Appendix I

Environmental Consequences of Long-Term Repository Performance

9909150202 - Part 5

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APPENDIX I. ENVIRONMENTAL CONSEQUENCES OF LONG-TERM REPOSITORY PERFORMANCE

This appendix provides detailed supporting information on the calculation of the environmental consequences of long-term (postclosure, up to 1 million years) repository performance. Chapter 5 summarizes these consequences for the Proposed Action, and Section 8.3 summarizes the cumulative impacts of Inventory Modules 1 and 2.

Section I.1 introduces the bases for long-term performance assessment calculations. Section I.2 provides an overview of the use of computational models developed for the Total System Performance Assessment – Viability Assessment used for this environmental impact statement (EIS). Section I.3 identifies and quantifies the inventory of waste constituents of concern for long-term performance assessment. Section I.4 details the modeling extensions to the Viability Assessment *base case* (high thermal load scenario with the Proposed Action inventory) developed to estimate potential impacts for other thermal load scenarios and expanded inventories. Section I.5 provides detailed results for waterborne radioactive material impacts, while Section I.6 provides the same for waterborne chemically toxic material impacts. Section I.7 describes atmospheric radioactive material impacts. To aid readability, all the figures have been placed at the end of the appendix.

I.1 Long-Term Repository Performance Assessment Calculations

This EIS analysis of postclosure impacts used and extended the modeling work done for the Total System Performance Assessment – Viability Assessment, as reported in the U.S. Department of Energy's (DOE's) *Viability Assessment of A Repository at Yucca Mountain, Volume 3* (DOE 1998a, Volume 3, all) and in the *Total System Performance Assessment – Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (TRW 1998a,b,c,d,e,f,g,h,i,j,k, all). The Proposed Action inventory under the high thermal load scenario is identical to the Viability Assessment base case, except that the Viability Assessment only considered 20 kilometers (12 miles) from the repository, while the EIS considers impacts of radiological dose to maximally exposed individuals through the groundwater pathway at 5, 20, 30, and 80 kilometers (3, 12, 19, and 50 miles) from the repository. The EIS analysis used a repository integrated program computer model (Golder 1998, all) that DOE used for the total-system model to calculate radiological doses through the groundwater pathway. This performance assessment model and supporting Viability Assessment process models were extended to predict waterborne chemically toxic material impacts. Additional calculations provided estimates of atmospheric radiological doses to local and global populations.

HOW ARE THE VIABILITY ASSESSMENT AND THIS EIS PERFORMANCE ASSESSMENT RELATED?

The long-term performance assessment for this EIS builds incrementally on the Viability Assessment (DOE 1998a, Volume 3, all; TRW 1998a,b,c,d,e,f,g,h,i,j,k, all).

This appendix reports only those aspects of the EIS long-term performance assessment that are incremental over the Viability Assessment. Only those parts of the analysis unique to the EIS are reported here, and the text refers to the appropriate Viability Assessment documents for information on the bases of the analyses.

The process of performing performance assessment analyses for this EIS required several steps. The EIS analysis was designed to incorporate the Total System Performance Assessment – Viability Assessment model of the base case repository configuration. Additional modeling (described in this appendix) was performed to evaluate the impacts of alternative thermal load scenarios and expanded waste inventories. The performance assessment model used for the Viability Assessment was expanded to accommodate calculations of the radiological dose to people at distances other than those used in the Viability

Assessment. Other adaptations to the model were made to calculate impacts from nonradiological materials not considered in the Viability Assessment.

The performance assessment model simulates the transport of radionuclides away from the repository into the unsaturated zone, through the unsaturated zone, and ultimately through the saturated zone to the accessible environment. Performance assessment analyses depend greatly on the underlying process models necessary to provide thermal-hydrologic conditions, near-field geochemical conditions, unsaturated zone flow fields, and saturated zone flow fields as a function of time. Using these underlying process models involves multiple steps that must be performed sequentially before performance assessment modeling can begin.

Figure I-1 shows the general flow of information between data sources, process models, and the total system performance assessment model. (Note: Figures are on pages I-67 to I-110.) Several computer models are identified in Figure I-1; these models are introduced in Section I.2. The general purpose of each of these computer models is described below its name in the figure. For example, TOUGH-2 is used for the mountain-scale thermohydrology model and the drift-scale and mountain-scale unsaturated zone flow model. The dashed box in the figure encompasses those portions of the performance assessment model that are modeled within the repository integration program. Other functions are run externally as "process models" to provide information to the repository integration program model. The ultimate result sought from performance assessment modeling is a characterization of radiological dose to humans with respect to time, which is depicted as the "Final Performance Measure" in the figure (the depiction is for illustrative purposes only).

I.2 Total System Performance Assessment Methods and Models

DOE conducted analyses for this EIS to evaluate potential long-term impacts to human health from the release of radioactive materials from the Yucca Mountain Repository. The analyses were conducted in parallel with, but distinct from, the Total System Performance Assessment calculations for the Viability Assessment (DOE 1998a, Volume 3, all). The methodologies and assumptions are detailed in the Total System Performance Assessment – Viability Assessment Technical Bases Document (TRW 1998a,b,c,d,e,f,g,h,i,j,k, all). Extensions of the Viability Assessment analyses to meet distinct EIS requirements were made using the same overall methodology.

The Total System Performance Assessment is a comprehensive systems analysis in which models of appropriate levels of complexity represent all important features, events, and processes to predict the behavior of the system being analyzed and to compare this behavior to specified performance standards. In the case of the potential Yucca Mountain Repository system, a Total System Performance Assessment must capture all of the important components of both the engineered and the natural barriers. In addition, the Yucca Mountain Total System Performance Assessment must evaluate the overall uncertainty in the prediction of waste containment and isolation, and the risks caused by the uncertainty in the individual component models and corresponding parameters.

The components of the Yucca Mountain Repository system include five major elements that the Total System Performance Assessment must evaluate:

- The natural environment unperturbed by the presence of underground openings or emplaced wastes
- Perturbations to the natural system caused by construction of the underground facilities and waste emplacement
- The long-term degradation of the engineered components designed to contain the radioactive wastes

- The release of the radionuclides from the engineered containment system
- The migration of these radionuclides through the engineered and natural barriers to the biosphere and their potential uptake by people, leading to a radiation dose consequence

The processes that operate within these five elements are interrelated. To model the complexity of the system efficiently, however, the following distinct process models were used in Total System Performance Assessment – Viability Assessment and in performance assessment calculations for this EIS:

- The unsaturated-zone flow was modeled directly with a three-dimensional, site-scale, unsaturated zone flow model, using the TOUGH2 program (Pruess 1991, all). Total System Performance Assessment calculations modeled *climate change* by assuming a series of step changes in climatic boundary conditions.
- Drift-scale unsaturated zone thermal-hydrology was modeled with the NUFT program (Nitao 1998, all) in three dimensions using a model domain that contains discrete waste packages and extends vertically from the water table to the ground surface.
- Waste package degradation was modeled using the WAPDEG program (TRW 1998l, all), which includes both individual package variability and package-to-package variability.
- Waste-form and cladding degradation was modeled in the repository integration program model using empirical degradation-rate formulas developed from available data. The model analyses used for the Total System Performance Assessment – Viability Assessment and for this EIS included representation of the protective benefits of fuel cladding for commercial spent nuclear fuel. The cladding failure model is described in detail in DOE (1998a, Volume 3, Section 3.5.2, pages 3-100 to 3-103).

CLIMATE CHANGE

The EIS performance assessment considered three climate scenarios: (1) a *present-day climate*, (2) a *long-term average* climate (wetter than the present-day climate) scenario, and (3) a scenario in which *superpluvial* conditions (much wetter than the present-day climate) are added at a short-duration fixed interval on a periodic basis 100,000 years after waste emplacement. The climate changes are step changes for the duration of the climate periods, and the lengths of the sequences are 10,000 years for the present-day dry climate and the superpluvial climate, and 90,000 years for the long-term average climate (DOE 1998a, Volume 3, Section 5.1.1, page 5-1).

- Engineered barrier-system transport was modeled in the repository integration program model (Golder 1998, all), using the program's cells algorithm. The transport modeling was based on an idealized representation consisting of a linked series of equilibrium batch reactors, including the waste form, waste package, corrosion products, and invert, and radionuclide transport through these reactors (TRW 1998e, all).
- Unsaturated zone radionuclide transport was modeled directly with a three-dimensional site-scale unsaturated zone-transport model using the FEHM model (Zyvoloski et al. 1995, all).
- Saturated zone flow and transport were modeled using a convolution method, in which the three-dimensional, site-scale, saturated zone, flow-and-transport FEHM model (Zyvoloski et al. 1995, all; TRW 1998g, all) was used to generate a library of solutions for translating time-varying mass inputs to the saturated zone into water concentrations at exposure locations downgradient.

- The biosphere was modeled using biosphere dose-conversion factors that convert saturated zone radionuclide concentrations to total radiological dose to an individual. The biosphere dose-conversion factors were developed using the GENII-S program (Leigh et al. 1993, all). The total radiological doses would be the final product of the Total System Performance Assessment calculations.

The performance assessment calculations for both the Total System Performance Assessment – Viability Assessment and this EIS were performed within a probabilistic framework combining the most likely ranges of behavior for the various component models, processes, and related parameters. This appendix presents the results in three main forms: (1) as probability distributions (for example, *complementary cumulative distribution functions*) for peak radiological dose to a maximally exposed individual during the 10,000 and 1 million years following repository closure; (2) as time histories of peak radiological dose to a maximally exposed individual over 10,000 and 1 million years following repository closure; and (3) in the case of this EIS only, as peak population radiological dose during 10,000 years for the local population using contaminated groundwater. For maximally exposed individuals, the Viability Assessment considered only a person 20 kilometers (12 miles) downgradient of the repository, while this EIS considers individuals 5, 20, 30, and 80 kilometers (3, 12, 19, and 50 miles) downgradient from the repository.

As noted above, the repository integration program model implements some of the individual process models directly, while other process models run outside the repository integration program model to produce *abstractions* in the form of data tables, response surfaces, or unit-response functions. The repository integration program model provides a framework for incorporating these abstractions, integrating them with other subsystem models. This is done in a *Monte Carlo* simulation-based methodology to create multiple random combinations of the likely ranges of the parameter values related to the process models. Probabilistic performance of the entire waste-disposal system is computed in terms of radiological dose to individuals at selected distances from the repository.

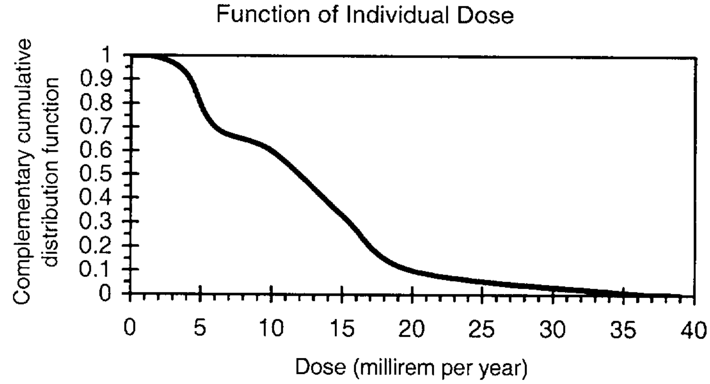
The EIS performance assessment methodology draws on the extensive analyses performed in support of the Total System Performance Assessment – Viability Assessment. Most of the process models (and their

THE COMPLEMENTARY CUMULATIVE DISTRIBUTION FUNCTION

Example application for individual radiological dose

The value of many variables such as individual radiological dose in the performance assessment models cannot be known precisely, but they can be described in a statistical sense. One of the statistical descriptions used is a complementary cumulative distribution function. The function for individual radiological dose is a curve that represents the probability of exceeding various levels of radiological dose. Although the complementary cumulative distribution function is a curve, one can make probability statements for points on the curve. For example, the stylized function for total radiological dose to an individual shown here indicates that there is a probability of 1 that radiological dose exceeds 0 millirem per year, a probability of 0.6 that radiological dose exceeds 10 millirem per year, a probability of 0.1 that radiological dose exceeds 20 millirem per year, and a probability of 0 that radiological dose exceeds 39 millirem per year.

Stylized Complementary Cumulative Distribution Function of Individual Dose



ABSTRACTION

Abstraction is the distillation of the essential components of a process model into a suitable form for use in a total system performance assessment. The distillation must retain the basic intrinsic form of the process model but does not usually require its original complexity. Model abstraction is usually necessary to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and uncertainty analyses (DOE 1998a, Volume 3, page A-1).

MONTE CARLO METHOD: UNCERTAINTY

An analytical method that uses random sampling of parameter values available for input into numerical models as a means of approximating the uncertainty in the process being modeled. A Monte Carlo simulation comprises many individual runs of the complete calculation using different values for the parameters of interest as sampled from a probability distribution. A different final outcome for each individual calculation and each individual run of the calculation is called a *realization* (DOE 1998a, Volume 3, page A-48).

abstractions) developed for the Viability Assessment were used directly in the analyses described in this appendix. Only components that were modified to account for the additional analyses considered in this EIS (but not the Viability Assessment) are described in this appendix.

I.3 Inventory

The analyses of long-term performance considered the following waste categories for radioactive materials:

- Commercial spent nuclear fuel comprised of both conventional enriched uranium fuel and mixed-oxide fuel using treated surplus fissile material that was reprocessed (consisting primarily of plutonium)
- DOE spent nuclear fuel
- High-level radioactive waste (some of which contains immobilized surplus weapons-usable plutonium)
- Greater-Than-Class-C waste and Special-Performance-Assessment-Required waste

The analysis assumed the waste would be in dual-shell waste packages. The outer shell would be comprised of corrosion-allowance material (carbon steel) with an inner shell of corrosion-resistance material (Alloy-22, a nickel-chromium alloy) (DOE 1998a, Volume 3, Figure 3-40, page 3-74). As described in TRW (1997a, Section 2.6), it was assumed that the waste packages would contain fuel assemblies from boiling-water reactors or pressurized-water reactors, naval ship or submarine reactors, DOE research reactors, foreign research reactors, or vitrified high-level radioactive waste in canisters. In addition, surplus plutonium not suitable for use in mixed-oxide fuel would be immobilized into 6.7-centimeter (2.6-inch)-diameter ceramic disks that would be packed in cylindrical *cans*, each containing approximately 1.0 kilogram (2.2 pounds) of plutonium (see Appendix A). Twenty-eight of these cans would be placed in a high-level radioactive waste canister and would occupy about 12 percent of the volume of the canister. The remainder of each canister would be filled with vitrified high-level radioactive waste. The plutonium encased in the high-level radioactive waste glass would then be incorporated in standard waste packages. This analysis assumed that the high-level radioactive waste would be in five-pack waste packages, each containing five high-level radioactive waste canisters and disposed of with or without a canister of DOE spent nuclear fuel. The inventory used for this EIS assessment was the same as that used in the Viability Assessment (TRW 1998m, all), which also considered more detailed sensitivity studies concerned with ceramic waste forms, alternative waste package configurations, individual fuel assembly configurations, and mixed waste forms (DOE 1998a, Volume 3, Section 5.5).

Thirty-nine radionuclides were included in the initial estimates of total inventories using the ORIGEN2 program (Croff 1980, all). In the Viability Assessment and the EIS performance assessment model, the list of 39 radionuclides was reduced to nine, based on the screening criteria discussed in this section and observing the nuclides that contributed most to total radiological dose as calculated in the performance assessment models. These nine radionuclides are carbon-14, iodine-129, neptunium-237, protactinium-231, plutonium-239, plutonium-242, selenium-79, technetium-99, and uranium-234.

This section discusses the inventories of waterborne radioactive materials used to model impacts and of some nonradioactive, chemically toxic waterborne materials used in the repository environment that could present health hazards. This section also discusses the inventory of atmospheric radioactive materials.

I.3.1 WATERBORNE RADIOACTIVE MATERIALS

There would be more than 200 radionuclides in the materials to be placed in the repository (see Appendix A). Because some of the radionuclides have a small inventory and some have short half-lives, this analysis did not need to consider all of these radionuclides when estimating long-term repository performance. Therefore, a screening analysis was performed to choose a subset of these radionuclides for further analysis.

I.3.1.1 Reduction of the List of Radionuclides for Performance Assessment Modeling

This evaluation of postclosure performance reduced the number of radionuclides considered by eliminating any radionuclides that:

- Have short half-lives and are not decay products of long-lived radionuclides

- Have high chemical sorption such that long travel times to a human exposure location would result in extremely low concentrations due to radioactive decay (unless the radionuclide has a large inventory and the potential for colloidal transport)

- Have low biosphere dose-conversion factors

Any one or any combination of these factors would result in a diminished contribution by the radionuclide to the total radiological dose; thus, eliminating that radionuclide from consideration would not reduce estimates of radioactive material impacts. Based on these considerations and previous performance analysis results (TRW 1995, all), DOE selected nine dominant radionuclides for analysis and focused on those radionuclides that would have the most impact on human health, thereby enhancing modeling efforts.

Two other factors were a part of the decision to reduce the list of radionuclides explicitly modeled in performance assessment calculations. First, there was a need to reduce the number of radionuclides in order to focus on only those radionuclides with the greatest impact on human health. Large multidimensional flow-and-transport models such as the unsaturated zone and saturated zone particle-tracking and transport models that are part of the repository integration program model require extensive computer time (days or weeks). Hence, it was necessary to focus on those radionuclides that would have the most impact on human health. The reduced list of radionuclides adequately characterized the impacts without requiring an unnecessary computer modeling effort. Second, knowledge and experience gained from earlier assessments (Wilson et al. 1994, all; TRW 1995, all), as well as the experience of other organizations (Wescott et al. 1995, all), were incorporated into the choice of radionuclides included for analysis. To be included for the Total System Performance Assessment – Viability Assessment, a radionuclide had to pass the elimination process performed under the basic criteria described above. It also had to have an overall larger inventory than a similar radionuclide with similar performance importance, or it had to have been identified as important in earlier studies.

The following is a discussion of the further rationale for the final selection of the specific radionuclides to model.

