CHAPTER 9

YUCCA MOUNTAIN EXPOSURE SCENARIOS AND COMPLIANCE ASSESSMENT ISSUES

9.1 INTRODUCTION

Chapter 7 described the proposed Yucca Mountain repository in terms of its site characteristics, engineering designs, and current performance assessments. In Chapter 8, derived unit-concentration exposure and risk estimates were presented. These estimates were based on a conceptual model in which radionuclides were released into the biosphere from a repository through the following sequence: degradation and failure of the waste canister(s) through corrosion; release of radionuclides from the waste package into host rock; migration of radionuclides through the unsaturated zone into the aquifer (saturated zone); and dissemination of contaminated ground water to wells used for drinking and agricultural purposes.

This scenario, involving a gradual release from an undisturbed repository, characterizes the most probable events and conditions of future human exposure. It also conforms with the primary objective of deep geological disposal which is to provide long-term barriers that isolate wastes and limit the release of radionuclides into the biosphere by virtue of siting and engineering design. Deep geologic disposal isolates the wastes for a sufficiently long period of time to allow most of the radionuclides to decay to natural background levels. While estimates of dose and risk for this gradual release process cannot be calculated with complete precision, there is a substantial scientific basis for modeling the various processes that take into account parameter variabilities. By means of statistical processes, such as the Monte Carlo method (see Section 8.5), these uncertainties can be minimized, thereby yielding dose/risk estimates that are reasonable.

Figure 9-1 illustrates the major release pathway leading to human exposure which involves ground water from an undisturbed repository at Yucca Mountain. The major reservoirs (source terms) containing radionuclides at various times following closure are depicted as rectangles. These reservoirs do not have discrete physical boundaries, but rather form a continuum. Solid arrows between reservoirs represent the probable processes by which radionuclides are transported from one reservoir to another in an undisturbed repository.

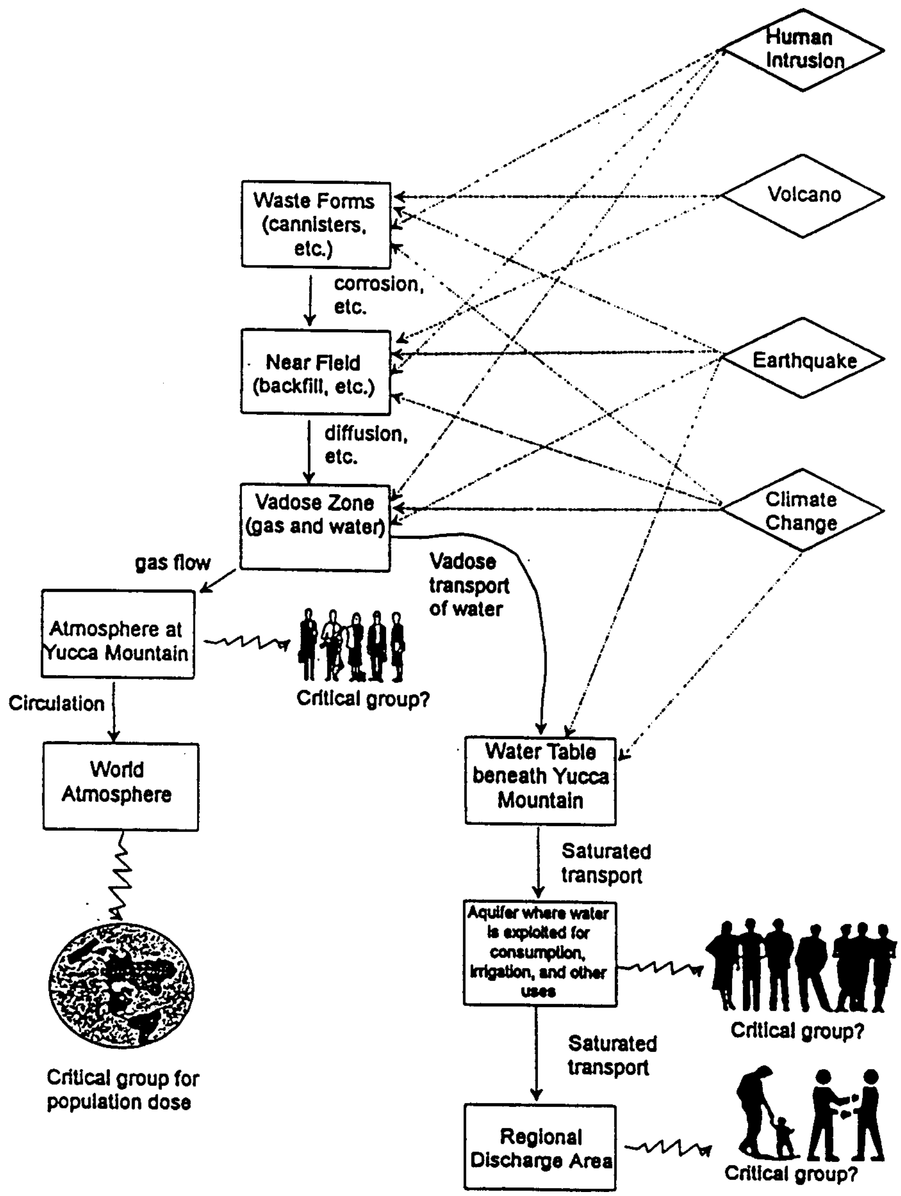

Figure 9-1. Schematic Illustration of the Major Pathways from a Repository at Yucca Mountain to Humans (NAS95)

Major processes and events[27] with potential to modify normal behavior or drastically alter the physical integrity of reservoirs are shown in Figure 9-1 as diamonds. These modifiers are connected by dashed lines to those reservoirs upon which they are likely to have the most significant impact.

To ensure maximum public protection, a standard for a repository at Yucca Mountain must also consider: (1) release pathways other than ground water and (2) improbable conditions that may lead to individual doses and risks well in excess of those specified for the undisturbed repository. A demonstration of compliance with such a standard, therefore, requires estimating potential doses resulting from secondary release pathways and predicting improbable events and processes that may disturb the repository and their corresponding outcomes.

This chapter summarizes issues involved in developing a repository standard that addresses the important exposure pathways and related performance issues that have been identified.

9.2 GASEOUS RELEASES: A SECONDARY PATHWAY FOR HUMAN EXPOSURE

The primary pathway of radionuclide releases from an undisturbed repository involves the introduction of radionuclides into the underlying aquifer, contaminating wells used for drinking or agricultural irrigation. However, as shown in Figure 9-1, humans could also be exposed to radiation as a result of gaseous emissions from the repository. Due to the ease with which gaseous contaminants are distributed in the atmosphere, human exposure would not be limited to the nearfield population but could extend to the world at large. The radionuclide with the highest potential for gaseous release and human exposure is carbon-14.

This section provides a brief overview of the primary parameters affecting the timing and magnitude of gaseous releases and assesses bounding values for human doses.

