materials, thereby making the extrapolation of materials performance uncertain.

Changes and uncertainties in other performance factors discussed by the Panel include:

- Changes in seepage rates and locations. The seepage rate has been identified by DOE as one of the most important repository performance factors. Seepage rates and locations can be changed as a result of climate change, alterations in the near-field geohydrologic regime as a result of the thermal pulse, and drift collapse as a result of coupled thermal/mechanical effects and seismic activity.
- Changes in fracture permeability. The ability of fractures in the unsaturated zone to transport water to and from the repository can be affected, in some cases irreversibly, by changes in the near-field geohydrologic regime resulting from the thermal pulse or seismicity. Such changes can occur with indeterminate probability and with good or bad consequences with respect to seepage into the drifts and movement of water from the repository to the underlying saturated zone.
- Formation of a precipitate cap. Movement of water in geologic formations above the repository in response to the thermal pulse may result in chemical changes

which produce a precipitate cap that affects flow of water to the repository. An analysis cited in PRP99 determined that a precipitate cap could persist for 900,000 years, could have a profound effect on repository performance, and is of indeterminate probability of occurrence.

- Changes in the chemical composition of dissolved species in water that seeps into the repository. The thermal pulse can lead to alterations in the composition of species dissolved in water that enters the repository and thereby alter its capacity to corrode waste package materials. The rate of dissolution of the waste form, the solubility of released radionuclides, and the sorptive capacity of geologic formations beneath the repository can also be changed by changes in dissolved species.
- Changes in configurations and temperature fields in the engineered features. Over long periods of time, corrosion will degrade the materials in the waste package and alter their configurations, e.g., produce collapse of package walls and package contents. Debris from rockfalls may blanket the waste packages and cause temperature elevations that result in increased corrosion rates for spent fuel cladding. After waste package degradation by corrosion has become significant, rockfalls may crush the packages, alter seepage flow fields, and alter temperature levels and fields.
- Changes in the near-field geochemical environment (NFGE). The thermal pulse, coupling effects, and climate change can produce changes in the NFGE that have a major impact on repository performance. PRP99 notes that a significant improvement in the data base for the NFGE is needed in comparison with that which was available for the VA, and that, even if significant improvement is obtained, there may still be large uncertainties relevant to effects on performance so that a bounding analysis would be needed.

These descriptions of changes that can occur and affect the reliability of long-term performance predictions indicate that the factors involved in producing changes are highly interactive and of uncertain consequence. The TSPA models used to evaluate repository performance attempt to reflect and characterize the potential for system changes (e.g., by estimating changes in repository temperatures and corrosion rates of waste package materials over long periods of time, and by estimating the frequency, size, and consequences of rockfalls), but the reliability of results of such analyses, especially in terms of reliability as a basis for evaluation of compliance with regulatory standards, is compromised by the uncertainties associated with the model assumptions, the complexity of the phenomena and processes involved, and the potential for interactive effects with uncertain impacts. The combined effects of these uncertainties is difficult, and probably impossible, to predict and characterize with a high degree of confidence.

These uncertainties in changes, performance, modeling results, and compliance with standards are all associated with the long term, i.e., periods on the order of a hundred thousand years and beyond. The uncertainties for shorter time periods, on the order of tens of thousands of years, can be characterized and bounded more reliably because many of the causes for change that could invalidate the predictive models and their results cannot occur or will not have had time to take effect.

Yucca Mountain Repository Design Features to Mitigate Uncertainties

As discussed in Section 7.3.9, repository design features described in Revision 3 of the Repository Safety Strategy (TRW00) were selected to mitigate uncertainties associated with the VA design and are expected to prevent any penetration of the waste packages and release of radionuclides for periods on the order of 100,000 years. The design would use waste package outer walls of Alloy-22, a highly-corrosion-resistant nickel-based alloy, and the waste packages would be protected from contact by seepage water through use of titanium drip shields. During the first tens of thousands of years after disposal, the mechanical strength of the waste packages would be such that rock falls could not deform or crush the packages. In addition, under the proposed design the drift spacing and temperature limits would be established so that the effects of the thermal pulse would be diminished and restricted to small regions around each drift, without drift-to-drift interactions.