Selected Radionuclides

- *Carbon-14, technetium-99, and iodine-129.* These radionuclides are highly soluble and exhibit little or no chemical sorption. Technetium-99 and iodine-129 were major radiological dose contributors in previous Total System Performance Assessments (Barnard et al. 1992, all; Wilson et al. 1994, all). Carbon-14 and iodine-129 could be liberated from the waste packages as gases and subsequently dissolved in water.

- *Selenium-79, protactinium-231, uranium-234, and neptunium-237.* These radionuclides are relatively soluble and have relatively low chemical sorption. Selenium-79 is the major radiological dose contributor through a cow's liver pathway. Protactinium-231 has a relatively high sorption coefficient, but because it is a decay product of uranium-235, it should be transported relatively quickly and have a long residence time. Uranium-234 has a large inventory, is a decay product of uranium-238, and has a high biosphere dose conversion factor. Neptunium-237 has been the most important radionuclide in previous Total System Performance Assessments for exposure periods between 20,000 and 1,000,000 years after repository closure.

- *Plutonium-239 and plutonium-242.* Although these plutonium isotopes are highly sorbing, they were included on the list because of their large inventory and the possibility that they might migrate by colloidal transport. These radionuclides would be among the most important radionuclides involved in colloid-facilitated transport, if colloidal transport of plutonium were determined to be important. Plutonium-242 was selected over plutonium-240 because of its longer half-life, thus making it more likely to reach the accessible environment (especially via colloidal transport).

Radionuclides Not Selected

- *Curium-246, curium-245, americium-241, americium-243, plutonium-240, uranium-238, thorium-230, radium-226, lead-210, cesium-137, cesium-135, niobium-94, and nickel-59.* These radionuclides were among those selected by the U.S. Nuclear Regulatory Commission for its Iterative Performance Assessment (Wescott et al. 1995, page 5-5). The Viability Assessment did not include curium isotopes because of their similarity to plutonium. Americium isotopes were not included directly because they have short half-lives, americium-243 was included in the plutonium-239 inventory, and the activity of americium-241 was included in the neptunium-237 inventory. Plutonium-240 was not selected because it is highly sorbing (although plutonium-242 was selected to address colloidal transport). Uranium-238 was not selected because its decay product uranium-234 was chosen. Ingrowth of uranium-238 was compensated for by increasing the uranium-234 inventory. Thorium, radium, lead, cesium, niobium, and nickel were generally not included because they are highly sorbing. In addition, lead-210, cesium-137, and radium-226 have relatively short half-lives, while cesium-135, nickel-59, and niobium-94 have low inventories. For these reasons, none of these radionuclides would contribute significantly to radiological dose (that is, including these radionuclides in the calculations would not change the estimates of dose within the number of significant figures reported for results).

Using only a subset of the radionuclides leads to potential underestimates of impacts to humans. The modeling results reported in Chapters 5 and 8 show that in the first 10,000 years, the radiological dose is dominated by technetium-99, iodine-129, and carbon-14. These radionuclides all have relatively high solubility and little chemical sorption. There are no other radionuclides with a meaningful inventory in the proposed repository that share these characteristics. Thus, the error introduced by excluding other radionuclides is very small in the first 10,000 years after repository closure.

The potential for underestimating impacts increases with time periods greater than 10,000 years after repository closure. The possible error is largely due to the modeling of a few nuclides without modeling the entire decay chain for the nuclide. Based on decay equilibrium calculations for the first 1,000,000 years after repository closure, the error from neglecting all other nuclides is about 5 percent of the total radiological dose rate (DOE 1998a, Appendix C, page C6-2 and Figure C6-1).

The inventories for the categories of spent nuclear fuel and high-level radioactive waste described in the following paragraphs include these nine radionuclides. The inventories of these radionuclides were used in the performance assessment model to estimate the impacts to people.

The Viability Assessment and these EIS performance assessment calculations included only certain nuclides of prominent decay chains. To account for the lack of ingrowth of decay products, modifications were made to the nine radionuclide inventories for commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste. These modifications helped produce conservative estimates of the activities of these nuclides (that is, estimates of the inventory would be equal to or greater than the real inventory, so that any uncertainty would tend to overpredict impacts), which were then used by the performance assessment model to determine impacts to individuals at specific exposure locations. Three of the radionuclide inventories were modified as follows:

- The amount of protactinium-231 was entered in the repository integration program model as grams per waste package of protactinium-231 rather than as curies per waste package, which allowed the inventory of protactinium-231 to be modeled in secular equilibrium with its parent nuclide uranium-235.

- The estimated activities of neptunium-237 and uranium-234 were increased by 58 percent and 13 percent, respectively. The increase in the activity of neptunium included the activity of the precursors californium-249, curium-245, plutonium-241, and americium-241 in the performance assessment model. Neptunium-237 transports faster than the precursor radionuclides, so putting the entire inventory in neptunium-237 would not underestimate the radiological dose. The increase of activity in uranium-234 included the activity of precursors such as californium-250, curium-246, plutonium-242, americium-242, curium-242, uranium-238, and plutonium-238.

I.3.1.2 Radionuclide Inventory Used in the Performance Assessment Model

Radioactive material inventories were included in the performance assessment model for Total System Performance Assessment calculations by the following waste categories: commercial spent nuclear fuel, high-level radioactive waste, and DOE spent nuclear fuel. For each waste category, an *abstracted waste package* was represented with an average radionuclide inventory for the nine radionuclides selected in the screening analysis (see Section I.3.1.1).

The quantity of abstracted packages was determined, in part, by averaging the characteristics of the several different types of actual waste packages planned for each waste category and, in part, by demands for a symmetrical, replicating arrangement of waste packages necessary for efficient thermal-hydrologic modeling. Therefore, the quantity of abstracted packages in the performance assessment model differed slightly from the actual quantity of waste packages identified in Appendix A and elsewhere. Other inventory differences between the performance assessment model and Appendix A, and the associated implications, are discussed in this section.

ABSTRACTED WASTE PACKAGES

The number of waste packages used in the performance assessment simulations do not exactly match the number of actual waste packages specified in TRW (1998n, all).

The performance assessment model uses three types of *abstracted waste packages,* representing the averaged inventory of all the actual waste packages used for a particular waste category (commercial spent nuclear fuel, DOE spent nuclear fuel, or high-level radioactive waste).

While the number of abstracted waste packages might vary from TRW (1998n, all), the total radionuclide inventory (activity) represented by all of the abstracted waste packages collectively is equivalent to the total inventory given in Appendix A, unless otherwise noted.

I.3.1.2.1 *Commercial Spent Nuclear Fuel*

The commercial spent nuclear fuel inventory is discussed in detail in Appendix A. The quantities and activities were weighted according to the contributors and the expected waste package configurations. Using these data, the analysis established an abstracted waste package commercial spent nuclear fuel radionuclide inventory for the Total System Performance Assessment – Viability Assessment and EIS performance assessment modeling (TRW 1998m, page 5-10). Table I-1 lists the radionuclide inventory for commercial spent nuclear fuel used for both the EIS and Viability Assessment analyses.

Table I-1. Performance assessment model radionuclide inventory (curies per waste package) for commercial spent nuclear fuel.[a]

Nuclide	Inventory
Carbon-14	12
Iodine-129	0.29
Neptunium-237	11
Protactinium-231[b]	5.1
Plutonium-239	3,100
Plutonium-242	17
Selenium-79	3.7
Technetium-99	120
Uranium-234	21

a. Source: DOE (1998a, Volume 3, page 3-96).
b. Protactinium-231 is listed in grams per package to facilitate modeling as an equilibrium decay product of uranium-235. The specific activity of protactinium-231 is 0.0000022 curies per gram.

I.3.1.2.2 *DOE Spent Nuclear Fuel*

The DOE spent nuclear fuel inventory is discussed in detail in Appendix A. Table I-2 lists the abstracted waste package radionuclide inventory for DOE spent nuclear fuel used for the Viability Assessment and the EIS analyses for the Proposed Action.

I.3.1.2.3 *High-Level Radioactive Waste*

High-level radioactive waste is the highly radioactive material resulting from the reprocessing of spent nuclear fuel, and the inventory for its disposal is presented in Appendix A. The high-level radioactive waste inventory assembled for Total System Performance Assessment – Viability Assessment and EIS performance assessment modeling was derived from the inventories of high-level radioactive waste at the Hanford Site, Savannah River Site, Idaho National Engineering and Environmental Laboratory, and West

Table I-2. Performance assessment model radionuclide inventory (curies per waste package) for DOE spent nuclear fuel.[a]

Nuclide	Inventory
Carbon-14	0.31
Iodine-129	0.0057
Neptunium-237	0.15
Protactinium-231[b]	0.66
Plutonium-239[c]	155
Plutonium-242	0.11
Selenium-79	0.089
Technetium-99	2.6
Uranium-234	0.54

a. Source: DOE (1998a, Volume 3, page 3-96).
b. Protactinium-231 is listed in grams per package to facilitate modeling as an equilibrium decay product of uranium-235. The specific activity of protactinium-231 is 0.0000022 curies per gram.
c. Inventory for plutonium-239 is correct; DOE (1998a, Volume 3, page 3-96) contains a typographical error.

Valley Demonstration Project. This inventory was established from the National Low-Level Waste Database and weighted for the expected contributions from the four principal high-level radioactive waste sites listed above using quantities calculated in the *Waste Quantity, Mix and Throughput Report* (TRW 1997a, all). This inventory is listed in Table I-3 for the nine modeled radionuclides.

Table I-3. High-level radioactive waste mass and volume summary.

Parameter	EIS analyses	Appendix A
Mass (metric tons)	NA[a]	58,000
Volume (cubic meters)	18,000	21,000
Number of canisters	19,234	22,280
Waste packages (5-packs)	3,848	4,456[b]

a. NA = not applicable.
b. Derived from data presented in Appendix A.

These data were included in the high-level radioactive waste inventory for the Viability Assessment base case (TRW 1998o, all); long-term performance assessment analyses for this EIS used this same inventory.

Recent updates of the waste inventories from the DOE sites are in Appendix A. The most recent estimates from these sites indicated a higher total volume of high-level radioactive waste but with an overall lower activity. Appendix A provides a 1998 summary of the potential total mass, volume, and number of canisters of high-level radioactive waste that would be available to the Yucca Mountain Repository from the principal waste sites.

These performance assessment analyses did not use the most recent information reported in Appendix A, because the more recent estimates of high-level radioactive waste activity were received too late for inclusion in the Viability Assessment and EIS performance assessment calculations (see TRW 1998f, page 6-16). A sensitivity analysis of high-level radioactive waste was performed by comparing the high-level radioactive waste inventory used in EIS analyses to the inventory in Appendix A. The results of the analysis showed that the estimate of total radiological dose to maximally exposed individuals at 20 kilometers (12 miles) from the Yucca Mountain Repository, using the high-level radioactive waste base case inventory for the Viability Assessment, led to higher amounts of radionuclides contributing to radiological dose than those calculated using the revised data from Appendix A. Therefore, actual impacts would be lower than estimated if the more recent information were used. Table I-4 compares the nine radionuclide inventories used in the Viability Assessment and EIS analyses with those used in the Appendix A inventory. Note that the nine modeled radionuclides do not contribute equally to radiological

Table I-4. Comparison of high-level radioactive waste inventories (curies per package).

Nuclide	TSPA-VA inventory[a] (3,848 packages)	Appendix A inventory (4,456 packages)
Carbon-14	0	0.032
Iodine-129	0.000042	0.0085
Neptunium-237	0.74	0.13
Protactinium-231[b]	0.036	0.82
Plutonium-239	24	68
Plutonium-242	0.02	0.014
Selenium-79	0.29	0.49
Technetium-99	30	13
Uranium-234	0.9	0.15

a. Source: TSPA-VA = (Total Systems Performance Assessment – Viability Assessment); DOE (1998a, Volume 3, page 3-96).

b. Protactinium-231 is listed in grams per package to facilitate modeling as an equilibrium decay product of uranium-235. The specific activity of protactinium-231 is 0.0000022 curies per gram.

dose, so a comparison of the inventories in Table I-4 can be misleading. For example, neptunium-237 typically contributes more than 90 percent of the dose in the 1-million-year period, so the larger inventory of neptunium-237 in the Total Systems Performance Assessment – Viability Assessment inventory column is more important that the smaller inventory of other radionuclides relative to the Appendix A inventory column. Similarly, iodine-129 and technetium-99 inventories contribute most of the dose in the 10,000-year period, so difference in those inventories are most important in that case.

The source used for the Viability Assessment to establish the inventory of high-level radioactive waste was the Characteristics Database (DOE 1992, all). Appendix A contains data submitted by the individual sites in response to an EIS data call. The differences in the data from each source are listed below by site.

Discussion of differences is limited to the nine radionuclides modeled in the performance assessment analyses.

Hanford Site

- The Characteristics Database (DOE 1992, all) assumes 1,650 kilograms (3,630 pounds) of glass per canister.

- Appendix A reports the mass of glass per canister as 3,040 kilograms (6,700 pounds). Values in Appendix A are generally higher than those presented in the Characteristics Database (DOE 1992, all); these values are listed in Table I-5. Nuclide values which are generally lower in Appendix A than the Characteristics Database are presented in Table I-6.

Table I-5. Nuclides at the Hanford Site for which Appendix A presents values greater than those in the Characteristics Database.[a]

Nuclide	Factor
Iodine-129	100
Protactinium-231	100,000
Plutonium-239	2.5
Selenium-79	8
Uranium-234	5

a. Source: DOE (1992, all).

Table I-6. Nuclides for which Appendix A presents values lower than those in the Characteristics Database.[a]

Nuclide	Factor
Neptunium-237	100
Technetium-99	3

a. Source: DOE (1992, all).

Idaho National Environmental and Engineering Laboratory

- The Characteristics Database (DOE 1992, all) inventory numbers do not include the projected high-level radioactive waste inventory from the Argonne National Laboratory-West ceramic and metal waste matrices (approximately 102 canisters).

- Appendix A reported values for carbon-14 and iodine-129 (0.000083 and 0.017 curie per canister, respectively), while the Characteristics Database (DOE 1992, all) reported no values for these nuclides.

- The Characteristics Database (DOE 1992, all) reported 0.08 curie per canister for selenium-79; however, no value is reported for use in Appendix A.

- For the other nuclides, the values reported in Appendix A are greater by a variety of factors, as listed in Table I-7.

Table I-7. Nuclides at the Idaho National Engineering and Environmental Laboratory for which Appendix A presents values greater than those in the Characteristics Database.[a]

Nuclide	Factor
Neptunium-237	270
Plutonium-239	2.25
Plutonium-242	1.65
Technetium-99	3.7
Uranium-234	200,000

a. Source: DOE (1992, all).

Savannah River Site

- In general, the Appendix A values for the other nuclides are slightly smaller (generally less than 1 percent) than those presented in the Characteristics Database (DOE 1992, all). The uranium-234 value reported in Appendix A is 77 percent less; however, most of the other nuclides are within 1 percent of the values in the Characteristics Database.

West Valley Demonstration Project

- The Characteristics Database (DOE 1992, all) does not include data for carbon-14 or iodine-129; Appendix A uses approximately 0.53 and 0.00081 curie per canister, respectively, for these nuclides.

- Neptunium-237, plutonium-239, plutonium-242, and protactinium-231 differ slightly in Appendix A (by about 1 percent) due largely to the difference in reporting accuracy (Appendix A reports two significant figures; the Characteristics Database reports three).

- Uranium-234 is increased by about 15 percent in Appendix A.

- Technetium-99 and selenium-79 are both higher in Appendix A by a factor of approximately 15.

I.3.1.2.4 *Greater-Than-Class-C and Special-Performance-Assessment-Required Wastes*

Wastes with concentrations above Class-C limits (shown in 10 CFR Part 61.55, Tables 1 and 2 for long and short half-life radionuclides, respectively) are called Greater-Than-Class-C low-level waste. These wastes generally are not suitable for near-surface disposal. The Greater-Than-Class-C waste inventory is discussed in detail in Appendix A.

DOE Special-Performance-Assessment-Required low-level radioactive waste could include production reactor operating wastes, production and research reactor decommissioning wastes, non-fuel-bearing components of naval reactors, sealed radioisotope sources that exceed Class-C limits for waste classification, DOE isotope production related wastes, and research reactor fuel assembly hardware. The Special-Performance-Assessment-Required waste inventory is discussed in detail in Appendix A.

The final disposition method for Greater-Than-Class-C and Special-Performance-Assessment-Required low-level radioactive waste is not known. If these wastes were to be placed in a repository, they would be placed in canisters before shipment. This appendix assumes the use of a canister similar to the naval dual-purpose canister described in Section A.2.2.5.6.

Table I-8 lists existing and projected volumes through 2055 for the three Greater-Than-Class-C waste sources. DOE conservatively assumes 2055 because that year would include all Greater-Than-Class-C low-level waste resulting from the decontamination and decommissioning of commercial nuclear reactors. The projected volumes conservatively reflect the highest potential volume and activity expected based on inventories, surveys, and industry production rates.

Table I-8. Greater-Than-Class-C low-level waste volumes (cubic meters)[a] by source.[b]

Source	1993	2055
Nuclear electric utility	26	1,300
Sealed sources	40	240
Other	74	470
Totals	**140**	**2,010**

a. To convert cubic meters to cubic feet, multiply by 35.314.
b. Source: DOE (1994, Tables 6-1 and 6-3).

The data concerning the volumes and projections of Greater-Than-Class-C low-level waste are from Appendix A-1 of the *Greater-Than-Class-C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities, and Other Characteristics* (DOE 1994, all). This appendix provides detailed radioactivity reports for such waste currently stored at nuclear utilities. Table I-9 summarizes the radioactivity data for the nine radionuclides modeled in performance assessment calculations, decayed to 2055.

I.3.2 WATERBORNE CHEMICALLY TOXIC MATERIALS

Waterborne chemically toxic materials that could present a human health risk would be present in materials disposed of in the repository. The most abundant of these chemically toxic materials would be nickel, chromium, and molybdenum, which would be used in the waste package, and uranium in the disposed waste. Uranium is both a chemically toxic and radiological material. Screening studies were conducted to determine which, if any, of these or other materials could be released in sufficient quantities to have a meaningful impact on groundwater quality.