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[27] The difference between an <u>event</u> and a <u>process</u> is the time interval over which the phenomenon occurs relative to the time frame of interest; events occur over relatively short time intervals and processes occur over relatively long time periods. For example, a disruptive seismic event may occur over minutes, hours, or days. Even a volcanic eruptive cycle that may have time frames extending over several years must be considered an event when judged in context with the lifespan of humans and/or a million-year repository assessment period. Phenomena that exceed human life expectancy or occur over a significant portion of the period of regulatory concern are considered to be processes.

9.2.1 Production and Early Containment of Carbon-14

Carbon-14 is produced in nuclear fuel as a result of neutron absorption by the following reactions: (1) N-14 (n, p) C-14 and (2) O-17 (n, He-4) C-14 (DAV79). Thus, the quantity of C-14 produced is governed by the amount of nitrogen and oxygen contained within the fuel core.

Carbon-14 in oxide fuels is assumed to exist as either CO_2 or low molecular weight hydrocarbons that in time are oxidized to CO_2. The total inventory of C-14 for the 63,000 tons of spent nuclear fuel is estimated to be about 91,000 Ci (DOE94).

When estimating gaseous releases from the repository through the unsaturated zone to the accessible environment, the following parameters should be considered: (1) container performance and (2) the bulk permeability and retardation capability of the tuffs. The retardation of gaseous CO_2 flow is due to its exchange with the relatively immobile bicarbonate (HCO_3^-) in the pore water of the unsaturated zone.

9.2.2 Impacts of Thermal Loading on Gaseous Releases and Transport

The emplacement configuration of heat-generating waste containers is likely to disturb the ambient environment of the repository in a number of ways. Waste-generated heat is expected to enhance vaporization of water within the tuff matrix and, at temperatures above 96°C, completely "dry out" the adjacent host rock and move the water into the surrounding rock.

The impact of thermally-displaced water in the vicinity of the repository has a dual effect on gaseous releases. Since most of the waste-container corrosion processes are known to be temperature and moisture dependent inclusive of: (1) general aqueous corrosion, (2) steam corrosion, (3) pitting corrosion, (4) dry oxidation corrosion, and (5) stress corrosion, the potential impacts of waste emplacement and thermal loading on container failure are highly critical for modeling the time and release fraction of gaseous C-14.

Estimates of travel time for C-14 released from a container into the unsaturated zone are strongly affected by the moisture content. Under conditions of 100 percent humidity, C-14 is assumed to exist for the majority of the time as bicarbonate (HCO_3^-) in the slow-moving aqueous phase (ROS93). Conversely, within the dry-out zone, C-14 can be assumed to exist almost exclusively in the fast-moving gaseous form (CO_2).

9.2.3 Estimates of Travel Time

Estimates of travel time for C-14 released from a failed container to the accessible environment are complicated by the fact that the radionuclide is likely to exist only a small portion of the time in gaseous form as $^{14}CO_2$; the majority of the time, it will exist as bicarbonate (HCO_3^-) in the aqueous phase. In the bicarbonate form, C-14 moves more slowly than in the fast moving uncondensible gaseous form (ROS93). This "slowing," or retardation, must be incorporated into the travel-time calculations by dividing the short-lived gas velocity at each point along the flow path by a retardation factor that accounts for the longer time and limited movement of C-14 in the aqueous bicarbonate phase. Travel-time probability distributions can be determined by coupled calculations of gas and heat flow (i.e., time-dependent temperature distributions in the repository environs).

Because estimates of early waste-container failure and release of C-14 are currently highly uncertain, travel times for release of C-14 have been estimated at 1,000-year intervals following waste emplacement. For example, Figures 9-2 and 9-3 show travel-time histograms for a thermal loading of 57 kW/acre and welded-tuff bulk permeability of 10^{-11} m^2 at 1,000 and 10,000 years (DOE94):

- At 1,000 years, temperature gradients in the vicinity of the repository are high due to the large heat output. Correspondingly, gas velocities in the nearfield ("dry-out zone") are larger than in the far field. Calculated C-14 travel times range from 200 to 600 years.

- At 10,000 years, heat has been conducted outward and temperature gradients have been reduced, resulting in estimated travel times that range from 500 to 1,200 years.

Given this, it appears that the magnitude of potential atmospheric releases would be greatest if containment failure were to occur early. At early times (i.e., 1,000 years), transport velocities can be expected to be maximal and the reduction of C-14 by natural decay is minimal.

It is reasonable, however, to expect that individual container failures will occur over a long period of time. This could substantially broaden the range of C-14 travel times in the unsaturated zone from as little as 200 years to as much as 1,800 years. The period of C-14 release into the accessible environment is further delayed by the fact that, at time of canister failure, only a small percentage of the C-14 inventory has leaked from the fuel matrix into the void spaces of the container for instantaneous release. Barnard et al. (BAR92) estimate that this quick release

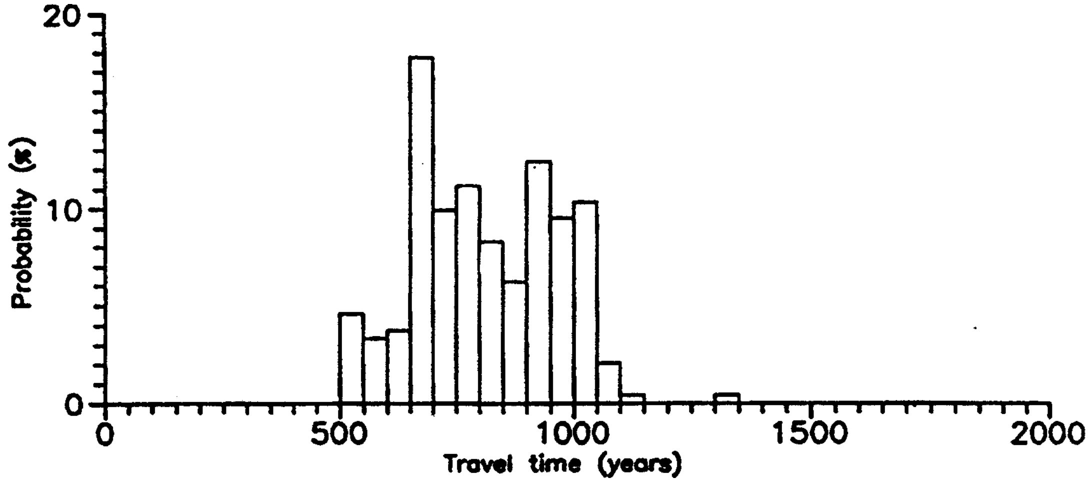

Figure 9-2. Retarded Travel Times of C-14 from the Repository to the Atmosphere for Particles Released at 1,000 Years (DOE94)