Design features such as use of highly-corrosion-resistant waste package wall materials and restricting the potential for the thermal pulse to alter the geologic, hydrologic, and chemical regimes near the repository therefore can, and are planned to, produce high confidence that the repository configuration will be maintained and radionuclide releases will be prevented for tens of thousands of years. In the long term, on the order of hundreds of thousands of years, degradation of the engineered features of the repository will occur, but the temperature levels and gradients that would drive radionuclide release and transport will greatly diminish. Peak radionuclide release rates may therefore not be achieved during one million years after disposal, and the levels of radionuclide release and potential radiation doses that are achieved may be highly restricted.

Revision 4 (Rev 4) of the Repository Safety Strategy, discussed in Section 7.3.10, is intended in part to improve understanding and characterization of uncertainties in performance assessments for the proposed repository for Yucca Mountain. As shown in Table 7-17, Rev 4 will identify principal performance factors on the basis of extensive TSPA analyses and barrier importance analyses. Performance assessments will use updated, fully-documented models, and will use

probabilistic analyses to address uncertainties. Features, events and processes that have potential to disrupt repository performance from expected conditions will be fully documented and analyzed. Through use of this approach, DOE expects to provide a readily-understood characterization of expected repository performance, uncertainties relevant to performance, and the effect of the uncertainties on performance.

Relationships Between the Regulatory Time Period, Performance Predictions, and Uncertainties

As discussed above, changes in the natural and engineered features of the repository are expected over long periods of time. The TSPA models attempt to anticipate and describe these changes and their effects on radionuclide releases, and the types of factors that can produce changes in repository conditions, such as seismicity and climate change, can be anticipated.

Details and interactions of factors that affect long-term radionuclide release from the repository cannot, however, be described and anticipated with a high degree of confidence. Long term uncertainties, and uncertainties about the uncertainties, are therefore embedded in the results of long-term performance predictions. The predictions of long-term radionuclide releases and radiation doses can be made using bounding estimates for the effects of factors that change repository performance, but such estimates necessarily will embody the long-term uncertainties and may unrealistically overstate the radiation dose potential because of conservatism associated with bounding estimates.

Boak and Dockery (GSA98) discuss at length the validity of long-term projections of geologic systems for policy decisions. They demonstrate that the scientific community can usefully project the long-range future behavior of geologic systems, and that such projections are necessary as a basis for well-reasoned assessments of human interaction with such systems, i.e., as a basis for policy decisions involving political, scientific and social requirements that frame the adequacy of predictive models and their results.

Boak and Dockery focus on the need to adequately validate models and the results of their use, and note that validation "...has greatest significance where scientific data and conclusions are brought to bear on difficult legal and regulatory decisions". They discuss three methods for validating models in detail: expert judgment, conservatism, and stochastic simulation. Expert judgment can assign values to parameters when data are sparse and can increase confidence that relevant features of the system have not been overlooked. Conservatism can be exercised in selection of parameter values and by taking no credit for performance of some subsystems (the authors note that "...the use of conservatism causes one to simulate unrealistic values to increase

confidence in the result, thereby invalidating the model in order to validate the result”). Stochastic simulation can examine the effect of uncertainty in system descriptions on prediction results, explore the sensitivity of performance predictions to uncertainty, and assess alternative scenarios and process models.

Models, and results of their use, can also be validated through use of external review by scientific and stakeholder communities, and by comprehensive documentation which enables others to duplicate model results and to examine the details of the model and possible alternatives. These validation methods are complementary to the expert judgment, conservatism, and stochastic simulation techniques which aid development and use of the models.

DOE has used all five methods to develop and implement performance assessment models for a repository at Yucca Mountain, and to present and analyze performance assessment results. The issue at hand is, When is validation of the models and assessment results for a repository at Yucca Mountain adequate for licensing reviews? More specifically, for present discussion, For what regulatory time frame do the uncertainties in performance results overwhelm confidence in assessment of compliance with regulatory standards?