Table I-9. Performance assessment model radionuclide inventory (curies per waste package) for Greater-Than-Class-C and Special-Performance-Assessment-Required waste.[a]

Nuclide	Inventory
Carbon-14	38
Iodine-129	0.000000012
Neptunium-237	0.000000052
Protactinium-231[b]	0.0000015
Plutonium-239	48
Plutonium-242	0.0000040
Selenium-79	0.0000010
Technetium-99	2.6
Uranium-234	0.00000062

a. Source: TRW (1999a, Table 2.2-6, page 2-10).
b. Protactinium-231 is listed in grams per package to facilitate modeling as an equilibrium decay product of uranium-235. The specific activity of protactinium-231 is 0.0000022 curies per gram.

I.3.2.1 Identification of Waterborne Chemically Toxic Materials

An inventory of chemical materials to be placed in the repository under the Proposed Action was prepared. The inventories of the chemical components in the repository were combined into four groups:

- Materials outside the waste packages (concrete, copper bus bars, structural members, emplacement tracks and supports, etc.)
- Carbon steel in the outer layer of the waste packages
- Alloy-22 in the inner layer of the waste packages
- Materials internal to the waste packages

These materials were organized into groups with similar release times for use in the screening study. Table I-10 lists the chemical inventories. Plutonium is not listed in Table I-10 because, while it is a heavy metal and therefore could have toxic effects, its radiological toxicity far exceeds its chemical toxicity (DOE 1998b, Section 2.6.1) (see Section I.5 for more information). Also, while there are radiological limits set for exposure to plutonium, no chemical toxicity benchmarks have been developed. Therefore, because of this lack of data to analyze chemical toxicity, plutonium was not analyzed for the chemical screening.

I.3.2.2 Screening Criteria

Only those chemicals likely to be toxic to humans were carried forward in the screening study. Uranium was an exception; it was carried forward due to its high inventory and also to serve as a check on the screening study. Chemicals included in the substance list for the U.S. Environmental Protection Agency's Integrated Risk Information System (EPA 1999, all) were evaluated further to determine a concentration that would be found in drinking water in a well downgradient from the repository. The chemicals on the Integrated Risk Information System substance list that would be in the repository are barium, boron, cadmium, chromium, copper, lead, manganese, mercury, molybdenum, nickel, selenium, uranium, vanadium, and zinc.

Table I-10. Inventory (kilograms)[a] of chemical materials placed in the repository under the Proposed Action.

Element	Inventory					
	Outside package	Carbon steel	Alloy-22	Internal	High-level radioactive waste	Totals
Aluminum	0	0	0	1,205,000	0	1,205,000
Barium	0	0	0	0	19,000	19,000
Boron	0	0	0	223,000	0	223,000
Cadmium	0	0	0	0	43,000	43,000
Carbon	286,000	796,000	8,000	5,000	0	1,096,000
Chromium	0	0	9,670,000	3,903,000	0	13,573,000
Cobalt	0	0	1,357,000	27,000	0	1,384,000
Copper	1,135,000	0	0	3,000	0	1,139,000
Iron	91,482,000	320,089,000	2,171,000	9,000	0	413,751,000
Lead	0	0	0	0	2,000	2,000
Magnesium	0	0	0	12,000	0	12,000
Manganese	234,000	3,007,000	271,000	2,000	0	3,514,000
Mercury	0	0	0	0	200	200
Molybdenum	0	0	5,934,000	302,000	0	6,236,000
Nickel	0	0	29,727,000	5,563,000	0	35,290,000
Phosphorus	37,000	114,000	11,000	0	0	161,000
Selenium	0	0	0	0	300	300
Silicon	361,000	943,000	43,000	7,000	0	1,354,000
Sulfur	46,000	114,000	11,000	0	0	170,000
Titanium	0	0	0	2,000	0	2,000
Tungsten	0	0	1,628,000	0	0	1,628,000
Uranium	0	0	0	70,000,000	0	70,000,000
Vanadium	0	0	190,000	0	0	190,000
Zinc	0	0	0	3,000	0	3,000

a. To convert kilograms to pounds, multiply by 2.2046.

I.3.2.3 Screening Application

The screening calculations for chemically toxic materials assume that groundwater would move through the repository, dissolving and transporting the potentially chemically toxic materials. This analysis treated the repository materials and the carbon-steel layer of the waste package as simultaneously degrading in the groundwater. After the carbon-steel layer of the waste degraded, the Alloy-22 corrosion-resistant material would start degrading. Finally, once the waste package was breached, the materials inside the waste packages would become available for dissolution and transport.

I.3.2.3.1 *Solubility of Chemically Toxic Materials in the Repository*

The release of chemically toxic materials to the accessible environment depends on the solubility of the materials in water. Table I-11 lists the solubility values used for the screening study.

Maximum source concentrations for materials in the repository that are not a part of the waste package materials were calculated as solubilities of an element in repository water. This calculation would provide the maximum possible concentration of that element in water entering the unsaturated zone if it dissolved at a sufficiently high rate. The solubilities were obtained by modeling with the EQ3 code (Wolery 1992, all). The simulations were started with water from well J-13 near the Yucca Mountain site (Harrar et al. 1990, all). EQ3 calculates chemical equilibrium of a system so that by making successive runs with gradually increasing aqueous concentrations of an element, eventually a result will show the saturation of a mineral in that element. That concentration at which the first mineral saturates is said to be

SCREENING ANALYSIS

A *screening analysis* is a method applied to avoid unnecessary calculations and focus on potentially large impacts.

The repository would contain many materials that could result in impacts to human health. However, most of these materials would either not be present in large enough quantities or not dissolve readily enough in water to pose a risk.

To evaluate the potential risk posed by so many materials, an analysis could either rigorously evaluate every material at great cost, or could apply a screening analysis to identify those materials with too little inventory or too little solubility to be of concern. The screening analysis applied for the EIS was a simplified scoping calculation which resulted in a short list of materials that merited further consideration. Any preliminary concentrations predicted under the simplified assumptions of the screening analysis were treated as conservative estimates used only to determine if the material should be rigorously modeled again using the performance assessment model. For those materials that the screening analysis indicated must be evaluated further, more realistic concentrations and impacts were computed with the performance assessment model and are reported in Sections I.5 and I.6.

Table I-11. Source concentrations[a] (milligrams per liter)[b] of waterborne chemically toxic materials for screening purposes.

Element	Concentration	Aqueous species	Reference
Boron	6,400	B(OH)3aq	Solubility in repository water by EQ3[c] simulation
Chromium	300	CrO_4^{--}	EQ6[d] simulation of Alloy 22 corrosion
Copper	0.018	$CuOH^+$, $Cu(CO_3)aq$, Cu^{++}	Solubility in repository water by EQ3[c] simulation
Manganese	4.40×10^{-11}	Mn^{++}	EQ6[d] simulation of Alloy 22 corrosion
Molybdenum	218	MoO_4^{--}	EQ6[d] simulation of Alloy 22 corrosion
Nickel	1.00×10^{-6}	Ni^{++}	EQ6[d] simulation of Alloy 22 corrosion
Uranium	0.6	$UO_2(OH)_2aq$	Derived from TRW (1997b), Figure C-3, page C-8[e]
Vanadium	4.8	VO_3OH^{--}, HVO_4^{--}	EQ6[d] simulation of Alloy 22 corrosion
Zinc	63	Zn++	Solubility in repository water by EQ3[c] simulation

a. Concentration at the point where the chemical enters unsaturated zone water, controlled by solubility or local chemistry of dissolution and interaction with tuff. Note that these concentrations are not used for transport modeling (which is discussed in Section I.6) but are used only for screening analysis purposes. Refer to Section I.6 for groundwater concentrations of chemically toxic materials that were selected for further consideration based on the screening analysis.
b. To convert milligrams per liter to pounds per cubic foot, multiply by 0.00000624.
c. EQ6 code, Version 7.2b (Wolery and Daveler 1992, all).
d. EQ3 code, Version 7.2b (Wolery 1992, all).
e. For ph=8 and $Co_2=10^{-3}$ atmospheric partial pressure.

the "solubility." For example, the solubility of copper (from the bus bars left in the tunnels) would be obtained by increasing copper concentrations in successive runs of EQ3. At a concentration of 0.01811 milligram per liter, tenorite (CuO) would be saturated. This mineral would then be in equilibrium with dissolved copper existing in approximately equal molar parts as $CuOH^+$, $Cu(CO_3)aq$, and Cu^{++}. The aqueous concentration was then reported in Table I-11 as a "solubility" of copper for the purposes of screening the potentially toxic chemicals.

The largest quantities of potentially toxic materials come from the construction materials of the waste packages themselves. The main source is the Alloy-22 material used in the corrosion-resistant layer. The possible maximum concentrations of these materials (chromium, nickel, molybdenum, manganese, and vanadium) were developed by examining the corrosion process. Corrosion was modeled in the EQ6 code

(Wolery and Daveler 1992, all), starting with the same repository water as used in the solubility calculations described above. In the corrosion step, EQ6 was run in the titration mode (that is, a confined area in which essentially stagnant water reacts with iron from existing corrosion-allowance material fragments and Alloy-22). Oxygen is fixed at atmospheric fugacity (which is analogous to partial pressure adjusted for nonidealities). After a few hundred years, the chemistry of the resultant solution stays relatively constant for a long period. Following that, ionic strength eventually exceeds limits for +EQ6. The chemistry during this "flat period" was used as the resultant solution, which contained very high quantities of dissolved chromium (as hexavalent chromium), nickel, and molybdenum, and small dissolved quantities of manganese and vanadium. The reaction of this solution with tuff was then modeled. The resultant solution showed that essentially all of the nickel and manganese were precipitated and that the original dissolved concentrations of chromium, molybdenum, and vanadium remained.

Two types of geochemical analyses were performed. The first was an analysis of the solution concentration obtained when J-13 water, adjusted for the presence of repository materials such as concrete (that is, the same water chemistry used for other process modeling work supporting the Total System Performance Assessment-Viability Assessment), reacts with a large mass of carbon steel and Alloy-22 for an extended period. The second was an analysis of the reaction of the solution from the first analysis with volcanic tuff. The resultant solution from the second analysis would represent a bounding value for the source term solution at the floor of the emplacement drift.

At each step of the reaction progress in which the titration mode of EQ6 was used, a small quantity of reactants (steel and Alloy-22) was added to the solution (starting as J-13 water). After each addition, the increment of reactant dissolves and all product phases would reequilibrate with the aqueous solution. After a long time, this process would produce a bounding concentration for the solution. This would be the case if the water had a very long contact time with the metals and a very limited amount of water was used.

The composition of J-13 water was taken from earlier studies (TRW 1997b, page A-5). The carbon dioxide and oxygen levels are maintained at atmospheric conditions during the reaction. This process promotes the formation of the chromate (CrO_4^{--}) ion, which represents the hexavalent (and most toxic) state of chromium. The complete oxidation of chromium and the formation of chromate creates a very low pH environment in the area immediately adjacent to the corrosion process. The result of a low pH level in the presence of sufficient oxygen would be dissolved chromium existing in the hexavalent state. Large amounts of soluble hexavalent molybdenum are also formed.

Once the corrosion solution left the waste package, it would quickly encounter rock material. The second analysis evaluated the effect of rock on the solution. The analysis used the option for a "Fluid-Centered Flow-Through Open System" in EQ6. In this type of simulation the solution is permitted to react with solid materials (in this case, the tuff) for some specified interval (either time or reaction progress). The solution is then moved away from the solid reaction products that would be created and allowed to react with the same initial solids for a further interval. In this way, the model simulates reaction of the solution as it percolates through a rock.

This analysis simulated the tuff rock with the elemental composition characteristics of volcanic tuff. Earlier waste package criticality studies used this formulation for tuff reactants (TRW 1997c, page 17).

The resultant solution from the simulated reaction of J-13 water with carbon steel and Alloy-22 has a very low pH and a high concentration of dissolved chromium, molybdenum, and nickel. The resulting pH 2.0 solution would have the elemental concentrations listed in the second column of Table I-12. When the solution from corrosion contacts the rock, it would be neutralized to a pH of 8. The availability of silica in the rock would promote the formation of silicates, which would precipitate most of the nickel and manganese but virtually none of the chromium, molybdenum, or vanadium. Some chromium would change to $Cr_2O_7^{--}$ (still hexavalent and very soluble). The molybdenum would behave in a very similar

Table I-12. EQ6-modeled concentrations (milligrams per liter)[a] in solution from reaction of J-13 water with carbon steel and Alloy-22.

Element	After corrosion of Alloy-22	After reaction with tuff rock
Chromium	299	299
Manganese	32	4.40×10^{-11}
Molybdenum	218	218
Nickel	750	9.9×10^{-5}
Vanadium	4.8	4.8

a. To convert milligrams per liter to pounds per cubic foot, multiply by 0.00000624.

fashion and remain in solution as hexavalent species. The resultant solution would have the elemental concentrations listed in the third column of Table I-12.

The mechanism for mass loss of the Alloy-22 remains an issue at this time. There is no reliable evidence to support or refute the idea that the chromium that is carried away from Alloy-22 is dissolved hexavalent chromium. What is known fairly well is that trivalent chromium is the likely constituent (as Cr_2O_3) of the passivation film and that it has a very low solubility. It is not known whether the film grows thick until it sloughs off or if the film oxidizes in place so that it loses hexavalent chromium into solution. It is also not known if the film would oxidize and dissolve if it did slough off. EQ6 simulates a process whereby the trivalent chromium oxidizes to hexavalent chromium by reaction with O_2. It is well known that if chromium is in solution, the predominant species will be hexavalent chromium, especially in oxidizing conditions. At the Eh for atmospheric oxygen, it is known that the ratio of hexavalent chromium to For purposes of analysis, DOE assumes hexavalent chromium is mobilized as a dissolved constituent, and its source term is represented by 0.22 times the bulk loss rate of Alloy-22. A parallel assumption has been made about hexavalent molybdenum, which is also present in meaningful quantities in the results of the corrosion simulation.

I.3.2.3.2 *Well Concentration of Chemically Toxic Materials*

After the materials would begin to be released from the repository, they would be transported through the unsaturated zone to the saturated zone and on to the accessible environment. The screening study assumed that the chemicals would flow to a well from which an individual received all of their drinking water. Table I-13 lists the concentrations for the chemically toxic materials.

The well concentrations listed in Table I-13 were based on a series of simple calculations. First, the release concentrations for each material were calculated. The release rate for the material in the carbon steel is based on a degradation rate of 0.025 millimeter (0.001 inch) per year and a thickness of 100 millimeters (3.9 inches); thus, the annual fractional release rate for carbon steel is 0.00025. The degradation rate for Alloy-22 is 0.000006 millimeter (0.00000024 inch) per year and the material thickness is 20 millimeters (0.79 inch); the resulting annual fractional release rate is 0.0000003. The internal materials were assumed to be released at the same rate as the carbon steel (a conservative assumption). The release rate for the high-level radioactive waste was taken from earlier studies (TRW 1998f, Section 6.4). The annual fractional release rate for the high-level radioactive waste is 0.000054. The well concentrations in Table I-13 are very conservative concentration estimates that are not used directly for impact estimates. Instead, they are used to screen potentially toxic chemicals for more detailed analyses. These estimates were then compared to the Maximum Contaminant Levels for each material, if available (40 CFR 141.2). Some of the estimated concentrations were orders of magnitude below their respective Maximum Contaminant Levels. As a result of this screening study, barium, copper, lead, mercury, and selenium were eliminated from further detailed analysis. All the other chemically toxic materials, including boron, cadmium, chromium, manganese, molybdenum, nickel, uranium, vanadium, and zinc, were carried forward for further detailed analysis (see Chapter 5, Section 5.6.1).

Table I-13. Concentrations (milligrams per liter)[a] of waterborne chemically toxic materials for screening purposes.[b]

Element	Concentration limit	Release concentration: Non-package	Carbon steel	Alloy-22	Internal	HLW	Maximum	Well concentration	Maximum contaminant level[c]
Barium	0.00412	0	0	0	0	0.99	0.00412	1.5×10^{-5}	2.0
Boron	6,400	0	0	0	50	0	52	1.9×10^{-1}	NA[d]
Cadmium	23	0	0	0	0	2.2	2.2	7.7×10^{-3}	0.005
Chromium	300	0	0	2.7	940	0	300	1.1	0.1
Copper	0.018	0.018	0	0	0	0	0.018	6.4×10^{-5}	1.3
Lead	NA	0	0	0	0	0.09	0.09	3.2×10^{-4}	0.015
Manganese	4.4×10^{-11}	4.4×10^{-11}	707	0.077	0.44	0	4.4×10^{-11}	1.6×10^{-13}	NA
Mercury	NA	0	0	0	0	0.01	0.01	3.6×10^{-5}	0.002
Molybdenum	218	0	0	2.07	71	0	71	2.5×10^{-1}	NA
Nickel	1.0×10^{-6}	0	0	8.4	1,310	0	1.0×10^{-6}	3.5×10^{-9}	NA
Selenium	NA	0	0	0	0	0.014	0.014	4.9×10^{-5}	0.05
Uranium	0.0023	0	0	0	16,500	0	0.0023	8.2×10^{-6}	NA
Vanadium	4.8	0	0	0.054	0	0	0.054	1.9×10^{-4}	NA
Zinc	63	0	0	0	0.73	0	0.73	2.6×10^{-3}	NA

a. To convert grams per cubic meter to pounds per cubic foot, multiply by 0.00000624.
b. Note that these concentrations are not used for transport modeling (as discussed in Section I.6), but only for screening analysis purposes. Refer to Section I.6 for groundwater concentrations of chemically toxic materials that were selected for further consideration based on the screening analysis.
c. Maximum contaminant levels are specified in 40 CFR 141.2.
d. NA = not available (no Maximum Contaminant Level established by the U.S. Environmental Protection Agency for this element).

For the chemicals in the nonpackaged materials, the degradation was assumed to be limited by the solubility of the chemical in water. The release concentration (in grams per cubic meter) was assumed to be equal to the elemental solubility for those chemicals with a nonzero inventory in the nonpackaged materials. For the remaining material categories, all part of the waste packages, the release concentration was calculated based on the per-package inventory and the release rate from a waste package.