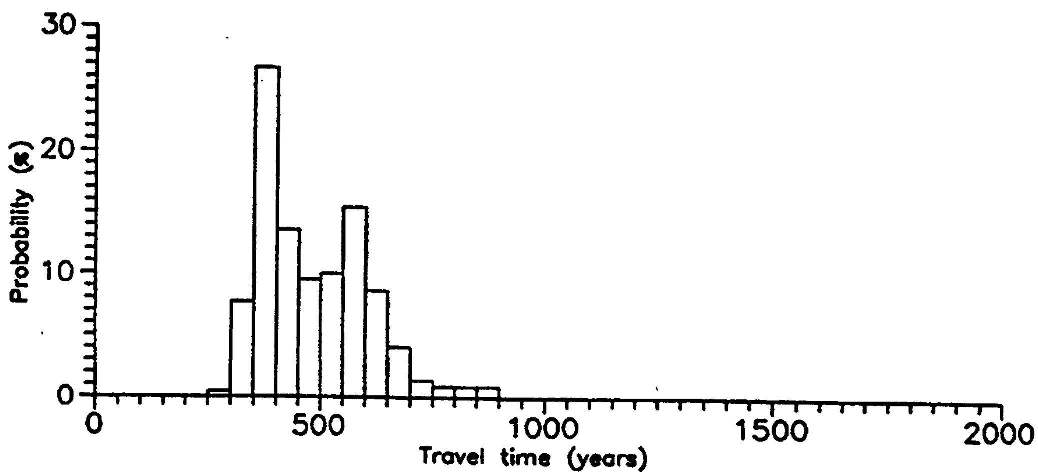

Figure 9-3. Retarded Travel Time of C-14 Particles from the Repository to the Atmosphere for Particles Released at 10,000 Years (Welded-tuff bulk permeability of 10^{-11} m^2 (DOE94))

fraction is likely to represent between 1.25 and 5.75 percent of the total inventory. The slow release of the larger, remaining fraction of C-14 from the fuel matrix to the container and subsequently into the repository environs is likely to further extend the time period during which C-14 can be expected to enter the accessible environment. With a physical half-life of 5,730 years, it is reasonable to conclude that the quantity of C-14 that will reach the accessible environment will be less than the 91,000 Ci inventory existing at the time of waste emplacement.

9.2.4 Dose Modeling and Exposure Estimates

For practical reasons, estimates of human exposure to C-14 assume that the specific activity of C-14 in the atmosphere in gaseous CO_2 form is equal to that of organically bound carbon contained in all plant and animal products that may be ingested as food. Thus, pathways for internal exposure may involve inhalation and ingestion.

For all practical purposes, under a steady-state distribution of C-14 in the environment, the inhalation of $^{14}CO_2$ contributes insignificantly when compared to the ingestion pathway and may, therefore, be eliminated from dose consideration.

Upon the ingestion of organically bound C-14, the uptake, retention, and excretion by the body involve numerous pathways that correspond to biologic half-times ranging from less than one hour to several years. Even for a specific category of organic molecules such as proteins, turnover times are highly variable. While structural proteins show relatively long turnover times, other proteins such as enzymes, plasma albumin, and hemoglobin have relatively short turnover times. When all protein compartments are considered, the half-time of carbon (and therefore C-14) is estimated to be 119 days (NCR93). For fats, which are largely stored in the body as adipose tissue, the biological half-time of carbon is estimated to be 99 days; for carbohydrates, the half-time is estimated to be one day. For a daily dietary intake of 300 grams of carbon, a weighted biologic half-time of about 39 days is obtained. For dosimetric purposes, the ICRP has suggested a biological half-time of 40 days for C-14 (ICR82).

For steady-state environmental conditions, estimates of individual organ and whole body doses from ingestion have been derived by Killough and Rohwer (KIL78). Their model assumes that the specific activity of C-14 (i.e., pCi/g carbon) in the human body will, in time, be the same as that observed in environmental media, inclusive of all plant and animal food products. Correspondingly, the model takes into account the carbon content of individual tissues and organs that will be subject to the beta-ray exposure of C-14. At the present specific activity of C-14 in the atmosphere of seven pCi/g carbon, C-14 is estimated to contribute an annual dose of about 1.5 mrem to humans throughout the world.

9.2.5 Dose Estimates from Repository Releases

Global Doses. Any gaseous release of C-14 from the proposed Yucca Mountain repository will disperse itself globally and, therefore, lead to relatively constant exposures among individuals within the world community. The global distribution model yields a population dose estimate of 399 person-rem per curie of C-14 for a world population of 12.2 billion over a 10,000-year period (EPA96). Using this dose-conversion factor and assuming that the entire repository inventory of 91,000 Ci of C-14 is released, an average individual dose of about 0.0003 mrem/yr is estimated. If this release is scaled down to the C-14 release from the 18 failed packages used in the TSPA-VA analyses (18 packages over 10,000 years), where each package has an inventory of 11.7 Ci, the global dose estimates become extremely small, particularly in comparison with the above estimate for doses from atmospheric C-14.

Local Doses. Estimating potential doses from gaseous releases from the repository is a very complicated assessment, particularily since any potential doses received would be strongly influenced by wind direction and population distributions. In addition, estimating the amount of C-14 that would be released at the ground surface in gaseous form rather than contained as biocarbonate ion dissolved in water is also difficult. Some insight relative to the potential magnitude of doses through the gaseous release pathway can be gleaned from looking at dose estimates from gaseous C-14 emissions from a nuclear power plant. In both the repository and nuclear power plant situations, gaseous C-14 is released into the atmosphere and dispersed downwind of the source.

Doses Within 50 Miles of a Nuclear Power Plant. In another study, air concentrations of $^{14}CO_2$ were modeled out to 50 miles for the Dresden Nuclear Power facility (NCR93). A standard diffusion model, as defined above, and local meteorological data were used to calculate concentrations for all sectors out to 50 miles from the plant. Figure 9-4 identifies isolines of the average annual $^{14}CO_2$ air concentrations.

The numbers not in parentheses in Figure 9-4, multiplied by the continuous source activity per second (e.g., Ci/s), yield the predicted ^{14}C concentrations in Ci/m^3. However, to estimate uptake by growing vegetation, the concentration of ^{14}C should be given as a ratio to stable carbon. To make the example more relevant, a continuous emission rate of 100 Ci/y of ^{14}C was used as the source strength. With the additional assumptions listed on the figure, the isolines are also labelled, in parentheses, in units of specific activity (pCi ^{14}C/g ^{12}C).

Parenthetical values may also be converted to dose based on the relationship that on average seven pCi ^{14}C per gram of carbon is estimated to result in an annual dose of about 1.5 mrem (see Section 9.2.5). Based on the model defined in Figure 9-4, annual doses from a continuous release of 100 Ci of ^{14}C can be estimated at various distances as defined in Table 9-1. The isolines can be scaled linearly to source strengths other than 100 Ci/yr.