The choice of a regulatory time period must balance two considerations: the degree of confidence possible for the long-term projections of repository performance and radiation doses, and the nature of the licensing process, which requires a reasonable consensus, by the parties involved, concerning the validity of repository performance projections. For a successful licensing process, a reasonable consensus must be developed that the performance projections are valid for evaluating compliance with standards with a reasonable degree of confidence. If uncertainties in assessment results are too high, the needed consensus is unlikely to be attained, and the licensing process is likely to fail under the weight of these uncertainties.

In examining the question of regulatory time periods, the NAS Committee (NAS95) stated,

*“The current EPA standard contains a time limit of 10,000 years for the purpose of assessing compliance. **We find there is no scientific basis for limiting the time period** of an individual-risk standard in this way. We believe that compliance assessment is feasible for most physical and geologic aspects of repository performance on the time scale that is on the order of 10^6 years at Yucca Mountain...we recommend that compliance assessment be conducted for the time the greatest risk occurs within the limits imposed by long-term stability of the geologic environment”.* (NAS95, pp.6-7; emphasis added)

Since dose is proportional to risk, this recommendation corresponds to recommendation that compliance with regulatory standards be evaluated at the time of peak dose. As discussed in Section 7.3.9, the time of peak dose for the EDA II design, which is the basis for the Site Recommendation TSPA evaluations, has been preliminarily estimated to be 650,000 years. Refinement of the models and analyses (Section 7.3.10) may demonstrate that peak dose is not attained for more than one million years, i.e., "...within the limits imposed by long-term stability of the geologic environment".

Establishing the regulatory time period to correspond to the time of peak dose presents two fundamental difficulties. First, given the long-term uncertainties in repository system performance over time periods of hundreds of thousands of years, as discussed above, there would be considerable variation in the estimates of time to peak dose, and the peak dose level, within the range of possible assumptions corresponding to the range of uncertainties. The degree of confidence that could be assigned to one set of assumptions in comparison with others is likely to be minimal; consequently, making use of such calculations as the basis for compliance evaluation would be difficult or impossible. Second, as a result of the inherent uncertainties in long-term performance projections, achieving a technical consensus concerning the validity of results would be extremely difficult, especially in adversarial situations, since repository performance projections require extrapolation of data and processes over unprecedented time frames, with little or no means available to verify the parameter values and assumptions.

While it may be scientifically possible to make long-term performance projections, as suggested by the NAS Committee, and these projections can serve the purpose of helping to estimate the long-term performance of any repository, a policy decision, rather than a scientific decision, is needed to establish the compliance time period. The policy decision should take into consideration the need to protect the public from unacceptable exposures consistent with societal expectations, and also take into consideration the need for the licensing process to establish technical consensus among its participants and confidence in its results among its stakeholders.

The 10,000-year period encompasses a time span sufficient for the expected variations in factors important to repository performance to be characterized and modeled without moving too far into the realm of unverifiable expectation. As the time frame extends into hundreds of thousands of years, the credibility and validity of extrapolating current conditions becomes progressively more questionable, and the results of evaluations become less acceptable as a basis for compliance evaluation.

As previously noted, the 10,000-year regulatory time frame has been incorporated into the regulatory framework of the United States and other nations (GAO94), thereby reflecting a broad consensus on societal expectations for viable assessments of protection. Performance projections over longer time frames are useful to support a compliance evaluation and possibly to aid design selection, and they should be performed for any repository subjected to licensing reviews. However, use of these evaluations as the basis for compliance evaluation is technically difficult to justify and potentially an unnecessary cause for contention in a licensing process.

Evaluation of compliance with EPA's radiation protection standards for a repository at Yucca Mountain at the time of long-term peak dose is inappropriate for three principal reasons: (1) results of long-term TSPA evaluations are highly uncertain; (2) use of bounding estimates and conservative values of performance parameters may, with inherent high uncertainty, significantly overstate the radiation dose potential; and (3) under current repository design concepts, peak dose potential may not be achieved within the time period for which bounding estimates can be made.

A compliance period of 10,000 years encompasses a time span sufficient for factors important to repository performance to be characterized and modeled, and also enables confidence in the characterization of uncertainties in performance parameters and results.

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