The per-package inventory (in grams for each material category) was calculated by dividing the total inventory (in grams) of the material type by the total number of waste packages in the repository (assumed to be 11,969). The release of material per cubic meter would be the fractional release rate divided by the rate of water flow past a waste package, based on an average 20-millimeter (0.79-inch) annual water flow rate through the repository. The release concentration is the per-package inventory in grams multiplied by the release per cubic meter.

To estimate the concentration in a well, two steps were performed. First, the maximum release concentration from the four material groups was selected. Then, two dilution factors were applied to the maximum release concentration. An unsaturated zone dilution factor was calculated as the ratio of the total cross-sectional area of all waste packages to the total surface area of the repository. Each of the 11,969 waste packages would have a cross-sectional area of 8.9 square meters (96 square feet), and the assumed repository surface area would be about 3 square kilometers (740 acres). This calculation resulted in an unsaturated zone dilution factor of 0.035. A dilution factor of 10 was applied to the saturated zone so the dilution factor, when combined for the unsaturated and saturated zones, would be 0.0035.

I.3.2.3.3 *Health Effects Screening for Chemically Toxic Materials*

The potential for human health impacts was estimated using a hazard index. The hazard index was determined by dividing the intake of a chemical by the *oral reference dose* for that chemical. A hazard index of 1.0 or above indicated the potential for human health impacts. Table I-14 lists the human health hazard indices.

ORAL REFERENCE DOSE

The *oral reference dose* is based on the assumption that thresholds exist for certain toxic effects such as cellular necrosis. This dose is expressed in units of milligrams per kilogram per day. In general, the oral reference dose is an estimate (with uncertainty spanning perhaps an order of magnitude) of a daily exposure to the human population (including sensitive subgroups) that is likely to be without an appreciable risk of deleterious effects during a lifetime (EPA 1999, all).

Table I-14. Human health hazard indices for chemically toxic materials.

Element	Intake (milligram per kilogram per day)	Oral reference dose[a] (milligram per kilogram per day)	Hazard index
Boron	0.0053	0.09	0.059
Cadmium	0.00022	0.0005	0.44
Chromium	0.030	0.005	6.1
Manganese	4.5×10^{-15}	0.14	3.2×10^{-14}
Molybdenum	0.0072	0.005	1.4
Nickel	1.0×10^{-10}	0.02	5.1×10^{-9}
Uranium	0.00000023	0.003	0.000078
Vanadium	0.0000054	0.007	0.00078
Zinc	0.000074	0.3	0.00025

a. Source: EPA (1999, all).

Intake was based on a 2-liter (0.53-gallon) daily consumption rate of drinking water, at the concentrations in the well, by a 70-kilogram (154-pound) adult. The oral reference doses were from the Integrated Risk Information System (EPA 1999, all), with the exception of doses for uranium (EPA 1994, all) and vanadium (International Consultants 1997, all).

Of the proposed chemically toxic materials in the repository, only chromium and molybdenum have a hazard index above 1.0. Because the inventories of a given material category in the repository should no more than double under any of the inventory modules, all chemically toxic materials (except chromium and molybdenum) can be eliminated from detailed analyses. However, the analysis also considered uranium in recognition of the special attention this element attracts and as a check for the screening analyses.

I.3.2.4 Chromium Inventory for Use in the Performance Assessment Model

The Alloy-22 that would comprise the inner corrosion-resistant material layer of the waste packages for the Yucca Mountain Repository design would contain 21.25 percent chromium and 55 percent nickel. In addition, stainless-steel containers and fuel cladding would contribute a meaningful but much smaller quantity of chromium. Table I-15 lists the chromium that would be present in the waste packages under the Proposed Action. Tables I-16 and I-17 list the chromium that would be present in the waste packages under Inventory Modules 1 and 2, respectively.

The performance assessment model simulates a number of abstracted waste packages for each waste category with a generalized inventory. Tables I-18 and I-19 summarize the assignment of the chromium inventory under the Proposed Action derived from the actual inventory listed in Table I-15 to the number of abstracted waste packages simulated with the model. The inventory is separated between interior stainless steel (Table I-18) and waste package Alloy-22 (Table I-19) because these two portions of the chromium inventory are modeled separately in a two-step process (see Section I.6 for details). Similarly, Tables I-20 and I-21 summarize the assignment of the chromium inventory derived from the actual inventory under Inventory Module 1, listed in Table I-16, to the number of abstracted waste packages

Table I-15. Chromium content (kilograms) of waste packages for the Proposed Action.[a]

Waste category	Waste package type[c]	Quantity actual waste packages[d]	Alloy-22 per waste package: Alloy mass	Alloy-22 per waste package: Chromium mass[e]	SS/B[b] alloy per waste package: Alloy mass	SS/B[b] alloy per waste package: Chromium mass[f]	Chromium mass per waste package type
Commercial spent nuclear fuel	21 PWR UCF (no absorber)	1,369	4,458	947	0	0	1,296,888
	21 PWR UCF (absorber plates)	2,641	4,458	947	1,883	546	3,944,056
	21 PWR UCF (control rods)	169	4,458	947	0	0	160,098
	12 PWR UCF (high heat)	394	3,282	697	0	0	274,785
	12 PWR UCF (South Texas)	179	3,717	790	1,071	311	196,981
	44 BWR UCF (no absorber)	773	4,261	905	0	0	699,923
	44 BWR UCF (absorber plates)	2,024	4,261	905	3,999	1,160	4,179,909
	24 BWR UCF (thick absorber)	93	3,342	710	2,141	621	123,789
High-level radioactive waste	5 HLW co-disposal	1,270	4,066	864	0	0	1,097,312
	5 HLW long co-disposal	1,007	5,687	1,208	0	0	1,216,947
DOE spent nuclear fuel	Navy SNF long	300[g]	6,306	1,340	0	0	381,907
Totals		**10,204**					**13,572,595**

a. To convert kilograms to pounds, multiply by 2.2046.
b. SS/B = stainless-steel boron.
c. Abbreviations: PWR = pressurized-water reactor; UCF = uncanistered fuel; BWR = boiling-water reactor; HLW = defense high-level radioactive waste; SNF = spent nuclear fuel.
d. Source: TRW (1999b, pages 6-5 to 6-12); quantities of waste packages modeled for results reported in Section I.6 differ slightly (because of the use of earlier estimates), resulting in a total chromium inventory about 1 percent less than indicated in this table. Final chromium impacts were not expected to differ because the inventory would not be exhausted during the period simulated.
e. Chromium constitutes 21.25 percent of Alloy-22.
f. Chromium constitutes 29 percent of SS/B alloy.
g. The analysis used 285 Navy SNF long waste packages in models for results discussed in Section I.6. The difference resulted in a chromium inventory that was about an additional 0.02 percent less than indicated in this table.

Table I-16. Chromium content (kilograms) of waste packages for Inventory Module 1.[a]

Waste category	Waste package type[c]	Quantity actual waste packages[d]	Alloy-22 per waste package: Alloy mass	Alloy-22 per waste package: Chromium mass[e]	SS/B[b] alloy per waste package: Alloy mass	SS/B[b] alloy per waste package: Chromium mass[f]	Chromium mass per waste package type
Commercial spent nuclear fuel	21 PWR UCF (no absorber)	2,339	4,458	947	0	0	2,215,793
	21 PWR UCF (absorber plates)	4,228	4,458	947	1,883	546	6,314,074
	21 PWR UCF (control rods)	314	4,458	947	0	0	297,460
	12 PWR UCF (high heat)	646	3,282	697	0	0	450,537
	12 PWR UCF (South Texas)	428	3,717	790	1,071	311	470,994
	44 BWR UCF (no absorber)	1,242	4,261	905	0	0	1,124,584
	44 BWR UCF (absorber plates)	3,195	4,261	905	3,999	1,160	6,598,226
	24 BWR UCF (thick absorber)	186	3,342	710	2,141	621	247,578
High-level radioactive waste	5 HLW co-disposal	1,557	4,066	864	0	0	1,345,287
	5 HLW long co-disposal	3,000	5,687	1,208	0	0	3,625,463
DOE spent nuclear fuel	Navy SNF Long	300	6,306	1,340	0	0	402,008
Totals		**17,435**					**23,092,003**

a. To convert kilograms to pounds, multiply by 2.2046.
b. SS/B = stainless-steel boron.
c. Abbreviations: PWR = pressurized-water reactor; UCF = uncanistered fuel; BWR = boiling-water reactor; HLW = defense high-level radioactive waste; SNF = spent nuclear fuel.
d. Source: TRW (1999b, pages 6-5 to 6-12); quantities of waste packages modeled for results reported in Section I.6 differ slightly (because of the use of earlier estimates), resulting in a total chromium inventory about 1 percent less than indicated in this table. Final chromium impacts were not expected to differ because the inventory would not be exhausted during the period simulated.
e. Chromium constitutes 21.25 percent of Alloy-22.
f. Chromium constitutes 29 percent of SS/B alloy.

Table I-17. Chromium content (kilograms) of waste packages for Inventory Module 2.[a]

Waste category	Waste package type[c]	Quantity actual waste packages[d]	Alloy-22 per waste package: Alloy mass	Alloy-22 per waste package: Chromium mass[e]	SS/B[b] alloy per waste package: Alloy mass	SS/B[b] alloy per waste package: Chromium mass[f]	Chromium mass per waste package type
Commercial spent nuclear fuel	21 PWR UCF (no absorber)	2,339	4,458	947	0	0	2,215,793
	21 PWR UCF (absorber plates)	4,228	4,458	947	1,883	546	6,314,074
	21 PWR UCF (control rods)	314	4,458	947	0	0	297,460
	12 PWR UCF (high heat)	646	3,282	697	0	0	450,537
	12 PWR UCF (South Texas)	428	3,717	790	1,071	311	470,994
	44 BWR UCF (no absorber)	1,242	4,261	905	0	0	1,124,584
	44 BWR UCF (absorber plates)	3,195	4,261	905	3,999	1,160	6,598,226
	24 BWR UCF (thick absorber)	186	3,342	710	2,141	621	247,578
High-level radioactive waste	5 HLW co-disposal	1,557	4,066	864	0	0	1,345,287
	5 HLW long co-disposal	3,000	5,687	1,208	0	0	3,625,463
DOE spent nuclear fuel	Navy SNF long	300	6,306	1,340	0	0	402,008
GTCC and SPAR[g]	5 HLW long co-disposal	608	5,687	1,208	0	0	734,760
Totals		**18,043**					**23,826,763**

a. To convert kilograms to pounds, multiply by 2.2046.
b. SS/B = stainless-steel boron.
c. Abbreviations: PWR = pressurized-water reactor; UCF = uncanistered fuel; BWR = boiling-water reactor; HLW = defense high-level radioactive waste; SNF = spent nuclear fuel.
d. Source: TRW (1999b, pages 6-5 to 6-12); quantities of waste packages modeled for results reported in Section I.6 differ slightly (because of the use of earlier estimates), resulting in a total chromium inventory about 1 percent less than indicated in this table. Final chromium impacts were not expected to differ because the inventory would not be exhausted during the period simulated.
e. Chromium constitutes 21.25 percent of Alloy-22.
f. Chromium constitutes 29 percent of SS/B alloy.
g. GTCC = Greater-Than-Class-C waste; SPAR = Special-Performance-Assessment-Required waste.

Table I-18. Modeled waste package interior chromium inventory for Proposed Action (kilograms).[a]

Waste category	Waste package type[b]	Mass per waste package type[c]	Mass per waste category	Number of abstracted waste packages	Mass per abstracted waste package
Commercial spent nuclear fuel	21 PWR UCF (no absorber)	0	3,902,762	7,760	503
	21 PWR UCF (absorber plates)	1,442,171			
	21 PWR UCF (control rods)	0			
	12 PWR UCF (high heat)	0			
	12 PWR UCF (South Texas)	55,596			
	44 BWR UCF (no absorber)	0			
	44 BWR UCF (absorber plates)	2,347,253			
	24 BWR UCF (thick absorber)	57,743			
High-level radioactive waste	5 HLW co-disposal	0	0	1,663	0
	5 HLW long co-disposal	0			
DOE spent nuclear fuel	Navy SNF long	0	0	2,546	0
Totals		**3,902,762**	**3,902,762**	**11,969**	

a. To convert kilograms to pounds, multiply by 2.2046.
b. Abbreviations: PWR = pressurized-water reactor; UCF = uncanistered fuel; BWR = boiling-water reactor; HLW = defense high-level radioactive waste; SNF = spent nuclear fuel.
c. Source: Table I-15.

Table I-19. Modeled corrosion-resistant material (Alloy-22) chromium inventory (kilograms) for Proposed Action.[a]

Waste category	Waste package type[b]	Mass per waste package type[c]	Mass per waste category	Number of abstracted waste packages	Mass per abstracted waste package
Commercial spent nuclear fuel	21 PWR UCF (no absorber)	1,296,888	6,973,667	7,760	899
	21 PWR UCF (absorber plates)	2,501,885			
	21 PWR UCF (control rods)	160,098			
	12 PWR UCF (high heat)	274,785			
	12 PWR UCF (South Texas)	141,385			
	44 BWR UCF (no absorber)	699,923			
	44 BWR UCF (absorber plates)	1,832,656			
	24 BWR UCF (thick absorber)	66,046			
High-level radioactive waste	5 HLW co-disposal	1,097,312	2,314,259	1,663	1,392
	5 HLW long co-disposal	1,216,947			
DOE spent nuclear fuel	Navy SNF long	381,907	381,907	2,546	150
Totals		**9,669,833**	**9,669,833**	**11,969**	

a. To convert kilograms to pounds, multiply by 2.2046.
b. Abbreviations: PWR = pressurized-water reactor; UCF = uncanistered fuel; BWR = boiling-water reactor; HLW = defense high-level radioactive waste; SNF = spent nuclear fuel.
c. Source: Table I-15.

Table I-20. Modeled waste package interior chromium inventory (kilograms) for Inventory Module 1.[a]

Waste category	Waste package type[b]	Mass per waste package type[c]	Mass per waste category	Number of abstracted waste packages	Mass per abstracted waste package
Commercial spent nuclear fuel	21 PWR UCF (no absorber)	0	6,262,475	12,932	484
	21 PWR UCF (absorber plates)	2,308,784			
	21 PWR UCF (control rods)	0			
	12 PWR UCF (high heat)	0			
	12 PWR UCF (South Texas)	132,933			
	44 BWR UCF (no absorber)	0			
	44 BWR UCF (absorber plates)	3,705,273			
	24 BWR UCF (thick absorber)	115,486			
High-level radioactive waste	5 HLW co-disposal	0	0	4,456	0
	5 HLW long co-disposal	0			
DOE spent nuclear fuel	Navy SNF long	0	0	4,340	0
Totals		**6,262,475**	**6,262,475**	**21,728**	

a. To convert kilograms to pounds, multiply by 2.2046.
b. Abbreviations: PWR = pressurized-water reactor; UCF = uncanistered fuel; BWR = boiling-water reactor; HLW = defense high-level radioactive waste; SNF = spent nuclear fuel.
c. Source: Table I-16.

simulated with the performance assessment model for interior stainless steel and corrosion-resistant material, respectively.

Inventory Module 2 is simulated as an incremental impact over Inventory Module 1, where the difference is in the Greater-Than-Class-C and Special-Performance-Assessment-Required wastes added under Inventory Module 2. Table I-22 summarizes the assignment of the additional chromium inventory derived from the actual inventory for Inventory Module 2 to the number of abstracted waste packages simulated with the performance assessment model. No interior stainless steel would be included in the additional waste packages under Inventory Module 2.

Table I-21. Modeled corrosion-resistant material (Alloy-22) chromium inventory (kilograms) for Inventory Module 1.[a]

Waste category	Waste package type[b]	Mass per waste package type[c]	Mass per waste category	Number of abstracted waste packages	Mass per abstracted waste package
Commercial spent nuclear fuel	21 PWR UCF (no absorber)	2,215,793	11,456,771	12,932	886
	21 PWR UCF (absorber plates)	4,005,290			
	21 PWR UCF (control rods)	297,460			
	12 PWR UCF (high heat)	450,537			
	12 PWR UCF (South Texas)	338,061			
	44 BWR UCF (no absorber)	1,124,584			
	44 BWR UCF (absorber plates)	2,892,953			
	24 BWR UCF (thick absorber)	132,093			
High-level radioactive waste	5 HLW co-disposal	1,345,287	4,970,749	4,456	1,116
	5 HLW long co-disposal	3,625,463			
DOE spent nuclear fuel	Navy SNF long	402,008	402,008	4,340	93
Totals		**16,829,528**	**16,829,528**	**21,728**	

a. To convert kilograms to pounds, multiply by 2.2046.
b. Abbreviations: PWR = pressurized-water reactor; UCF = uncanistered fuel; BWR = boiling-water reactor; HLW = high-level radioactive waste; SNF = spent nuclear fuel.
c. Source: Table I-17.

Table I-22. Additional corrosion-resistant material (Alloy-22) chromium inventory for Inventory Module 2 in excess of inventory for Module 1 (kilograms).[a]

Waste category	Waste package type[b]	Mass per waste package type[c]	Mass per waste category	Number of abstracted waste packages	Mass per abstracted waste package
GTCC+SPAR[d]	5 HLW long co-disposal	734,760	734,760	1,642	447

a. To convert kilograms to pounds, multiply by 2.2046.
b. Abbreviations: HLW = high-level radioactive waste.
c. Source: Table I-17.
d. GTCC = Greater-Than-Class-C waste; SPAR = Special-Performance-Assessment-Required waste.

I.3.2.5 Elemental Uranium Inventory for Use in the Performance Assessment Model

Table I-23 lists the total inventory of elemental uranium (that is, all isotopes of uranium) for consideration as a chemically toxic material for the Proposed Action and Inventory Modules 1 and 2. The total uranium inventory for both Inventory Modules 1 and 2 would be about 70 percent greater than that for the Proposed Action. The uranium content in high-level radioactive waste was set to the equivalent of metric tons of heavy metal (MTHM) for this analysis, though much of the uranium would have been removed during reprocessing operations. The elemental uranium inventory for Modules 1 and 2 would be essentially equivalent because Greater-Than-Class-C and Special-Performance-Assessment-Required wastes (the only additional waste in Module 2 over Module 1) do not contain substantial quantities of uranium.