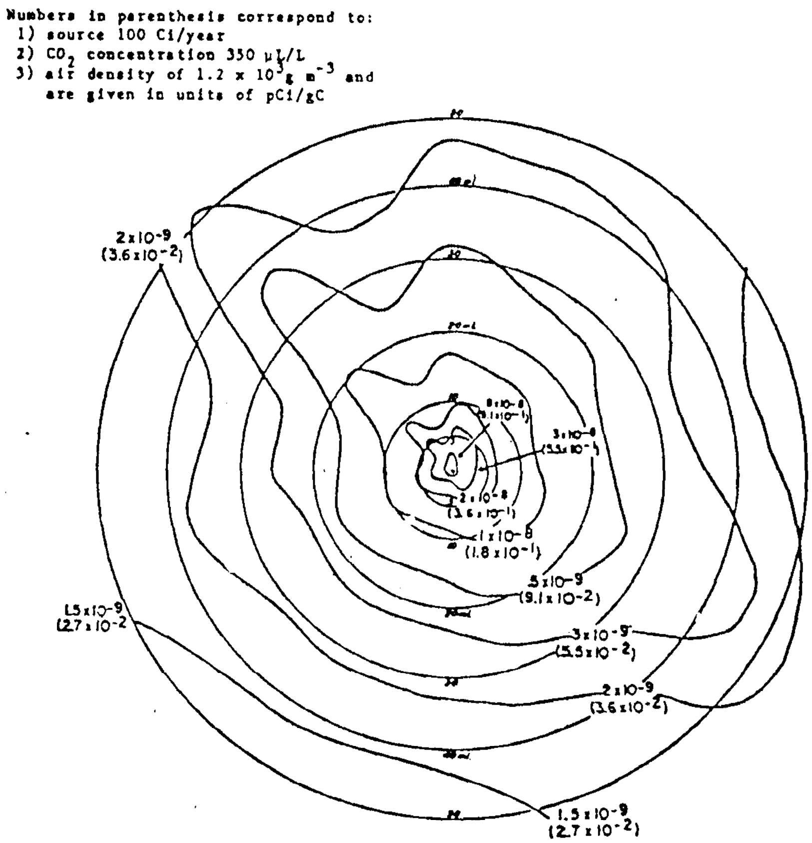

Figure 9-4. Annual Average Concentration for Uniform Continuous Source and Specific Activity (in parentheses) for 100 Ci/year (NCR93)

Table 9-1. Annual Average Doses Resulting from the Release of 100 Ci $^{14}CO_2$ for Distances Out to 50 Miles

Distance (miles)	**Atmospheric ^{14}C Concentration (pCi ^{14}C/g C)***	**Annual Dose - CEDE (mrem/yr)**
10	1.8×10^{-1}	3.8×10^{-2}
20	9.1×10^{-2}	2.0×10^{-2}
30	5.5×10^{-2}	1.2×10^{-2}
40	3.6×10^{-2}	7.7×10^{-3}
50	2.7×10^{-2}	5.8×10^{-3}

* Numbers were based on atmospheric CO_2 concentration of 350 μL/L and air density of 1.2×10^{-3} g/cm^3.

For the Yucca Mountain situation, the assumed 100 Ci/yr source release assumed in the power plant assessment can be approximately scaled to estimate potential doses from failed waste packages. The quick release fraction of C-14 in the waste packages may vary from 1.25 to 5.75 percent of the inventory. Using the high end estimate and a C-14 inventory of 11.4 Ci/waste package (DOE98, Vol. 3 p. 3-96), 0.673 Ci/waste package would be released. Assuming that this instantaneous release is transported completely to the ground surface and released, scaling the doses calculated for the power plant (for a location of 20 km from the source) gives a dose estimate on the order of 1.3×10^{-4} mrem/yr for gaseous releases from a single waste package. This estimate represents a conservative upper bound estimate for gaseous phase released in that actual gaseous releases from the repository would be attenuated during upward transport through the unsaturated zone by dissolution in pore waters in the overlying rocks, and by partitioning between fractures and the matrix porosity voids in the overlying rocks. Gaseous C-14 releases after initial breaching of the waste package are likely to be less than the instantaneous release fraction since ground waters interacting with the wastes would dissolve the C-14, with its subsequent transport through the ground water pathway rather than gaseous release.

9.2.6 Potential Non-Radiological Impacts of C-14

A concern uniquely associated with some contaminant radionuclides involves the transmutation effect and its potential for inducing molecular disorientation. The potential impact of chemical transmutation is of particular concern for genetic macro-molecules of DNA and RNA. Chemical transmutation refers to the fact that when a radioactive isotope emits a beta particle, it also undergoes chemical transformation due to the change in atomic number. For example, when C-14 undergoes radioactive decay, it becomes nitrogen. When such atoms are incorporated in critical molecules such as DNA, the resulting change in atomic number, recoil, or excitation may give rise to biologic effects, including mutation, beyond those induced by the attendant ionizing radiation. At issue, therefore, is whether or not dose-response values, involving cytogenetic/genetic effects for absorbed radiation energy, might underestimate the hazards presented by these potential radionuclide contaminants. Potential impacts of transmutation have been reviewed by the National Academy of Sciences (NAS). In their first report, the NAS Committee on the Biological Effects of Ionizing Radiation (BEIR) concluded:

> *...that the genetic effects of decays of H-3, C-14, and P-32 can, in fact, be attributed almost entirely to their beta radiation and that the contribution from transmutation is so small in comparison that it is justified to consider the main effect to come from the radiation emitted when the isotope disintegrates (NAS72).*

However, in the Committee's subsequent report (BEIR III), evidence was acknowledged which indicated a modest transmutation effect when C-14 (and H-3) occupied highly specific locations within DNA (NAS80). The Committee concluded that it still seems unlikely that neither H-3 nor C-14 decay are significantly underestimated by considering only the ionizing radiation dose accumulated by germ-line cells .

9.3 DEVELOPMENT OF PERFORMANCE SCENARIOS AND COMPLIANCE ISSUES

9.3.1 <u>Identification of Improbable Phenomena</u>

For a regulatory time frame that can extend to thousands of years, it is reasonable to conceive of circumstances defined by various natural and human-induced events and processes that may result in some persons at some time being exposed to levels well in excess of anticipated levels considered acceptable for an undisturbed repository. In recognition of the need to address repository performance under disturbed conditions, the NAS Committee on Technical Bases for Yucca Mountain Standards stated the following:

> *...the <u>probabilities</u> and <u>consequences</u> of modifications by climate change, seismic activity, and volcanic eruptions at Yucca Mountain are sufficiently boundable that these factors can be included in performance assessments that extend over this time frame. ... The challenge [therefore] is to define a standard that specifies a high level of protection but that does not rule out an adequately sited and well-designed repository because of highly improbable events* (NAS95). (Emphasis added.)

Substantial difficulties are likely to be encountered in making these predictions. Both the NRC and EPA have explicitly recognized that no analyses of compliance will ever constitute an absolute proof; the objective instead is a reasonable level of confidence in analyses that indicates whether limits established by the standard will be exceeded. Thus, in 40 CFR Part 191 (Appendix B), the EPA stated the following for a disturbed disposal system:

> *In making these various predictions, it will be appropriate for the implementing agencies to make use of rather complex computational models, analytical theories, and prevalent expert judgement relevant to the numerical predictions. Substantial uncertainties are likely to be encountered in making these predictions. In fact, sole reliance on these numerical predictions to determine compliance may not be appropriate; the implementing agencies may choose to supplement such predictions with qualitative judgement as well* (EPA85). (Emphasis added)

Similarly, in 10 CFR Part 60, the NRC acknowledged that for performance assessment "...it is not expected that complete assurance that they [performance objectives and criteria] will be met can be presented." (NRC81)

Events and processes that may require consideration are not limited to those identified by the NAS and shown in Figure 9-1. Over the years, numerous reports have identified generic events and processes that do not consider geographical or site-specific features (DOE74, DOE79, BUR80, IAE83, AND89, and DOE90a). Table 9-4 represents a consolidated listing that was used as a starting point in the development of disruptive scenarios for the Waste Isolation Pilot Plant (WIPP).

9.3.2 Screening of Events and Processes

Not all events and processes cited in Table 9-2 need necessarily be considered for Yucca Mountain. Phenomena such as erosion, sedimentation, etc. are certain to occur during extended time periods such as the NAS-suggested one million-year time frame. This suggests that these phenomena should be part of the base-case scenario. The effects of other events (e.g., sea-level variations, hurricanes, seiches, and tsunamis) are restricted to coastal areas.

To analyze the potential relevance of events and processes to a specific repository site, three criteria must be considered:

- Probability of occurrence
- Physical reasonableness
- Consequence

Table 9-2. Potentially Disruptive Events and Processes (DOE91)

Natural Events and Processes	**Human-Induced Events and Processes**
Celestial Bodies:	Inadvertent Intrusions
Meteorite Impact	Explosions
	Drilling
Surficial Events and Processes:	Mining
Erosion/Sedimentation	Injection Wells
Glaciation	Withdrawal Wells
Pluvial Periods	
Sea-Level Variations	Hydrologic Stresses:
Hurricanes	Irrigation
Seiches	Damming of Streams and Rivers
Tsunamis	
Regional Subsidence or Uplift	Repository- and Waste-Induced Events and Processes:
Mass Wasting	Caving and Subsidence
Flooding	Shaft and Borehole Seal Degradation
	Thermally-Induced Stress Fracturing in Host Rock
	Excavation-Induced Stress Fracturing in Host Rock
	Gas Generation
	Explosions
	Nuclear Criticality

To analyze the likelihood of a given event, it is most desirable to express its probability of occurrence in quantitative terms that draw on scientific data. Physical reasonableness as a screening criterion is a qualitative estimate of low probability that reflects subjective judgment. For subjective probability, the ICRP states:

> *...a number is assigned to the likelihood of an event occurring in a defined period of time, as a measure of the degree of belief that the event will actually occur during that time.... The assignment can be made on the sole basis of subjective judgement, no statistical experience being needed. The result is conceptually identical to a traditional probability and can be used in the same way* (ICR85a). (Emphasis added)

In instances where events are assigned subjective probabilities of occurrence, the ICRP offers an additional note of caution:

> *It is important to distinguish between the degree of belief and the idea of confidence limits applicable to an estimate of probability, which itself has some associated uncertainty.*

The third screening criterion is consequence. An assessment of consequence determines whether the event or process either alone or in combination with other phenomenon may adversely affect performance of the repository.

On the basis of these criteria, a proposed future standard may, for example, specify that events and processes with less than a specified chance of occurring within the regulatory period do not have to be considered in scenarios used to demonstrate compliance with the standard. Conversely, physically reasonable events and processes with significant impacts and probabilities greater than a threshold value would be considered for scenario development.

The likelihood of a disruptive event and its consequence must also be defined temporally. For some events (e.g., meteorite impact), the probability of occurrence over time is a constant. For these cases, the probability of events occurring within a year's time interval can be assessed from Poisson statistics. For other types of events, the probability of occurrence will vary with time after repository closure, or it may be co-dependent on the occurrence of other time-dependent events. This second and more complex event scenario is described in ICRP Publication 46 (ICR85b) and is illustrated in Figure 9-5. For this type of event-induced scenario, the probabilistic annual individual dose rate is a function of both the time of occurrence of the initiating event, t, and the time elapsed since its occurrence, (T-t).

9.3.3 Compliance With a Standard

It is the responsibility of the Nuclear Regulatory Commission (NRC) to assure compliance with EPA’s environmental radiation protection standards for Yucca Mountain. Accordingly, the following discussion is for illustrative purposes only and not indicative of how the NRC will discharge this responsibility.

A total system performance assessment employs a quantitative approach that characterizes the releases and health impacts of the disposal system. Key questions that must be addressed in the performance assessment are: "How reliable are the models employed in the performance assessment?" and "What is the uncertainty in the results of the performance assessment?" Preceding portions of this document have acknowledged uncertainties associated with all major elements affecting repository performance. Important sources of uncertainties pertain to the appropriateness of selecting scenarios representing conditions far into the future; the variability and/or lack of knowledge regarding many parameter values employed by the models; the reliability of historical data in predicting the time-dependent probability of future events that may

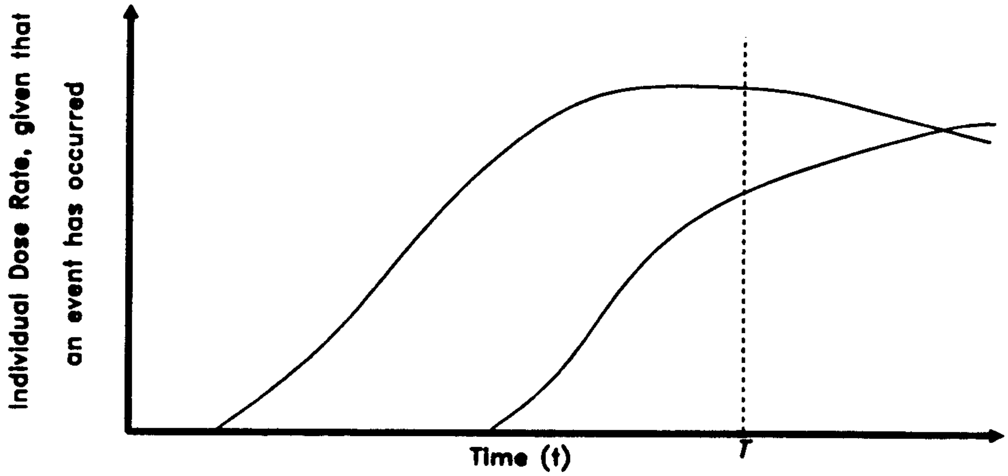

Figure 9-5. An Illustration of Hypothetical Individual Dose Rates Associated with a Disruptive Event Happening at Two Different Times after Disposal of Radioactive Waste

disrupt the repository; and the complex, but uncertain, interaction of independent variables on repository performance. While the uncertainty for some of the sources can be reasonably quantified (e.g., quantities of food and water ingested by humans), others are considerably more difficult (e.g., the probability of human intrusion). While there are no rigorous techniques for quantifying or eliminating uncertainties, several techniques for mitigating their impacts have been proposed by Bertram-Howery and Hunter (BER89), as summarized in Table 9-3.