I.3.2.6 Molybdenum Inventory

The Alloy-22 used for the corrosion-resistant material contains 13.5 percent molybdenum. During the corrosion of the Alloy-22, molybdenum behaves almost the same as chromium. Due to the corrosion conditions, molybdenum also dissolves in a highly soluble hexavalent form. Therefore, the source term for molybdenum will be exactly 13.5/21.25 times the source term for chromium (or 64 percent) from Alloy-22 only.

Table I-23. Total elemental uranium inventory (kilograms)[a] for Proposed Action and Inventory Modules 1 and 2.[b,c,d]

Inventory	Commercial SNF[e]	HLW[f]	DOE SNF	Totals
Proposed Action	63,000,000	4,700,000	2,300,000	70,000,000
Modules 1 and 2[g]	105,000,000	13,000,000	2,500,000	120,000,000

a. To convert kilograms to pounds, multiply by 2.2046.
b. The uranium content in high-level radioactive waste was set to the MTHM equivalent for this analysis, even though much of the uranium would have been removed during reprocessing operations.
c. Rounded to two significant figures.
d. Source: Appendix A, Tables A-12, A-13, A-19, A-29 to A-34.
e. SNF = spent nuclear fuel.
f. HLW = high-level radioactive waste.
g. Inventory Module 1 and 2 will have the same total uranium inventory because Greater-Than-Class-C and Special-Performance-Assessment-Required waste (the only additional waste in Module 2 over Module 1) does not contain a substantial quantity of uranium.

I.3.3 ATMOSPHERIC RADIOACTIVE MATERIALS

The only radionuclide that would have a relatively large inventory and a potential for gas transport would be carbon-14. Iodine-129 can exist in a gas phase, but it is highly soluble and therefore likely to dissolve in groundwater rather than migrate as a gas. After carbon-14 escaped from the waste package, it could flow through the rock in the form of carbon dioxide. About 2 percent of the carbon-14 in commercial spent nuclear fuel occurs in a gas phase in the space (or *gap*) between the fuel and the cladding around the fuel (Oversby 1987, page 92). The gas-phase inventory consists of 0.23 curie of carbon-14 per commercial spent nuclear fuel waste package. Table I-24 lists the total carbon-14 inventory for the repository under the Proposed Action and Inventory Modules 1 and 2.

Table I-24. Total carbon-14 inventory (curies).[a]

Inventory	Solid[b]	Gaseous[c]	Totals[d]
Proposed Action	92,000	1,800	93,000
Module 1	150,000	3,200	160,000
Module 2	240,000	3,200	240,000

a. Source: Appendix A, Table A-10.
b. Impacts of carbon-14 in solid form are addressed as waterborne radioactive material impacts.
c. Based on 0.234 curie of carbon-14 per commercial spent nuclear fuel waste package.
d. Totals are rounded to two significant figures.

I.4 Extension of Total System Performance Assessment Methods and Models for EIS Analyses

DOE conducted analyses for the Total System Performance Assessment – Viability Assessment to evaluate potential long-term impacts to human health from the release of radioactive materials from the Yucca Mountain Repository. The analyses for this EIS were conducted in conjunction with, but distinct from, the calculations for the Viability Assessment (DOE 1998a, Volume 3, all). The methodologies and assumptions for the Viability Assessment are detailed in TRW (1998a,b,c,d,e,f,g,h,i,j,k, all). Extensions of the Viability Assessment analyses to meet distinct EIS requirements (for example, consideration of different thermal load scenarios or inventories) were made using the same overall methodology, and details of these extensions are provided in this section. Additional information on EIS performance-assessment analyses can be found in TRW (1999a, all).

I.4.1 REPOSITORY DESIGN FOR ALTERNATIVE THERMAL LOADS

The spatial density at which the waste packages are emplaced in the repository is generally quantified using *thermal load*, which is the MTHM emplaced per acre of repository area. The higher the thermal load, the smaller the spacing between waste packages, resulting in a higher thermal output per unit area.

The area required for emplacement is based on the target thermal loads attained by varying the spacing between the waste packages and the distance between the emplacement drifts. The commercial spent nuclear fuel heat output dominates the overall heat load and thus the total emplacement area required. Thus, for purposes of thermal modeling, the Proposed Action inventory implies the nominal value of 63,000 MTHM commercial spent nuclear fuel, whereas Inventory Modules 1 and 2 have the same expanded inventory of 105,000 MTHM commercial spent nuclear fuel.

Table I-25 gives the estimates of repository area required for the emplacement of wastes, ranging from a low of 740 acres for the high thermal load scenario with the Proposed Action inventory case to a high of 4,200 acres for the low thermal load scenario with the Inventory Module 1 or 2 case. Most of the options require waste emplacement in areas beyond the primary, or *upper,* emplacement block, which is juxtaposed between the Solitario Canyon Fault and the Ghost Dance Fault. The upper emplacement block is the reference repository region in the Viability Assessment base case facility design (63,000 MTHM high thermal load scenario). Selection of potential expansion blocks near the upper block was carried out using several criteria:

- Availability of 200 meters (660 feet) of overburden
- Consistency of elevation and dip with the upper block
- Distance from the saturated zone
- Favorable excavation characteristics

These considerations are described in detail in TRW (1999b, all).

Table I-25. Estimates of repository emplacement area.[a]

Thermal load (MTHM per acre)	Drift spacing (meters)[c]	Area (acres)[b] Proposed Action	Area (acres)[b] Inventory Modules 1 and 2
85	28	740	1,240
60	40	1,050	1,750
25	38[d]	2,520	4,200

a. Source: TRW (1999a, Table 2.3-1, page 2-12) based on 63,000 MTHM of commercial spent nuclear fuel.
b. To convert acres to square miles, divide by 640.
c. To convert meters to feet, multiply by 0.3048.
d. Under the low thermal load, the waste packages would be placed in an approximately square pattern so that the thermal load was distributed evenly. To accomplish this, the emplacement drift spacing and the spacing of the waste packages in the emplacement drift would be approximately equal (TRW 1999c, page F-2).

The selected inventory layouts for the Proposed Action and Inventory Modules 1 and 2 for the high, intermediate, and low thermal load scenarios are shown in Figures I-2 through I-7. These layouts, simplified from the original engineering layouts presented in TRW (1999c, Figures 3.3-1 through 3.3-6), indicate that the wastes for these thermal loads can be accommodated within the upper blocks, the lower block, and one additional region (Block 1a) to the west of the Solitario Canyon Fault.

As described in TRW (1999c, all), additional subsurface blocks for emplacement of waste according to intermediate and low thermal load scenarios were identified by:

- Expanding the upper block to the north and south
- Expanding the lower block to the north and east
- Lowering the elevation of Block 1a, combining it with Block 1b, and designating the combined area as Block 5
- Raising the elevation of Block 2 by 15 meters (50 feet) and designating it as Block 6
- Raising the elevation of Block 3 by 12 meters (39 feet) and designating it as Block 7
- Raising the elevation of Block 4 by 2 meters (6.6 feet), extending the area to the south, and designating it as Block 8

The corresponding layouts for the low thermal load scenario for the Proposed Action and for Inventory Modules 1 and 2 are shown in Figures I-6 and I-7, respectively. Figure I-8 shows the relationship between the early Proposed Action designs and the design areas considered in these EIS analyses.

I.4.2 THERMAL HYDROLOGY MODEL

Evaluation of the intermediate (60 MTHM per acre) and low (25 MTHM per acre) thermal load scenarios for this EIS diverged from the high thermal load base case evaluated in the Viability Assessment. Extensions of the thermal-hydrologic modeling supporting the total systems performance assessment model were required to evaluate these additional thermal load scenarios. These extensions are detailed in this section.

I.4.2.1 Thermal-Hydrologic Scenarios

The analysis of waste package degradation and engineered barrier system release for the EIS requires information regarding waste package temperature and relative humidity, and liquid saturation and temperature within the repository invert. These data were derived from the development and application of a suite of three-dimensional, drift-scale models for predicting the thermal-hydrologic environment near the waste packages. Six sets of calculations were carried out to handle the two inventory options (63,000 and 105,000 MTHM) and the three thermal load scenarios (85, 60, and 25 MTHM per acre). The simulations were performed using NUFT, an integrated finite-difference code capable of modeling multidimensional fluid flow, solute migration, and heat transfer in porous and/or fractured media (Nitao 1998, all).

These calculations closely parallel the thermal-hydrologic modeling study performed in support of Total System Performance Assessment – Viability Assessment (TRW 1998c, all). The main difference between the two studies is in the treatment of thermal-hydrologic conditions at the edge of the repository. In Total System Performance Assessment – Viability Assessment, a hybrid methodology with complementary thermal-hydrologic and thermal conduction models is used to delineate different thermal-hydrologic zones within the repository horizon (TRW 1998c, all). In this study, a less detailed scaling methodology is used to divide the repository into center and edge regions because of the computational complexities associated with larger inventories and expanded emplacement regions. This less detailed scaling methodology is not expected to adversely impact the results.

I.4.2.2 Waste Package and Drift Geometry

Following the approach taken in Total System Performance Assessment – Viability Assessment, the basic three-dimensional drift-scale model was developed around a discrete waste package symmetry element. This model extends:

- In the x-direction, from the drift centerline to the midpoint between adjacent drifts
- In the y-direction, over a representative number of packages to capture the package-to-package variability in heat output
- In the z-direction, from the ground surface to the water table

The vertical discretization between the ground surface and the water table was chosen to be consistent with the Lawrence Berkeley National Laboratory three-dimensional, site-scale unsaturated flow model (Bodvarsson, Bandurraga, and Wu 1997, all). The basis for the model discretization in the other two dimensions is described in the following paragraphs.

The Proposed Action inventory consists of 63,000 MTHM of commercial spent nuclear fuel, 4,667 MTHM of high-level radioactive waste, and 2,333 MTHM of DOE spent nuclear fuel. As described in DOE (1998a, Volume 3, Figure 3-18, page 3-31), the corresponding symmetry element contains seven packages:

- Three 21-pressurized-water-reactor waste packages
- Two 44-boiling-water-reactor waste packages
- One-half of a 12-pressurized-water-reactor waste package
- One-half of a direct-disposal waste package (containing four DOE spent nuclear fuel N-reactor canisters)
- One co-disposal waste package (containing five high-level radioactive waste glass-filled canisters with or without a DOE spent nuclear fuel canister)

Inventory Module 1 consists of 105,000 MTHM of commercial spent nuclear fuel, 12,600 MTHM of high-level radioactive waste (based on MTHM equivalency discussion in Section A.2.3.1 of Appendix A of this EIS), and 2,500 MTHM of DOE spent nuclear fuel. Accordingly, the expanded inventory symmetry element was created using a total of nine packages:

- Three and one-half 21-pressurized-water-reactor waste packages
- Two and one-half 44-boiling-water-reactor waste packages
- One 12-pressurized-water-reactor waste package
- Two co-disposal waste packages containing five high-level radioactive waste glass-filled canisters (with or without a DOE spent nuclear fuel canister)

Note that this symmetry element model maintains the relative percentage (and heat output) of different package types while minimizing the total number of discrete packages for computational convenience. This package discretization model was deemed adequate from the standpoint of thermal-hydrologic modeling, although it is only an approximation of the true inventory.

For the high (85 MTHM per acre) and intermediate (60 MTHM per acre) thermal load scenarios, the waste package arrangement within the drifts was kept constant, and the drift spacing was adjusted to attain the correct thermal load levels. Thus, the high thermal load scenario yields drift spacing of 28 meters (about 92 feet) and the intermediate thermal load scenario yields drift spacing of 40 meters (about 130 feet). For the low (25 MTHM per acre) thermal load scenario, maintaining the same waste package arrangement as for the high and intermediate thermal load scenarios would have required the drifts to be spaced too far apart in the x-direction, resulting in localized heating effects. Therefore, the package-to-package spacing in the y-direction was increased for the low thermal load scenario to create an approximately square symmetry element, including drift spacing of 38 meters (about 120 feet). Waste package spacing for the Proposed Action and for Inventory Modules 1 and 2 is summarized in Table I-26 and Table I-27, respectively.

Table I-26. Waste package spacing for the Proposed Action inventory.[a]

		Spacing of gap after given package (meters)[b]	
Waste package type	Waste package width (meters)	High and intermediate thermal load	Low thermal load
12-PWR	½ (5.87)	6.021	26.424
21-PWR	5.3	9.276	31.215
21-PWR	5.3	2.949	15.415
Co-disposal	5.37	2.2535	13.676
21-PWR	5.3	8.929	30.345
44-PWR	5.3	7.98	27.969
44-BWR	5.3	1.305	11.2996
Direct-disposal	½ (5.37)		

a. Source: TRW (1999a, Table 3.2-1, page 3-3).
b. To convert meters to feet, multiply by 0.3048.

Table I-27. Waste package spacing for Inventory Modules 1 and 2.[a]

		Spacing of gap after given package (meters)[b]	
Waste package type	Waste package width (meters)	High and intermediate thermal load	Low thermal load
21-PWR	½ (5.3)	2.949	11.3055
Co-disposal	5.37	2.2535	17.79
21-PWR	5.3	9.95	32.902
21-PWR	5.3	10.02	33.081
21-PWR	5.3	7.39	26.9175
12-PWR	5.87	6.368	24.3615
44-PWR	5.3	7.98	27.969
44-BWR	5.3	1.305	12.599
Direct-disposal	5.37	1.305	10.0
44-BWR	½ (5.3)		

a. Source: TRW (1999a, Table 3.2-2, page 3-4).
b. To convert meters to feet, multiply by 0.3048.

I.4.2.3 Selection of Submodels

Engineering layouts developed for waste emplacement were shown in Figures I-2 through I-7. These layouts suggest that multiple, discontinuous heated regions will develop in the postclosure period for some of the options. A full three-dimensional representation of all heated regions (such as emplacement areas) was not considered computationally practical. Therefore, for modeling purposes each region was treated as an isolated entity by assuming that boundaries existed for no heat flow and no fluid flow between the regions. Furthermore, to capture the effects of varying stratigraphy and variable surface infiltration on the thermal-hydrology response at the repository, each emplacement block was modeled by a representative stratigraphic column or submodel. These submodel solution assumptions are unlikely to affect adversely the results reported in this EIS.

Based on the original design layouts (see Figure I-2), each thermal load scenario was to be modeled using some combination of each of the following seven stratigraphic columns:

- Upper Block (stratigraphic column 1)
- Lower Block (stratigraphic column 2)
- Block 1a (stratigraphic column 3)
- Block 1b (stratigraphic column 7)
- Block 2 (stratigraphic column 5)
- Block 3 (stratigraphic column 6)
- Block 4 (stratigraphic column 4)

These submodels were used for the high and intermediate thermal load scenarios. However, because of the large areal extent required for the low thermal load scenario, the engineering layout changed for those two design options. In the new design layout, Block 1b has been combined with part of Block 1a to form Block 5, while part of Block 1a has been combined with Block 4 to form Block 8. These two new areas can be represented by two existing submodels: stratigraphic column 7 for Block 5 and stratigraphic column 4 for Block 8. This information is summarized in Table I-28 and shown on Figure I-8.

Table I-28. Areas of submodels (stratigraphic columns) used in thermal-hydrologic calculations.[a]

Thermal-hydrologic scenario	Loading (MTHM per acre)	Waste package inventory module	Emplacement block	Stratigraphic column number	Actual area (acres)	Percent of area
1	85	Proposed Action	Upper Block	1	740	100.0
2	60	Proposed Action	Upper Block	1	1,050	100.0
3	25	Proposed Action	Upper Block	1	1,110	44.0
			Lower Block	2	596	23.7
			Block 5	7	814	32.3
4	85	Inventory	Upper Block	1	1,180	95.5
		Modules 1 and 2	Lower Block	2	55	4.5
5	60	Inventory	Upper Block	1	1,180	67.4
		Modules 1 and 2	Lower Block	2	380	21.7
			Block 1a	3	190	10.9
6	25	Inventory	Upper Block	1	1,110	26.4
		Modules 1 and 2	Lower Block	2	596	14.2
			Block 5	7	814	19.4
			Block 6	5	420	10.0
			Block 7	6	440	10.5
			Block 8	4	820	19.5

a. Source: TRW (1999a, Table 3.2-3, page 3-5).

For all submodels, the vertical stratigraphic data for the model stratigraphic columns were extracted from the Lawrence Berkeley National Laboratory site-scale model (Bodvarsson, Bandurraga, and Wu 1997, all), with the exception of Block 2 and Block 3, which lie outside the boundaries of the site-scale model. The geologic framework model (TRW 1997d, all) was used to develop the stratigraphy for the columns corresponding to Block 2 and Block 3 even though very little information is available regarding the stratigraphy, hydrology, and infiltration conditions in this sector of the Yucca Mountain site. Thermal-hydrologic simulations were carried out with these two submodels for the low thermal load with expanded inventory scenario, but the simulations were not used for the subsequent total-system calculations. It was assumed that the thermal-hydrologic results for these regions could be approximated by the neighboring regions within the Berkeley model domain. Thus, the submodel for Block 8 (stratigraphic column 4) was assumed also to represent Block 3, and the submodel for Block 5 (stratigraphic column 7) was assumed also to represent Block 2.

I.4.2.4 Hydrology and Climate Regime

Hydrologic properties for the thermal-hydrologic models were taken to be the same as the Total System Performance Assessment – Viability Assessment base case (TRW 1998c, Section 3.5). These properties include matrix and fracture characteristics describing capillary retention and relative permeability for a dual-permeability model, including fracture-matrix-interaction area-reduction factor terms that were adjusted to match observed borehole saturations. As described in RamaRao, Ogintz, and Mishra (1998, pages 116 to 118), the dual-permeability model parameters have been adjusted for the present study using the "satiated saturation" concept in the generalized equivalent continuum model. Using a porosity-weighted average, the dual-permeability model fracture and matrix parameters (porosity and permeability) are combined to create corresponding parameters for the generalized equivalent continuum model, while the satiated saturation concept is used to set the threshold for the initiation of flow in fractures (before the attainment of full matrix saturation). Subsequently, the composite medium capillary characteristics are generated by a porosity-weighted average of the individual media curves. These hydrologic properties, as well as other thermal properties used in the thermal-hydrologic calculations, are discussed in TRW (1998c, Section 3.2.1, pages 3-21 to 3-26).