The EPA has acknowledged that performance assessments will contain uncertainties and that many of these uncertainties cannot be eliminated. Accordingly, the EPA has previously stated that:

> *...standards must accommodate large uncertainties, including uncertainties in our current knowledge about disposal system behavior and the inherent uncertainties regarding the distant future.* (EPA85)

Uncertainty and sensitivity analyses are, therefore, important aspects of performance assessment. Uncertainty analysis involves determining the uncertainty in model projections that results from imprecisely known (or variable) model input parameters. Sensitivity analysis involves determining the contribution of individual input parameters to the uncertainty in model predictions.

Table 9-3. Techniques for Quantifying or Reducing Uncertainty in the Performance Assessment

Type of Uncertainty	Technique for Assessing or Reducing Uncertainty
Scenarios (Completeness, Logic, and Probabilities)	Expert Judgment and Peer Review Quality Assurance
Conceptual Models	Expert Judgement and Peer Review Sensitivity Analysis Uncertainty Analysis Quality Assurance
Computer Models	Expert Judgment and Peer Review Verification and Validation* Sensitivity Analysis Quality Assurance
Parameter Values and Variability	Expert Judgement and Peer Review Data-Collection Programs Sampling Techniques Sensitivity Analysis Uncertainty Analysis Quality Assurance

* To the extent possible. (BER89)

Because of the many uncertainties associated with the events and processes affecting repository performance, probability distributions of human exposure (and risk) are likely to vary over several orders of magnitude within the 5th and 95th percentile range. An important limitation of such a probability distribution is that no single value is correct in predicting future exposures. The probability distribution, however, does identify mean and median values, which represent <u>expected</u> values of dose (or risk) most likely to be received by individuals considered at maximum risk. To that extent, EPA (EPA85) has previously acknowledged that the most probable (or expected) value of a probabilistic distribution of estimated radiation exposure may be used to demonstrate compliance:

> *...the implementing agencies need not require that a very large percentage of the range of estimated radiation exposures...fall below limits [of the standard]. The Agency assumes that compliance can be determined based on best estimate predictions (e.g. the mean or the median of the appropriate distribution, whichever is higher).*

9.3.4 Development of Site Performance Issues

The subject of defining performance issues for a geologic repository has been addressed on a generic basis both in this country and abroad. Processes and events that could potentially affect repository integrity and performance have been identified so that they can be critically examined to develop site-specific performance scenarios for a particular candidate repository site. The challenge in developing site-specific scenarios for a candidate repository location is to credibly incorporate all the relevant processes and events that significantly affect repository performance.

To develop site-specific scenarios that will be addressed in licensing assessments, an iterative process of site characterization and evolving performance assessments has been used since the initiation of extensive site characterization work. Chapter 7 describes the sequence of total system performance assessments for the Yucca Mountain site that have been performed by DOE as site characterization has proceeded. These assessments have identified which processes and events have most relevance to sub-system and total system performance, and indicated the areas where more rigorous assessment capability was needed and areas where more extensive laboratory and field studies are necessary. As a result of this iterative site characterization - performance assessment process, a number of technical issues have been defined through interactions between the DOE and the NRC staff. Resolution of these issues is critical in developing a credible assessment of the site's performance that can be carried into the licensing process. These issues are discussed in more detail later in this section.

The progress of DOE's Yucca Mountain site characterization efforts is documented in semi-annual progress reports prepared as required by provisions of the Nuclear Waste Policy Act of 1982 (NWP83). Descriptions of the laboratory and field studies conducted by the DOE and other parties are reported in numerous reports published in the open technical literature. In Chapter 7, descriptions of the total system performance assessments carried out by the DOE, the NRC, and EPRI are given. These descriptions illustrate the iterative and evolving nature of the site characterization and performance assessment efforts carried out by these organizations. In developing the Yucca Mountain standard, the Agency has relied heavily on the information provided from these sources as well as the reviews of these efforts by experts outside the program.

The NRC staff have been in dialog with DOE concerning technical issues for the repository since inception of site characterization and repository design work. As a result of this dialog, the NRC staff has identified 10 Key Technical Issues (KTIs) to be resolved as part of demonstration of regulatory compliance, and has established Issue Resolution Status Reports (IRSRs) as the means by which the requirements for, and the status of, issue resolution are documented (see Chapter 4 of this BID).

The IRSRs are updated periodically as progress is made in issue resolution as a result of, for example, data additions or improvements in performance models. Resolution of the KTIs is sought before submission of the LA, but any issues resolved to the satisfaction of DOE and NRC staff during pre-licensing interactions are subject to being opened again during licensing reviews, e.g., by the licensing board.

9.3.4.1 Reviews of Recent Yucca Mountain Performance Assessments

DOE's Viability Assessment (DOE98) provided a recent basis for NRC expression of issues of current concern. The most recent DOE assessment, the TSPA for Site Recommendation (TSPA-SR) has not yet been formally reviewed by NRC, so the VA remains the most recent TSPA to undergo a thorough review by outside parties.

Although NRC had no statutory or other official role in review of the VA, staff performed a review of the total system performance assessment (TSPA) and licensing plan elements of the VA (i.e., cost estimates were not reviewed) as an aid to DOE's development of a complete and high-quality LA. The NRC comments were officially transmitted to DOE in a June 2, 1999 letter (NRC99). This transmittal of the results of the staff review had been preceded by informal expression of comments and an NRC/DOE Technical Exchange (NRC99a) in which NRC staff provided detailed feedback to DOE concerning issues associated with the TSPA-VA.

Comments on the VA were also documented by the NWTRB (TRB99) and DOE's independent Peer Review Panel (PRP99). In general, the NRC, NWTRB, and Peer Review Panel comments (summarized in Section 7.2.4 of this BID) were consistent with DOE's own assessment of issues associated with the VA's reference engineered design and the status of data and models used in the TSPA-VA (documented in Volume 4 of the VA, DOE98). Overall, there was consensus that the data base and performance models available for the VA were inadequate for the LA, and that there were technical issues stemming from the VA reference repository design that would be difficult to resolve in licensing reviews. These findings were consistent with DOE's assertion that the VA was a snapshot in time of evolution of repository design, performance modeling, and the technical data base.

9.3.4.2 Compliance Issues for Licensing Reviews

On the basis of their VA reviews, NRC staff identified, in NRC99, the following current issues, in order to help DOE "...to focus its program and develop a high-quality LA." Broad issues identified by the NRC staff are identified below along with associated issues more specific to the performance of the natural and engineered barriers.

While the growing emphasis on the engineered barriers is apparent in the evolution of DOE's efforts, there are some data gaps in the characterization of the natural barrier system that have important performance implications. Hydrologic characterization data are sparse for the down gradient portion of the ground water flow system at distances greater than 5 km from the repository. The absence of hydrologic data seriously limits the reliability of radionuclide transport calculations for projected repository releases. Some specific technical issues involved include:

- The range of infiltration estimates for water moving through the unsaturated zone and entering waste emplacement drifts in the repository

- The nature of the flow regime in the alluvial deposits lying beyond the Yucca Mountain location southward into the Amargosa Valley area, i.e., where and how flow from the tuff aquifer enters the alluvial sediments

- The extent of down gradient dispersion expected for the contamination plume from ant repository releases and the extent of mixing that can be assumed when releases from the repository enter the saturated zone beneath the repository

- The areal extent of the upward hydrologic gradient assumed to exist between the lower carbonate aquifer and the overlying tuff and alluvial flow system

These concerns, and planned efforts to address them and others, are discussed below.

Timely Selection of a Preferred Repository Design

NRC staff noted that the TSPA-VA identified, and provided preliminary evaluations of, design enhancements in comparison with the VA reference design that could provide defense-in-depth and improved performance for the engineered features of the repository system. The staff expressed concern that continued retention of flexibility and analysis of design options might not permit DOE to meet its schedule for preparation of a high-quality LA focused on a specific design.

DOE's selection of the EDA II design as the basis for future evolution of engineered barrier system design provides focus for future design development efforts. It will be necessary, however, that evolution of the data base and performance models needed to support use of the EDA II design (or its progeny) in the LA proceed at a rate, and with content, sufficient to produce a high-quality LA.

Waste Package Corrosion

Under present repository design concepts and waste isolation strategy, waste package corrosion resistance (i.e., lifetime before penetration which allows water to contact the waste form) is one of the most important factors in repository system performance. The waste package design for the VA used a so-called Corrosion Allowance Material (CAM), as the outer package wall, with a primary purposes of providing structural strength and radiation shielding. The CAM was A 516 steel. The VA waste package design also included a high-nickel alloy, Alloy 22, as the inner package wall material. This element of the waste package design served as the Corrosion Resistant Material (CRM) and was intended to be the principal basis for VA-design waste package lifetime.

The waste package design for the VA posed two major licensing issues. One was potential for rapid crevice corrosion of the Alloy 22 CRM, as a result of its being under the A 516 steel CAM. Rapid penetration of the CRM would negate its effectiveness as the means for achieving a long package lifetime. The other concern was the lack of a data base for corrosion of the Alloy 22 CRM under repository conditions. As noted in NRC99, the corrosion rate values used in the TSPA-VA, especially those for the CRM, were based primarily on results of expert elicitations rather than experiments under representative service conditions.

A key current licensing issue concerning waste package performance is therefore the sufficiency, and applicability to the repository environment, of data for waste package corrosion performance. NRC staff are concerned that, under current schedules, there is not enough time to gather quality data for the LA that are sufficient in quantity and duration of test conditions. NRC99 notes that it is appropriate for DOE and NRC to take into consideration more long-term data at later times (e.g., at the time disposal is approved), but sufficient data must be available to support the LA. In sum, corrosion parameter values used in the TSPA-VA that were based on results of expert elicitations must be adequately supplanted, for the LA, by data derived from measurements under environmental conditions expected for a repository at Yucca Mountain.

Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms

NRC99 states that DOE and NRC analyses both indicate that the fraction of waste packages contacted by water is the most important factor affecting dose for the ground water pathway. It also notes that the quantity and chemistry of water contacting the waste packages are the major factors determining waste package lifetime, and that these water characteristics also affect radionuclide release from the waste forms and waste packages.

NRC99 indicates that DOE recognizes the need for additional data concerning water quantities and chemistry in the repository, and also recognizes that performance models used in the TSPA-VA do not adequately capture the effects of coupled processes (thermal, chemical, hydrologic, and mechanical) on the quantity and chemistry of water that contacts the waste packages. NRC99 also states, however, that the range of licensing plan activities outlined by DOE in the VA are unlikely to provide, for the LA, an adequate basis for assessing the quantity and chemistry of water that contacts the waste packages and waste forms.

Two additions to DOE's suite of pre-LA activities, which could be completed before submission of the LA, are suggested in NRC99, based on a Drift Seepage Peer Review held in January 1999 (HUG99). One action would be to conduct systematic measurements of air permeability in horizontal boreholes in the repository host rock units. These measurements would provide data on the scales of variability in rock properties that are needed to adequately describe seepage. The other action would be to expand model development efforts to focus on explanation of patterns of seepage observed in the niche experiments.

The amount of additional data and analyses needed for an adequately supported LA will depend on how important the quantity and chemistry of water contacting the waste package is to the DOE's safety case.

Saturated Zone Flow and Transport

The saturated zone (SZ), which is the principal pathway for radionuclide transport from the repository to the dose receptor location, has been shown by DOE and NRC sensitivity studies to be an important factor in repository system performance. Potentially important SZ performance factors include low flow rates, dilution along the SZ pathway, and radionuclide holdup in alluvium between the repository and the dose receptor location. Dilution effects expected during pumping of contaminated water by the dose receptor may also help reduce the potential dose, but DOE did not take credit for this performance factor in the TSPA-VA.

A major issue for assessing the SZ contribution to performance in the TSPA-LA is the fact that the SZ is at present poorly characterized, and current DOE plans to improve characterization before submittal of the LA are limited. Flow and transport characterization data between about 5 and 20 km from the repository are at present highly limited; Gelhar (GEL98) described the situation as a “data hole”. Nye County is implementing an “Early Warning Drilling Program” (NYE99) in which shallow and deep wells will be drilled and tested down gradient of the repository; these wells, which are basically arrayed along Highway 95 at a radius of about 20 km south of the proposed repository location, will provide data concerning SZ flow and transport properties, but the data field will be limited under current DOE and Nye County plans. Well log data to date (NYE99a) indicate that the geologic features of the alluvial deposits are highly heterogeneous.

NRC99 expresses concern about availability of sufficient SZ characterization data for the LA. It notes that DOE’s planned activities are of low priority and will extend beyond current cutoff dates for refinement and update of SZ flow models, and that DOE might supplement the Nye County program with additional field work that could produce meaningful data for the LA, but has no present plans to do so. NRC99 also notes that DOE’s licensing plan in the VA documentation characterized the SZ flow and transport uncertainties as “moderate”, but that this designation appears to be inconsistent with the results of DOE and NRC/CNWRA sensitivity studies which show an important role for the SZ in overall repository system performance. Defensible demonstration in the LA of the SZ contributions to the natural system component of a defense-in-depth repository design will be essential.