This EIS performance assessment considered three climate scenarios: *present-day, long-term average* (wetter than the present-day climate), and *superpluvial,* which are added at short-duration, fixed intervals on a periodic basis during the 100,000-year period after waste emplacement. In the performance assessment model, the initial conditions (that is, the present-day climate) are multiplied by 5.45 to obtain the long-term average climate and by 14.30 to obtain the super-pluvial climate (DOE 1998a, Volume 3, Figure 4.2, page 4-4). The climate changes are measured in step-changes for the duration of the climate periods, and the sequence lengths are 10,000 years for the present-day dry climate and the super-pluvial climate, and 90,000 years for the long-term average climate. The sequence of climate changes used for expected-value simulations (which use the mean value of probabilistically defined input variables) is:

- 0 to 5,000 years - present-day (dry) climate
- 5,001 to 95,000 years - long-term average climate
- 95,001 to 105,000 years - present-day (dry) climate
- 105,001 to 195,000 years - long-term average climate
- 195,001 to 205,000 years - present-day (dry) climate
- 205,001 to 285,000 years - long-term average climate
- 285,001 to 295,000 years - super-pluvial climate
- 295,001 to 305,000 years - present-day (dry) climate

This sequence is repeated for the duration of the simulation period.

Expected-value simulations were carried out for the first 1 million years after closure, to include the complete decay of waste heat caused by radioactive decay and a return to ambient conditions. To establish appropriate initial conditions for the thermal-hydrologic simulations, the nominal present-day (dry) climate scenario, as used in the Viability Assessment base case (TRW 1998c, Section 3.5), was used for the ambient hydrologic calculations. A separate set of thermal-hydrologic simulations was then performed for each climate condition, as required. This approach is consistent with that used in the Viability Assessment, in which climate effects on thermal hydrology for the entire period were included by making three sets of calculations (for present-day, long-term average, and superpluvial climates). The influence of climate change on thermal-hydrologic system response was then approximated in the performance assessment model total-system simulator by switching from one set of results to the other at the time of climate change.

For both the present-day and long-term average climate, the infiltration flux at the top of each representative column was extracted from the flux associated with the nearest element in the Lawrence Berkeley National Laboratory site-scale model (Bodvarsson, Bandurraga, and Wu 1997, all). However, there was no infiltration information available for stratigraphic columns 5 and 6, which are located outside the Berkeley model boundary. Therefore, the infiltration fluxes for these columns were assumed to be equal to the fluxes at the nearest element within the Berkeley model boundary. Note that these infiltration rates were assumed to be constant throughout the 1-million-year postemplacement period with climate changes implemented by multiplying the infiltration rate as described above.

I.4.2.5 Treatment of Edge Effects

The drift-scale modeling results, developed using a representative symmetry element with periodic lateral boundary conditions, best represents the conditions at the center of the repository. To account for the edge-cooling effects experienced by exterior drifts located near unheated rock mass, a scaling methodology was developed based on the hypothesis that the repository can be divided into at least two thermal-hydrologic regions for grouping waste packages, a center region and an edge region. The center region was designed so periodic boundary conditions (no-flow thermal and hydrologic boundaries) could be assigned in a lateral direction. The edge region has a more complicated response because of edge-cooling effects. However, it is believed that the thermal-hydrologic response at the edge is similar to that for the center, albeit at a lower thermal load. Thus, the objective of the scaling methodology was two-fold:

1. Devise a strategy for generating the thermal load scale factors so models representative of the center can be used to simulate the edge response.
2. Estimate the fraction of the repository area enclosed within the center or edge regions.

The following sections briefly describe the development and testing of the components of this scaling methodology.

I.4.2.5.1 *Scaling Factors for Edge Effects*

Based on the conceptual model that the edge response is similar to the center response at a lower thermal load, two-dimensional results from an east-west cross-section scale model of the mountain were compared to a set of one-dimensional runs representing the edge at a series of different thermal loads. The objective was to find a scaling factor for the thermal loads which would provide agreement between the two-dimensional and one-dimensional runs with respect to (1) time history of temperature, liquid saturation, and the mass fraction of air at the repository horizon; and (2) vertical profiles of temperature, liquid saturation, and the mass fraction of air at different points in time.

These calculations were carried out for the base case hydrologic properties and infiltration regime described earlier. The selection of the optimal scaling factor was performed by visual examination and restricted to one scaling factor for the early-time period (0 to 1,000 years) and a second scaling factor for the late-time period (1,000 years to 100,000 years).

Figure I-9 shows the comparison between the two-dimensional and one-dimensional model results using scale factors of 0.8 and 0.6. This comparison suggests that a scale factor of 0.8 is more appropriate for the early-time period, and a scale factor of 0.6 is more suitable for the late-time period. Although not shown here, examining vertical profiles of the primary variables at two different points in time (100 years and 10,000 years) yielded similar observations. Note that a single scaling factor can only provide a gross average match of all stated variables; thus, the match between two-dimensional and scaled one-dimensional results is never perfect. Furthermore, categorization of only two scale factors (early-time and late-time periods) is primarily for computational convenience. These simplifications notwithstanding, the scaling methodology appears to be a reasonable and practical strategy for generating the edge response without resorting to more complex three-dimensional models containing both heated drifts and unheated rock mass.

I.4.2.5.2 *Definition of Thermal-Hydrologic Zones*

The spatial division of the repository into center and edge regions is based on the approximation of the diffusive temperature profile at the repository by a step function. The temperature profile at selected time steps was extracted and fitted with equivalent step functions. The fraction of area enclosed within the temperature discontinuity was then taken as the fraction of repository belonging to the center region. This process is schematically demonstrated for the high thermal load scenario in Figure I-10.

The fractional areas were found to be time-dependent. For the high thermal load scenario, the thermal-hydrologic response is nearly the same for the entire repository as long as the boiling period is active. Thereafter, for all practical purposes, the fraction belonging to the center stabilizes at about 0.66 (this is the recommended fraction to be used at all times for waste package degradation calculations). For the intermediate thermal load scenario, the fractional area belonging to the center region is found to be close to unity at early- and late-time periods, dropping to approximately 0.6 at intermediate times. Therefore, a time-averaged value of 0.8 is recommended as the fractional area belonging to the center for this thermal load. Edge effects are not considered important for the low thermal load scenario, because the use of multiple emplacement blocks will tend to elevate the temperature between adjacent blocks, thus minimizing edge-cooling effects.

I.4.2.6 Results

As mentioned earlier, thermal-hydrologic modeling results in the form of waste package temperature and relative humidity are required for waste package degradation calculations in WAPDEG. In addition, temperature and liquid saturation within the invert supporting the waste packages is required for Engineered Barrier System release calculations in the repository integration program model. Such information is extracted from NUFT output files and archived in tabular form for input to WAPDEG and the repository integration program model. In this section, a brief discussion of the sensitivity of the thermal-hydrologic simulation results to various design options and natural-system uncertainties will be presented.

I.4.2.6.1 *Variability Among the Waste Packages*

Figures I-11 and I-12 show the temperature and relative humidity histories for the various waste package types for the Proposed Action inventory at high and low thermal loads, respectively. For the high thermal load scenario, the highest peak temperature would result from the use of the 21-pressurized-water-reactor design package, whereas the lowest peak temperature would result from the use of the direct disposal package. These peaks differ by approximately 80°C (176°F). The temperature history for the 21-pressurized-water-reactor average waste package falls near the middle of this range. Note, however, the convergence in temperature and relative humidity for all packages as the temperature drops below the nominal boiling point [100°C (212°F)]. The small differences in temperature and relative humidity histories for the waste packages from this time onward would not affect the WAPDEG-predicted package degradation rates in a meaningful manner. Therefore, results from only the 21-pressurized-water-reactor average waste package are provided as representative inputs to WAPDEG.

I.4.2.6.2 *Sensitivity to Thermal Loads*

Figure I-13 shows the temperature and relative humidity histories for the three thermal loads and both Proposed Action and Inventory Modules 1 and 2 scenarios. As expected, the relative peak temperatures correspond to the magnitude of the thermal loads. For each thermal load, the expanded inventory gives a slightly higher peak temperature result, but the two inventories converge quickly at later times. Calculations for the high and intermediate thermal load scenarios result in similar curves, both in terms of temperature and relative humidity. For the low thermal load scenario, the shape of the curve is much flatter and the temperature drops below 100°C (212°F) much earlier than the other scenarios.

I.4.2.6.3 *Comparison Between Center and Edge Locations*

Figure I-14 shows a comparison between temperature and relative humidity histories calculated for the high thermal load scenario using both center and edge models. The edge model is essentially the center model with a lower heat load. As described in Section I.4.2.5, the heat flux for the center model is scaled by 0.8 prior to 1,000 years and by 0.6 after 1,000 years, to provide the thermal input for the edge model. As expected, the temperature history for the edge model falls below, and the relative-humidity history lies above, the response for the center model.

I.4.3 WASTE PACKAGE DEGRADATION MODEL

Evaluation of Inventory Modules 1 and 2 for this EIS diverged from the Proposed Action, or base case, inventory evaluated in the Viability Assessment. Extensions of the waste package degradation modeling supporting the total systems performance assessment model were required to evaluate the additional inventories. These extensions are detailed in this section.

One component of the EIS and Total System Performance Assessment – Viability Assessment performance assessments pertains to quantifying the degradation of the metallic waste packages. A waste package would be a double-walled disposal container consisting of an outer 10-centimeter (4-inch)-thick layer of carbon steel (the corrosion-allowance material), and an inner 2-centimeter (0.8-inch)-thick layer of chromium-molybdenum Alloy-22 (the corrosion-resistant material) (DOE 1998a, Volume 3, page 3-74). A statistically based waste package degradation numerical code, WAPDEG (TRW 1998l, all), was developed to quantify the ranges in expected degradation of the waste packages. The corrosion rates for the corrosion-allowance materials and corrosion-resistant materials included in the code were abstracted from several sources (TRW 1998e, pages 5-11 to 5-16). The development of WAPDEG indicated that the major environmental factors in waste package degradation were temperature and moisture availability. These data were input into WAPDEG after conducting thermal-hydrologic modeling to establish the temperature and relative humidity histories, as described in Section I.4.2.

I.4.3.1 WAPDEG Development and Application to Total System Performance Assessment – Viability Assessment

The EIS WAPDEG calculations were based on the Total System Performance Assessment – Viability Assessment model configuration of this code (TRW 1998e, page 5-3). The performance assessment analysis conducted for the Total System Performance Assessment – Viability Assessment considered a repository thermal load of 85 MTHM per acre, with the base case waste inventory of 63,000 MTHM commercial spent nuclear fuel and 7,000 MTHM DOE spent nuclear fuel and high-level radioactive waste. Numerical thermal-hydrologic modeling was conducted to generate transient temperature and relative humidity histories within the emplacement drift. These histories were then used as input into the WAPDEG code to determine the time of initiation, type, and rate of waste package corrosion during a 100,000-year simulation. The WAPDEG simulations generated a suite of waste package failure distributions that were incorporated into the Total System Performance Assessment – Viability Assessment model.

Two corrosion modes were implemented by the WAPDEG code for each waste package, general corrosion and localized corrosion. These modes were applicable to both the corrosion-allowance-material outer wall/barrier and the corrosion-resistant-material inner wall/barrier. The conditions under which the corrosion modes applied in WAPDEG depended primarily on temperature, relative humidity, the geochemistry of the water, and the presence or absence of dripping or pooled water.

The corrosion-allowance material undergoes general corrosion according to one of two models, a humid-air corrosion model and an aqueous corrosion model, depending on the relative humidity at the waste package surface. Both models are based on statistical analysis of corrosion data observed for carbon-steel corrosion (DOE 1998a, Volume 3, pages 3-81 to 3-82). However, neither corrosion model will be applicable if the temperature at the waste package surface is too high. The thermal calculations for the potential repository typically show an initial postclosure increase in repository temperature due to radioactive decay, followed by a cooling period that eventually reaches ambient temperature. Laboratory and modeling studies indicate that general corrosion of the corrosion-allowance material can only start when the temperature cools to a value near the boiling point of water (DOE 1998a, Volume 3, page 3-82). The temperature-dependent corrosion data are input into the model and applied to waste packages based on a user-defined temperature threshold either in the form of a fixed value or a probability distribution that is sampled for each package.

Relative humidity generally increases as the temperature cools and vaporized moisture condenses. If the relative humidity is sufficiently high and the temperature threshold is met, the corrosion-allowance material can undergo humid-air corrosion. An input to the model is the relative humidity threshold sufficient for initiation of humid-air general corrosion either as a fixed value or a probability distribution that is sampled for each package.

The relative humidity may rise sufficiently to cause a thin film of water to form on the waste package surface. At that point, the aqueous corrosion model more appropriately describes general corrosion. The relative humidity threshold is input either as a fixed value or a probability distribution that is sampled for each package. When the relative humidity exceeds the threshold, WAPDEG transitions from the humid-air corrosion model to the aqueous corrosion model.

Neither general corrosion model for corrosion-allowance materials is expected to behave in a uniform manner over the entire waste package surface. WAPDEG includes a provision for nonuniform corrosion in two ways; it discretizes the waste package surface into segments called *patches* with roughness factors applied to each patch. The number of patches per waste package and the roughness factors are input, with the latter either as a fixed value or a probability distribution. WAPDEG obtains a statistical sample of the distribution (if provided) to be used for each patch on the package. The product of the general corrosion depth at a given time and the roughness factor gives the total corroded depth at a particular location on the patch at that time. When the corroded depth at any point on a patch equals or exceeds the thickness of the corrosion-allowance material, WAPDEG assumes that the patch has failed.

When a patch is breached on the corrosion-allowance material, WAPDEG assumes that part of the surface area of the corrosion-resistant material is then subject to corrosion. In fact, there is a one-to-one correspondence of patches for corrosion-allowance material and corrosion-resistant material. Even though only a fraction of the corrosion-allowance material patch may be breached, the crevice between the two materials will likely grow over time to allow water and air to access the entire corrosion-resistant material patch. WAPDEG conservatively assumes that the entire area of this patch is immediately subject to corrosion upon breach of its overlying corrosion-allowance material patch.

The general corrosion of the two materials differs due to the composition of the two waste package wall materials. The general corrosion rate applied by WAPDEG to the corrosion-resistant material was derived from data gained from the Waste Package Degradation Expert Elicitation. A compilation of the elicited results was then used to create a cumulative distribution function for general corrosion rates of corrosion-resistant materials at temperatures of 25°C, 50°C, and 100°C (77°F, 122°F, and 212°F, respectively) (DOE 1998a, Volume 3, pages 3-85 to 3-88). WAPDEG samples a corrosion rate from each cumulative distribution function for a package in such a manner that, if the points were joined on a plot comparing corrosion rates and temperatures, the curve for a waste package is parallel to the curves for all the other waste packages. When WAPDEG encounters a temperature between the specified temperatures, it linearly interpolates the logarithm of the corrosion rate versus the reciprocal of the temperature to estimate the corrosion rate at the given temperature.

According to a follow-up question for the Waste Package Degradation Expert Elicitation, the spread of the general corrosion rates at a given temperature was due to a combination of uncertainty and natural variability. Waste Package Degradation Expert Elicitation panelists estimated the Alloy-22 general corrosion rate and the allocation of the total variance to its variability and uncertainty. The effect of the corrosion rate variability among waste packages, patches, and the corrosion rate uncertainty on waste package failure and, ultimately, radiological dose was evaluated by splitting the total variance into three different variability and uncertainty combinations: 75-percent variability and 25-percent uncertainty; 50-percent variability and 50-percent uncertainty; and 25-percent variability and 75-percent uncertainty. Uncertainty was interpreted as the uncertainty of the mean of the distribution. To capture this uncertainty, a given percentage was used to establish three possible values for the mean which were based on the 5th, 50th, and 95th percentiles of the uncertainty about the global mean. Three uncertainty splits, combined with these three estimates of the mean, produced nine new cumulative distribution functions for general corrosion rate, which implied nine WAPDEG runs. These runs are summarized in Table I-29.

Table I-29. Uncertainty/variability splitting sets for corrosion rate of corrosion-resistant material.[a]

	Uncertainty/variability splitting ratios		
Percentile	25% and 75%	50% and 50%	75% and 25%
5th	Set 1	Set 2	Set 3
50th	Set 4	Set 5	Set 6
95th	Set 7	Set 8	Set 9

a. Source: TRW (1999a, Table 3.3-1, page 3-12).

In the presence of water or water vapor, localized corrosion could occur on the corrosion-resistant material in the form of pitting or crevice corrosion. Information from the Waste Package Degradation Expert Elicitation indicates that localized corrosion would begin only if the temperature was sufficiently high. The user supplies the temperature threshold for initiating pitting either in the form of a fixed value or a probability distribution that is sampled for each waste package. If pitting is allowed to begin as the result of sufficient water and heat levels, WAPDEG implements an Arhennius model for pit growth. Thus, the corrosion-resistant material could be breached either by the general corrosion of patches on the waste package surface or by pit penetration. WAPDEG output files indicate the number of patch failures and pit penetrations over time for each waste package.

The local environment in the waste-emplacement areas could differ from package to package, a factor treated as variability in WAPDEG. To implement this concept, WAPDEG assumes that the variances of the probability distributions that describe general corrosion are due to spatial variability and the variances should be allocated. Using the treatment described above for splitting the cumulative distribution functions for general corrosion of the corrosion-resistant material, the variance of each of the resulting nine distributions is due to natural variability. Some variance accounts for package-to-package variability, and the rest accounts for variable conditions along a waste package (patch-to-patch variability). The user supplies the fraction of variance to be shared by the waste packages, and the remaining fraction is applied to patches. In the Viability Assessment analysis, variance between packages and between patches is 35 percent/65 percent for patches dripped on and 50 percent/50 percent otherwise.

In practice, WAPDEG samples a corrosion parameter using the global distribution but with only a fraction of its variance. The sampled value is then treated as the mean value for the patches on that waste package. For each patch, WAPDEG samples the distribution using the waste package mean and the remaining variance. The results are used to model general corrosion for the patch. WAPDEG also applies this variance-sharing technique to the general corrosion of the corrosion-allowance material and to the temperature threshold for pitting initiation on the corrosion-resistant material.