Volcanic Disruption of the Waste Package

NRC99 asserts that the TSPA-VA analyses may underestimate the contribution to risk associated with future igneous activity. It states that NRC calculations show a small but finite risk from volcanic disruption of the proposed repository; DOE concluded in the TSPA-VA that risks from volcanism are negligible during a 10,000-year post-closure period. The NRC staff review of the TSPA-VA concluded that the DOE analyses are based on assumptions of physical conditions not representative of basaltic volcanism; that the data base was insufficient for evaluation of waste package performance under appropriate physical conditions; and that the modeling assumptions were incongruent with those used elsewhere in the TSPA-VA.

NRC99 asserts that models used by DOE in their volcanism evaluations are nonconservative, and that key assumptions are not supported by available data. It also notes that, although the VA licensing plan does not indicate planned activities to resolve these issues, supplemental work plans are being developed which could resolve them if implemented. NRC staff positions will be well documented in future versions of the Igneous Activity IRSR.

DOE responded to this criticism in its development of the TSPA for Site Recommendation (TRW00b). In the TSPA-SR, igneous activity was treated in a highly conservative manner, and proved to be the only mechanism for release of radionuclides in the first 10,000 years. However, despite the conservatism of the analysis, the mean dose rate remained below proposed regulatory criteria during 10,000 years.

Quality Assurance

NRC99 notes that DOE has consistently had problems in implementing its Quality Assurance (QA) program, which was reviewed and accepted by NRC. Audits have determined that some data in the technical data base are not traceable to their origins and could not be ensured to be applicable, correct, and technically adequate. The Technical Basis Document supporting the VA (DOE98a) indicated that a major portion of the data supporting the VA is not qualified in accord with QA program requirements.

NRC99 states that DOE must be able to demonstrate in its LA that the data, analyses, and designs of barriers and systems important to safety or waste isolation meet QA requirements of Appendix B to CFR Part 50.

DOE has recognized the need to meet QA requirements and has committed resources to development of an overall data qualification strategy and to resolution of QA issues. Plans for resolving QA issues address identification of unqualified data sets approved for qualification; methods for qualification and their rationale; technical disciplines required to achieve qualification; data evaluation criteria to be used; and criteria for changing data status from “unqualified” to “qualified”. The NRC has formed a QA Task Force which will conduct an independent and objective review of the DOE program and its implementation.

9.3.4.3 New Repository Design Concepts

In the DOE TSPA-VA a number of alternative engineered barrier design elements were evaluated relative to their effects on the performance of the reference repository design in the VA. These alternatives included such design options as drip shields and backfilling the emplacement drifts. Reviews of the VA assessments pointed out a number of uncertainties in DOE's VA performance assessments concerning the performance of engineered and natural barriers as well as site characteristics. These uncertainties might be reduced in two ways: by more extensive site characterization work to more defensibly quantify the range of expected natural barrier performance conditions and by introducing additional engineering features that compensate for the uncertainties in site conditions and in performance of the engineered repository system components.

Subsequent to issuance of the VA report in December 1998, DOE assessed alternative engineered repository designs intended to reduce issues associated with uncertainties in performance of the VA reference design. As described in Section 7.2.2.5 of this BID, DOE characterized and evaluated six design options and selected, as basis for the Site Recommendation, the so-called EDA II design, whose key design features are compared with those of the VA reference design in Table 7-9 of this BID.

The EDA II design features are in part responsive to repository performance and regulatory compliance issues identified by NRC staff, the NWTRB, and the Peer Review Panel as a result of their review of the TSPA-VA. NRC staff also, however, identified and characterized issues that are important to DOE's effort for timely development of a high-quality LA, and for pre-licensing resolution of KTIs (NRC99) so that the basis for LA-review findings concerning reasonable assurance of compliance with regulatory standards is as clear and well-established as possible. These issues were identified and discussed in Section 9.3.3.

Key performance issues and uncertainties associated with the reference VA design that are intended to be mitigated by the EDA II design include:

- The reduced areal mass loading and increased drift spacing for the EDA II design are intended to reduce uncertainties associated with the effect of heat emissions on the movement and chemical characteristics of water in the geologic formations adjacent to the repository, and issues concerning coupling of thermal, hydrologic, chemical, and mechanical phenomena such as the areal extent of the "dry out" zone and its changes over time.

- Elimination of the concrete lining and invert is intended to eliminate uncertainties about the effect of concrete materials on the chemical characteristics of water that can contact and corrode the waste packages and dissolve the waste form.

- Revision of the waste package design to use Alloy 22 as the outside wall material is intended to eliminate the potential for crevice corrosion of this material when it is under carbon steel, as was the case for the waste package design for the VA. This change would significantly increase waste package lifetimes by significantly reducing or eliminating the potential for rapid corrosion of the Corrosion Resistant Material. The EDA II waste package design concept uses, in addition to Alloy 22 as the outer wall material, 316L stainless steel as the inner wall material. As described by DOE at the summer meeting of the NWTRB (DOE99), this design concept, in conjunction with the planned heat load and expected repository temperatures, is expected to keep corrosion conditions for the Alloy 22 outside the "window of vulnerability" to rapid crevice corrosion, which spans the range of 80-100 ^{0}C and 50-100 percent relative humidity. If sufficiently supported by data and analyses, this strategy could eliminate crevice corrosion as a potential cause of early waste package failure.

- Use of drip shields and backfill is intended to add defense-in-depth to the engineered design and to defer the time at which seepage water could contact the waste packages and initiate aqueous corrosion. Drip shields would significantly delay the start of waste package corrosion, and would also dramatically reduce the uncertainty in projecting the effects of premature failures due to manufacturing defects. The occurrence of premature failures is a difficult performance factor to characterize because there is no directly-transferrable empirical data base from industrial experience that matches expected repository conditions. DOE's's base case VA assessments show that the projected dose rates for 10,000 years are strongly influenced by radionuclides releases from the single package assumed to be prematurely failed at 1,000 years. If more than one package is assumed to fail, dose estimates would increase in proportion. Use of drip shields is an example of an additional engineered measure to compensate for uncertainties that can not be reduced in any other way. By shielding the waste packages with the drip shield, ground water infiltrating into the drift has to first cause corrosion of the shield rather than directly contacting a prematurely-failed waste package.

- Blending of spent fuel subassemblies to reduce variations in waste package heat emissions is intended to reduce temperature levels and gradients that could stimulate corrosion and other degradation processes, and to reduce uncertainties in modeling of EBS performance. By eliminating "hot spots" in the repository, the averaging approach needed to model at the repository scale becomes more defensible.

TSPA results for this design in the TSPA-SR have demonstrated that the design, accompanied by improved modeling and scenario analysis, have accomplished the goal of improved repository performance. Unlike the VA design, the TSPA-SR showed that there are no releases from the repository under nominal conditions during 10,000 years.

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