One difference between waste package environments would be the presence or absence of dripping or pooled water. WAPDEG allows the user to specify the fraction of patches that contact such water, either as a fixed value or using a probability distribution. The user can also specify when drips start, stop, or experience a change in water chemistry. For dripping conditions, model inputs can be used to specify roughness factors on the corrosion-allowance material, the cumulative distribution functions of general corrosion rates for corrosion-resistant material, and all the temperature and relative humidity thresholds as different from those for nondripping conditions. WAPDEG determines if an individual patch is dripped on or not and uses the appropriate model parameters.

For the Total System Performance Assessment – Viability Assessment configuration, waste package failure distributions were generated based on always-dripping or no-dripping conditions. For each infiltration (I) case where I varied from I multiplied by 3 to I divided by 3 (I, I × 3, and I ÷ 3), nine simulations were conducted based on the always-dripping corrosion rates. Because of the small number of failures for the no-dripping case, only one case was simulated (Set 6).

I.4.3.2 Application of WAPDEG for the EIS

This EIS analyzes the effects of three different thermal loads (high, intermediate, and low) and three waste inventories (Proposed Action, Inventory Module 1, and Inventory Module 2) to determine their impact, if any, on total system performance. The comparison of thermal output versus time for the Inventory Module 1 and Inventory Module 2 waste inventories were considered identical for the thermal-hydrologic modeling (see Section I.4.2). Therefore, only the Proposed Action inventory and Inventory Module 1 (the expanded inventory) were considered.

Section I.4.2 describes the number of repository regions that were simulated depending on the thermal load requirements for each scenario. To incorporate the potential cooling effects around the edges of a repository region, some regions were simulated using a conceptualized center and edge, resulting in multiple NUFT simulations for certain regions. Table I-30 lists the number of individual simulations conducted for each thermal load/inventory combination, for each climate scenario.

As with the Total System Performance Assessment-Viability Assessment analyses, only the long-term average climate scenario was used in the EIS WAPDEG simulations. Therefore, the six thermal-hydrologic scenarios listed in Table I-30 were used in the generation of an equal suite of WAPDEG simulations that assumed long-term average infiltration conditions. Table I-30 lists 18 total individual thermal-hydrologic simulations for the six scenarios. WAPDEG simulations were performed using the temperature and relative humidity histories generated from each of the 18 simulations. Each set of WAPDEG simulations consisted of nine always-dripping and one no-dripping case, based on uncertainty/variability splitting.

The EIS analyses used one always-dripping case and the no-dripping base case input files from the Total System Performance Assessment – Viability Assessment as starting points. The EIS models used the same corrosion model configuration and the same corrosion rate probability distribution functions as those used in the Total System Performance Assessment – Viability Assessment base case configuration. However, the EIS analysis used a lower, fixed relative humidity threshold for corrosion initiation of the corrosion-resistant material than that used in the Total System Performance Assessment – Viability Assessment analysis. The threshold used in the EIS analysis is based on a better understanding of the factors that initiate corrosion. This difference resulted in an earlier estimate of failure of the corrosion-resistant material for the EIS analysis. This earlier failure is evident in the results of the 10,000-year analysis but does not affect the 1-million-year analysis.

Table I-30. Thermal-hydrologic and waste package degradation simulation matrix.[a]

Thermal-hydrology scenario	Inventory module	Thermal load (MTHM per acre)	Repository block(s)	Stratigraphic column	Block simulation location	WAPDEG simulation number
1	Proposed Action	85	Upper Block	1	Center	1-10
					Edge	11-20
2	Proposed Action	60	Upper Block	1	Center	21-30
					Edge	31-40
3	Proposed Action	25	Upper Block	1	Center	41-50
			Lower Block	2	Center	51-60
			Block 5	7	Center	61-70
4	Inventory Modules 1 and 2	85	Upper Block	1	Center	71-80
					Edge	81-90
			Lower Block	2	Center	91-100
					Edge	101-110
5	Inventory Modules 1 and 2	60	Upper Block	1	Center	111-120
			Lower Block	2	Center	121-130
			Block 1a	3	Center	131-140
6	Inventory Modules 1 and 2	25	Upper Block	1	Center	141-150
			Lower Block	2	Center	151-160
			Block 8	4	Center	161-170
			Block 5	7	Center	171-180

a. Source: TRW (1999a, Table 3.3-2, page 3-13).

Each WAPDEG run generated a failure curve that contained a probability distribution function of the first corrosion-resistance-material breach, average pit failures, and average patch failures (as a function of time). These files were transferred to the repository integration program model.

I.4.3.3 Results

Figure I-15 shows the temperature and drift relative humidity history curves, respectively, for all three thermal loads (high, intermediate, and low) with the Proposed Action inventory. Figure I-16 shows the temperature and relative humidity history curves, respectively, for all three thermal load scenarios with the expanded inventory (Inventory Modules 1 and 2). These figures show that when the temperature threshold [100°C (212°F)] for corrosion initiation is met, the relative humidity within the drifts for most of the runs is within the range of aqueous corrosion (80 to 100 percent). The time to reach the temperature threshold is less for the low thermal load scenario (less than 100 years) than for the high and intermediate thermal load scenarios (200 to 700 years). Corrosion of the corrosion-allowance material for the low thermal load scenario is initiated sooner but only by a few hundred years. This difference will become relatively small when discussing the differences in package failure rates at times greater than 10,000 years.

The thermal histories generated from the thermal-hydrologic modeling indicate that the hottest and coolest thermal histories correspond to the high thermal load, expanded-inventory scenario and the low thermal load, Proposed Action inventory scenarios, respectively. Thus, the results from these two configurations bound the range of potential WAPDEG failure responses. In addition, the waste package failure results were dominated by the packages that were dripped on; therefore, the failure results for the packages that were not dripped on are not presented.

WAPDEG simulations for the low thermal load with Proposed Action inventory case were generated for three repository regions corresponding to the upper (primary) block, lower block, and Block 5. The thermal output for this layout did not include edge effects (see Section I.4.2); therefore, only one thermal simulation per repository block was generated. Temperature and relative humidity histories generated from each repository block were used to define the conditions within the drifts. Figure I-17 shows the time to first breach or failure of the corrosion-allowance material for the always-dripping packages in each of the three emplacement blocks. The failures of the corrosion-allowance material are very similar for all three stratigraphic columns, with failures starting at approximately 800 years and extending approximately 4,000 years. Figure I-18 shows the time to first breach of the corrosion-resistant material for the always-dripping packages in each of the three emplacement blocks, for each of the nine uncertainty/variability splitting sets (defined in Table I-29). The failure of the corrosion-resistant material barriers in the three regions were very similar, given the same uncertainty/variability splitting set (set 5). For example, the responses observed for stratigraphic columns 2 and 7 overlie each other. The variability in the failure of the corrosion-resistant material in a particular region (for example, stratigraphic column 1), due to the introduction of the uncertainty/variability splitting, ranges from a few thousand years (set 7) to no failures within 1 million years (set 3).

Given the relatively cool thermal history for the low thermal load scenario and the 70,000 MTHM inventory, no pits (localized corrosion) would penetrate through the corrosion-resistant material for the always-dripping packages for all three emplacement blocks. All failures (see Figure I-18) would be due to general corrosion because the temperature threshold for localized corrosion was not reached. Figure I-19 shows the average number of patches penetrated through the corrosion-resistant material as a function of time for the always-dripping packages, all three emplacement blocks, and the uncertainty/variability splitting sets. Figures I-20 through I-22 show that the variability in the results of the failure for the three emplacement blocks is dominated by the corrosion rate uncertainty/variability splitting of corrosion-resistant material, with little variability attributed to the different thermal-hydrologic inputs.

WAPDEG simulations for the high thermal load scenario with the expanded inventory were generated for the upper (primary) and lower repository blocks. The repository blocks were simulated with both a center and an edge region (see Section I.2). Figure I-20 shows the time to first breach or failure of the corrosion-allowance material for the always-dripping packages, for all four simulations, and the uncertainty/variability splitting sets. Figure I-21 shows the time to first breach of the corrosion-resistant material. Figure I-22 shows the average number of patches penetrated through the corrosion-resistant material as a function of time. Previous analyses have shown that the releases from the waste packages are dominated by advection through the patch. Therefore, the patch failure history is a representative indicator of the overall performance. The results shown in Figure I-22 also show that the variability in the failures for the four center and edge simulations is dominated by the uncertainty/variability splitting, with little variability attributed to the different thermal-hydrologic inputs.

These results show that the variability in the corrosion-resistant material failures as a function of time has a greater dependency on the variability/uncertainty splitting associated with the corrosion-resistant material corrosion rate than on the variation in the temperature and relative humidity histories. The results for the high and intermediate thermal load scenarios for the Proposed Action inventory and the intermediate and low thermal load scenarios for the expanded inventory simulations showed similar behavior to the results discussed above.

I.4.3.4 Discussion

Corrosion of the corrosion-allowance material is not initiated until the waste package temperature decreases below the thermal threshold selected for the model [100°C (212°F)]. For the majority of the thermal-hydrologic simulations conducted for the EIS, once the thermal threshold is satisfied, the humid-air corrosion is initiated. Figure I-23 shows the time to the first breach of the corrosion-allowance material for all expected-value always-dripping WAPDEG simulations (Set 5). The time to first breach of the corrosion-allowance material is earliest for the low thermal load scenarios as expected from the temperature profiles shown in Figures I-15 and I-16. Because the thermal threshold is satisfied sooner, corrosion of the corrosion-allowance material is initiated sooner.

Figure I-23 also shows that by 5,000 years, almost each waste package has had at least a single corrosion-allowance material failure, thereby allowing corrosion of corrosion-resistant material. Figure I-24 shows the time to the first breach of the corrosion-resistant material for all expected-value always-dripping WAPDEG simulations (Set 5). The first corrosion-resistant-material breach for most scenarios occurs between 20,000 to 30,000 years, with the high thermal load, expanded-inventory scenario having a very low fraction of packages failing within 10,000 years. Figure I-24 also shows that the higher thermal loads generate the earliest corrosion-resistant material failures, even with later corrosion-allowance material failures. This behavior is due to the temperature-dependent, corrosion-resistant-material corrosion models, which have higher corrosion rates at higher temperatures. The thermal profiles in Figures I-23 and I-24 show that temperature is lower for the lower thermal load scenarios, resulting in slower corrosion rates and delayed failure relative to the higher loads.

Figure I-25 shows the average number of patches that failed per package as a function of time for all thermal loads and inventories, all regions, always-dripping, and uncertainty/variability splitting (set 9). Figure I-26 shows the average number of patches that failed per package as a function of time for all thermal loads and inventories, all regions, always-dripping, and uncertainty/variability splitting (set 5). These plots show a factor-of-five difference between the failure results for the two different uncertainty/variability-splitting sets.

The degradation results show that for each thermal-hydrologic scenario, the variability in the failures due to the uncertainty/variability splitting in the corrosion rate of the corrosion-resistant material would be considerably greater than the variability due to the different thermal histories. Therefore, for each thermal and inventory scenario, a set of -failure distributions from a single region was selected and included in the RIP model simulations.

I.4.4 WASTE FORM DISSOLUTION MODELS

Evaluation of Inventory Modules 1 and 2 for this EIS diverged from the Proposed Action, or base case, inventory evaluated in the Viability Assessment. Specifically, additional waste forms were included in Inventory Modules 1 and 2 that were not considered in the Viability Assessment base case, and waste form dissolution models were required to model these additional waste forms. Extensions of the waste form dissolution modeling that supported the Total Systems Performance Assessment model were required to evaluate the additional inventories. These extensions are detailed in this section.

I.4.4.1 Spent-Fuel Dissolution Model

A semi-empirical model for intrinsic dissolution (alteration) rate of the spent fuel matrix was developed from experimental data (TRW 1995, page 6-2). If the postclosure environment inside the potential repository can be assumed to maintain the atmospheric oxygen partial pressure of 0.2 atmosphere (TRW 1995, page 6-1), the dissolution model becomes a function of temperature, total carbonate concentration, and pH of contacting water. The dissolution rate strongly depends on temperature and total carbonate concentration but is less influenced by pH. The spent fuel dissolution rate increases with temperature and is enhanced by the total carbonate concentration of the contacting water, although to a smaller extent than by temperature. The mixed oxide spent nuclear fuel from plutonium disposition was modeled as commercial spent nuclear fuel.

I.4.4.2 High-Level Radioactive Waste Glass

As in the spent fuel alteration/dissolution modeling discussed above, the entire surface area of defense high-level radioactive waste in the glass waste form is assumed to be exposed to the near-field environment as soon as the first pit penetrates the waste package. The waste forms are assumed to be covered by a "thin" water film when the water contacts the glass and the alteration/dissolution processes are initiated. The "can-in-canister" ceramic from plutonium disposition was modeled as high-level radioactive waste.

High-Level Radioactive Waste Glass Dissolution Model

Details concerning the intrinsic glass dissolution rate model, as a function of temperature and pH, are presented along with rate data (TRW 1995, pages 6-4, 6-5, and 6-37). The relationship indicates that the rate model represented by the equation predicts a monotonically increasing dissolution rate with temperature.

This dissolution conceptualization contains several assumptions and limitations. The radionuclides are assumed to be released as fast as the glass structure breaks down, which is a conservative assumption because it does not account for solubility-limited radionuclides. No credit is taken for the fact that "experiments have shown that the actinides more commonly are included in alteration phases at the surface of the glass either as minor components of other phases or as phases made up predominantly of actinides" (TRW 1995, page 6-5). The model includes neither solution chemistry (other than pH and dissolved-silica concentration) nor vapor-phase alteration of the glass. Glass has been observed to undergo hydration in a humid environment and, on subsequent contact with water, radionuclide releases from a hydrated glass layer were several orders of magnitude higher than those from an unhydrated (fresh) glass waste form (TRW 1995, page 6-5).

I.4.4.3 Greater-Than-Class-C and Special-Performance-Assessment-Required Waste

The alteration/dissolution processes for Greater-Than-Class-C and Special-Performance-Assessment-Required waste forms were assumed to be similar to those for high-level radioactive waste glass.

I.4.5 RIP MODEL MODIFICATIONS

The EIS RIP model simulations are based on the Total System Performance Assessment – Viability Assessment (Revision 1) base case RIP model (TRW 1998n, all). To perform the EIS performance assessment analyses, the base case model was modified primarily to allow input of the different repository areas corresponding to the thermal load scenarios and the expanded waste inventories of Modules 1 and 2, and the repository-block configurations used in the thermal-hydrologic modeling. The EIS analysis also considered the impact to individuals at distances other than the 20 kilometers (12 miles) used for the Viability Assessment. Therefore, the analysis expanded the saturated-zone convolution model used in the Viability Assessment to include development of convolution stream tubes from the repository to distances of 30 kilometers (19 miles) and 80 kilometers (50 miles) and postprocessing of the 20-kilometer output to extract the radiological dose to individuals at the 5-kilometer (3-mile) distance described in Section I.4.5.4. This section describes the modifications. Knowledge and understanding of the RIP model (Golder 1998, all) and the Viability Assessment model (TRW 1998a,b,c,d,e,f,g,h,i,j,k, all) are necessary to fully understand the differences discussed in this section.

I.4.5.1 Modifications to the RIP Model in the Repository Environment

The RIP model conceptualization for the Yucca Mountain Repository performance assessment considers waste forms in discrete regions of the repository as source terms for flow and transport. The RIP model conceptualization for the Viability Assessment considered the primary repository block, corresponding to the high thermal load scenario, to be comprised of six regions. For any particular case analyzed for the

EIS, the EIS thermohydrologic simulations were used to determine the number of repository regions used. In adapting the Viability Assessment base case as the model for the EIS analyses, the repository regions had to conform to the center/edge model conceptualization. For each of the unused Viability Assessment regions, the source terms (commercial spent nuclear fuel, high-level radioactive waste, and DOE spent nuclear fuel) and all associated RIP model cells were removed from the model, and the remaining source terms and associated connecting cells were adapted to the center/edge model. In all cases, a total of 60 concentration parameters and all of the "connection" groups, except the 10 groups that provided total radiological dose at various points, were removed from the model. Then, the new region-specific connection groups were added as appropriate to account for the calculation of advective and diffusive releases from the center and edge regions of the EIS simulations. The calculated flux data, developed from the Lawrence Berkeley National Laboratory hydrologic model of the repository area (Bodvarsson, Bandurraga, and Wu 1997, all), was used to modify the flux into and fluid saturations applicable to the various source terms in the EIS RIP model.

Another modification resulted from the fact that although the Total System Performance Assessment – Viability Assessment considered sensitivity variations in the infiltration to the repository, the EIS simulations used only the infiltration (I) option. This was done to reduce the number of calculations, because the three thermal loads and two extra inventories greatly multiplied the number of cases to be simulated. The (I × 3) and (I × 3) options of the Total System Performance Assessment – Viability Assessment were not considered. Therefore, only the WAPDEG results for the "always-dripping" and "no-drip" scenarios were selected for model input. This change resulted in appropriate changes to the fraction-of-packages-failed parameters to allow the appropriate (I) WAPDEG to be incorporated into the model. To accommodate these differences to the RIP model, the fraction-of-packages-failed parameters for the (I × 3) and (I × 3) options were redirected to call the applicable WAPDEG tables for the long-term average climate case. The effect of neglecting this variation is minor. Sensitivity studies with the Viability Assessment model for the high thermal load scenario (DOE 1998a, Volume 3, pages 5-3 to 5-5) showed that the 10,000-year peak dose is actually decreased by 30 percent for the I × 3 case, while the peak is moved back from 10,000 years to about 5,000 years and the 1-million-year peak dose is increased about 30 percent.

The EIS simulations used only one thermal table rather than the six used in the Viability Assessment base case. Therefore, the thermal parameters were updated to refer to only one unique thermal table for each of the thermal load scenarios and inventory combinations:

- High thermal load, Proposed Action inventory
- Intermediate thermal load, Proposed Action inventory
- Low thermal load, Proposed Action inventory
- High thermal load, Inventory Modules 1 and 2
- Intermediate thermal load, Inventory Modules 1 and 2
- Low thermal load, Inventory Modules 1 and 2

The thermal hydrology modeling indicated that a single invert saturation was sufficient for all regions and all layers of the invert. Based on this information, all invert saturation parameters were fixed to a value of 0.993.

I.4.5.2 Modifications to Input and Output FEHM Model

The particle-tracking files used in the Viability Assessment (TRW 1998g, all) were modified for each EIS case to allow a different number of FEHM input regions to be used, depending on the number of input regions used in the engineered barrier system model. The "Zone 6" interface file was modified for each EIS case by changing the FEHM nodes to be used for input of mass from the engineered barrier system. The FEHM nodes were chosen to correspond to the coordinates of the EIS repository emplacement blocks. For the low thermal load scenario for Inventory Modules 1 and 2 shown in Figure I-7, proposed Blocks 6 and 7 fell outside the model boundaries. To allow the unsaturated zone particle tracker in the FEHM model to account for all mass in the repository, the mass from areas 6 and 7 were allocated to Blocks 5 and 8, respectively. Figures I-27 through I-32 show the repository emplacement blocks used for each case.

The "Zone 6" interface file was also modified for each EIS case by defining the saturated zone area that would capture the mass coming out of the FEHM model. It was necessary to modify the capture regions in order to ensure inclusion of all of the mass and to distribute the mass amongst the six stream tubes based on its repository emplacement block of origin. For the high and intermediate thermal load scenarios with Proposed Action inventories, the same regions were used for this EIS as were used for the Viability Assessment base case (Figure I-33). Figure I-34 shows the capture regions used for the low thermal load scenario with the Proposed Action inventory; the low thermal load scenario with Inventory Modules 1 and 2, and the intermediate thermal load scenario with Inventory Modules 1 and 2. Figure I-35 shows the capture regions used for the high thermal load scenario with Inventory Modules 1 and 2.

I.4.5.3 Modifications to Saturated Zone Stream Tubes for Different Repository Areas

The saturated zone stream tubes consist of a unit-breakthrough curve and a scaling factor. The unit-breakthrough curves are all the same for a given radionuclide at a given distance. The scaling factor is the product of the flux coming from the repository and a dilution factor. The dilution factor is a lumped parameter that is used to account for mixing and lateral dispersion. For the multiple-realization cases, the dilution factor is assumed to have lognormal distribution with a mean value of ten.

In order to use the stream tubes for different repository regions, flux multiplier values were calculated for each stream tube. The flux multiplier value is the ratio of the new flux into a stream tube to the flux into that stream tube in the base case (Proposed Action inventory, high thermal load scenario). The saturated zone module of RIP requires the concentration of water entering the saturated zone from the unsaturated zone, so the water flux at this interface is needed to compute the mass concentration of contaminants in the water. The resulting flux multiplier is used to scale the water flux predicted by the FEHM transport module in RIP to properly account for the larger capture zone areas for other cases. Each stream tube is associated with one of the unsaturated zone capture regions described above. The flux into a given stream tube is the sum of the fluxes from the repository regions that are in that capture region. The high thermal load scenario with Proposed Action-inventory used the same fluxes as the Viability Assessment base case. Tables I-31 and I-32 list the contribution to each of the stream tubes from each of the repository areas for the intermediate and low thermal load scenarios with Proposed Action inventory, respectively. The same information is provided for the high, intermediate, and low thermal load scenarios with Inventory Modules 1 and 2 inventory, respectively, in Tables I-33 through I-35. The fluxes used in these tables were obtained from the results of the base case Lawrence Berkeley National Laboratory site-scale unsaturated zone flow model (Bodvarsson, Bandurraga, and Wu 1997, all).

Table I-31. Summary of fluxes (cubic meters per year) from repository area to convolution stream tubes for intermediate thermal load scenario with Proposed Action inventory.[a]

Stream tube	Flux from each repository area into each stream tube				Total flux	85-MTHM-per acre, base case inventory flux	Flux multiplier
	Upper block	Lower block	Blocks 5 & 6	Blocks 7 & 8			
1	6,410	0	0	0	6,410	3,162	2.03
2	3,480	0	0	0	3,480	3,482	1.00
3	3,990	0	0	0	3,990	3,993	1.00
4	4,060	0	0	0	4,060	4,060	1.00
5	8,090	0	0	0	8,090	10,103	0.801
6	5,320	0	0	0	5,320	2,077	2.56

a. Source: TRW (1999a, Table 3.5-1, page 3-19).

Table I-32. Summary of fluxes (cubic meters per year) from repository area to convolution stream tubes for low thermal load scenario with Proposed Action inventory.[a]

	Flux from each repository area into each stream tube						
Stream tube	Upper block	Lower block	Blocks 5 & 6	Blocks 7 & 8	Total flux	85-MTHM-per acre, base case inventory flux	Flux multiplier
1	16,570	0	0	0	16,570	3,162	5.24
2	16,570	0	0	0	16,570	3,482	4.76
3	0	5,250[b]	0	0	5,250[b]	3,993	0.131
4	0	5,250[b]	0	0	5,250[b]	4,060	0.129
5	0	0	6,750	0	6,750	10,103	0.668
6	0	0	6,750	0	6,750	2,077	3.25

a. Source: TRW (1999a, Table 3.5-2, page 3-19).
b. Typographical error in source document.

Table I-33. Summary of fluxes (cubic meters per year) from repository area to convolution stream tubes for high thermal load scenario with Inventory Modules 1 and 2.[a]

	Flux from each repository area into each stream tube						
Stream tube	Upper block	Lower block	Blocks 5 & 6	Blocks 7 & 8	Total flux	85-MTHM-per acre, base case inventory flux	Flux multiplier
1	7,050	0	0	0	7,050	3,162	2.23
2	7,050	0	0	0	7,050	3,482	2.02
3	7,050	0	0	0	7,050	3,993	1.77
4	7,050	0	0	0	7,050	4,060	1.74
5	7,050	0	0	0	7,050	10,103	0.698
6	0	969	0	0	969	2,077	0.466

a. Source: TRW (1999a, Table 3.5-3, page 3-20).

Table I-34. Summary of fluxes (cubic meters per year) from repository area to convolution stream tubes for intermediate thermal load scenario with Inventory Modules 1 and 2.[a]

	Flux from each repository area into each stream tube						
Stream tube	Upper block	Lower block	Blocks 5 & 6	Blocks 7 & 8	Total flux	85-MTHM-per acre, base case inventory flux	Flux multiplier
1	17,620	0	0	0	17,620	3,162	5.57
2	17,620	0	0	0	17,620	3,482	5.06
3	0	3,350	0	0	3,350	3,993	0.838
4	0	3,350	0	0	3,350	4,060	0.824
5	0	0	0	4,090	4,090	10,103	0.404
6	0	0	0	4,090	4,090	2,077	1.97

a. Source: TRW (1999a, Table 3.5-5, page 3-20).

Table I-35. Summary of fluxes (cubic meters per year) from repository area to convolution stream tubes for low thermal load scenario with Inventory Modules 1 and 2.[a]

	Flux from each repository area into each stream tube						
Stream tube	Upper block	Lower block	Blocks 5 & 6	Blocks 7 & 8	Total flux	85-MTHM-per acre, base case inventory flux	Flux multiplier
1	17,620	0	0	0	17,620	3,162	5.57
2	17,620	0	0	0	17,620	3,482	5.06
3	0	5,250	0	0	5,250	3,993	1.31
4	0	5,250	0	0	5,250	4,060	1.29
5	0	0	10,240	0	10,240	10,103	1.01
6	0	0	10,240	54,200	64,440	2,077	31.0

a. Source: TRW (1999a, Table 3.5-4, page 3-20).

REPOSITORY SIZE AND SATURATED ZONE DILUTION FACTORS

Increasing repository size could cause either a reduction or no change in the relative lateral dispersive effects of saturated zone transport. Consider a rectangular repository oriented normal to the direction of flow in the saturated zone. The cross-sectional area of the resultant contaminant plume at a downstream well would be larger than that at the cross-sectional area of the plume at the source (below the repository), causing dilution of the radionuclide concentration at the downstream well. However, if the area of the repository was doubled, the plume at the exposure location would increase, but by less than twice. Hence, lower dilution factors would occur for larger repositories. Analytical modeling provides quantification for lower dilution factors.

The validity of using lower dilution factors for larger repositories can be illustrated by considering two hypothetical repositories with equal waste inventory, one having twice the emplacement area of the other. The concentration at the base of the unsaturated zone below the larger repository would be - half the concentration below the smaller repository (a direct result of different spacing of the waste). Using a one-dimensional saturated zone transport model without dilution, for times far greater than the groundwater travel time, the concentrations at a downstream well would be equal to those at the base of the unsaturated zone (provided the contaminant release was continuous). If the same dilution factor was applied in both cases, the downstream well concentrations for the larger repository would be half those in the smaller repository. On the other hand, if the repository was treated as a point source in each case, the dilution factor for the larger repository would be half that of the smaller repository, resulting in equal concentrations at a downstream well. These two outcomes correspond to two alternative ways of doubling the repository area. Thus, the dilution factors for expanded area repositories can be lower or equal to those of the base-case repository.

I.4.5.4 Modifications to the Stream Tubes for Distances Other Than 20 Kilometers

One-dimensional stream-tube runs for the saturated zone were conducted for generating unit-breakthrough curves at distances of 30 and 80 kilometers (19 and 50 miles) downstream from the repository. This was accomplished using the Los Alamos National Laboratory simulator FEHM (Zyvoloski et al. 1995, all) and developing a finite-element mesh that extended beyond the 25-kilometer (16-mile) mesh previously used to develop the 20-kilometer (12 mile) stream tube used for the Viability Assessment. The sets of transport parameters used in the previous model runs were also applied in the extended mesh simulations for distances up to 25 kilometers. Beyond 25 kilometers, the model properties were made identical to those assigned to the undifferentiated valley fill. On completing the FEHM runs for each of nine radionuclides, model output was postprocessed to take into account mass loadings from the unsaturated zone to each of six different stream-tube capture areas and to adjust model results for dilution attributed to transverse dispersion. This last step involved the determination of distance-dependent dilution factors by using dilution information previously developed from exposure concentrations at the 20-kilometer distance. An analytical transport solution in the program 3DADE (Leij, Scaggs, and van Genuchten 1991, all) was used to determine dispersion coefficients that resulted in dilution factors of 10, 50, and 100 at 20 kilometers and to determine corresponding dilution factors at distances of 30 and 80 kilometers. The resulting data indicated a logarithmic relationship between the 20-kilometer dilution factors and those occurring at the longer distances, making it possible to determine appropriate dilution parameters used in postprocessing of the extended-distance FEHM runs.

The saturated zone transport in the Viability Assessment is essentially based on a one-dimensional analysis that precludes lateral dispersion in the y and z directions. To simulate the realistic results of three-dimensional transport, the results of the one-dimensional analysis are divided by a dilution factor. Thus, the dilution factor accounts for attenuation of concentrations caused by the spread of the contaminant plume as the result of lateral dispersion. The dilution factor approximates numerical dispersion for the one-dimensional saturated zone model, as can be achieved using a three-dimensional advective-dispersive numerical model. This simulates the real dilution in the system.

The Viability Assessment dilution factors were based on the results of the Expert Elicitation Panel Project (TRW 1998h, Section 8.2.3.2), which assigned a median value of 10, a maximum value of 100, and a minimum value of 1.0 (no dispersion). Consideration of Inventory Modules 1 and 2 and/or the reduced thermal load resulted in a larger-area repository than that considered in the Viability Assessment analysis. Simplified logical models were developed to study the impact of the larger-area repository configurations for this EIS. In general, a larger inventory at the same thermal load results in lower concentrations at the base of the unsaturated zone (barring some exceptionally adverse infiltration conditions) because the spacing between disposal blocks results in the additional amount of waste being spread over a larger area. The larger size of the repository also tends to cause a reduction in the lateral dispersive effects of saturated zone transport, implying lower dilution factors for larger repository configurations. If the dilution factors of the Viability Assessment were to be used in this EIS, the dose rates would be predicted (albeit erroneously) to be lower than their true values for cases with expanded repository areas.

The dilution factors appropriate for the larger-area repository configurations were computed for the EIS analyses. The analytical solution for the three-dimensional transport in a one-dimensional flow field (Leij, Scaggs, and van Genuchten 1991, all) was used to relate the lateral dispersion lengths (in the y and z directions) and the dilution factors. Considering a rectangular source oriented normally to the flow direction, the steady-state concentrations at the locations [5, 20, 30, and 80 kilometers (3, 12, 19, and 50 miles)] were computed based on the assumed dispersion lengths described below.

The ratio between the concentration from the one-dimensional and three-dimensional analyses gives the dilution factor, which enables a "translation" of the Saturated Zone Expert Elicitation Panel's dilution factors to "dispersion lengths." The Panel's dilution estimates were for a 25-kilometer (16 miles) distance and the Viability Assessment adjusted this estimate for estimates at 20 kilometers (12 miles). The dispersion lengths so derived for the Viability Assessment are assumed to remain the same for larger repository configurations. Using the same dispersion lengths, as implied in the Viability Assessment, the dilution factors for the larger repository configurations were computed using the analytical solution. The Darcy flux used in the calculations for the saturated zone flow fields was the same 0.6 meters (2 feet) per year used in the Viability Assessment (DOE 1998a, Volume 3, page 3-138). The actual repository geometry was a rectangular source with an area equivalent to that of the repository configuration for the appropriate thermal load. The larger dimension of the rectangular source was normal to the flow direction and assumed equal in the unsaturated and saturated zones. The smaller dimension of the rectangular source, parallel to the flow in the saturated zone, was modified in the saturated zone to fulfill the continuity of flow requirement (that is, to reconcile large differences in the flow velocities in the unsaturated and saturated zones).

The matrix of dilution factors (given in Table I-36), calculated using the 3DADE computer code (Leij, Scaggs, and van Genuchten 1991, all), was dependent on the major influences on the calculated dilution factors, namely:

- The orientation of each repository configuration relative to the direction of groundwater flow
- The total area of each repository configuration
- The average percolation flux of each sector (or block) of the repository based on the Lawrence Berkeley National Laboratory hydrologic model

Extension of the repository area in a direction orthogonal to that of groundwater flow had little effect on the calculated dilution factor. However, for dilution factors calculated for the repository and enlarged in the direction parallel to that of groundwater flow, there were changes on the order of factors of two or three. Thus, the intermediate thermal load scenario had the same dilution factor as the high thermal load Proposed Action scenario for the 20-kilometer (12-mile) distance, because the repository shape was relatively similar with essentially no changes parallel to the flow direction. In contrast, the low thermal

Table I-36. Dilution factors for three thermal load scenarios and four exposure locations.[a]

		Proposed Action			Inventory Modules 1 and 2		
	Thermal load (MTHM per acre)[b]	High (85)	Intermediate (60)	Low (25)	High (85)	Intermediate (60)	Low (25)
Distance	Repository area (acres)	740	1,050	2,520	1,240	1,750	4,200
5 kilometers[c]	Minimum	1.0	1.0	1.0	1.0	1.0	1.0
	Median	5.15	5.15	2.9	5.15	3.8	2.5
	Maximum	50.02	50.02	24.6	50.02	354	19.2
20 kilometers	Minimum	1.0	1.0	1.0	1.0	1.0	1.0
	Median	10.0	10.0	5.1	10.0	7.2	4.1
	Maximum	100.0	100.0	49.2	100.0	70.8	38.4
30 kilometers	Minimum	1.0	1.0	1.0	1.0	1.0	1.0
	Median	12.2	12.2	6.2	12.2	8.8	4.9
	Maximum	122.0	122.0	60.2	122	86.7	47
80 kilometers	Minimum	1.0	1.0	1.0	1.0	1.0	1.0
	Median	19.894	19.84	9.9	19.84	14.2	7.8
	Maximum	200.04	200.04	98.4	200.04	141.6	76.7

a. Source: TRW (1999a, Table 4.1-1, page 4-6).
b. To convert acres to square miles, multiply by 0.0015625.
c. To convert kilometers to miles, multiply by 0.62137.

load Proposed Action scenario has almost double the area of the intermediate thermal load Proposed Action scenario. The repository is approximately twice the distance in the direction parallel to flow, resulting in a dilution factor almost twice that of the intermediate thermal load Proposed Action scenario. Thus, because of the repository geometry, the differences in the dilution factors between the low and intermediate thermal load Proposed Action scenarios resulted in less dilution in the low thermal load Proposed Action scenario.

I.4.5.5 Modifications to the RIP Model to Account for Unsaturated Zone and Saturated Zone Particle Transport

Transport through the unsaturated zone is modeled in RIP using particles that are assigned a "start location" at the level of the repository. The Viability Assessment analysis considered particle releases only in the upper block of the repository. For the EIS analyses, the Lawrence Berkeley National Laboratory model (Bodvarsson, Bandurraga, and Wu 1997, all) element centroids were mapped to the outline of the upper block, and particles were released from these locations.

Because the EIS analysis considered expanded areas for the emplacement of waste, additional particle coverage was needed to represent transport throughout the entire region of interest. This region included the additional repository blocks for the expanded waste inventories considered in Inventory Modules 1 and 2. An orthogonal grid was mapped for each of the emplacement zones within the area covered by the Lawrence Berkeley National Laboratory model, and this grid was use to determine the coordinates of particle start points at the repository horizon. These coordinates were then converted to the centroid of the nearest Lawrence Berkeley National Laboratory model elements. In this way, a file containing Lawrence Berkeley National Laboratory element numbers was created for each waste emplacement zone for the particle-start coordinates. From this functional area of the RIP model, both the EIS and Viability Assessment performance assessment analyses used the FEHM model (Zyvoloski et al. 1995, all) to model particle transport through the unsaturated zone.

At the base of the unsaturated zone, a corresponding change of coordinates was used to collect and distribute the mass transported through the unsaturated zone to the saturated zone convolution stream tubes that carried dissolved radionuclides to the various exposure locations. The unsaturated and saturated zone capture regions for the EIS analysis were scaled-up modifications of the six regions used by the Total System Performance Assessment – Viability Assessment analysis, as extended to the edge of the Lawrence Berkeley National Laboratory model area. The nodes at the bottom of the unsaturated zone were calculated to ensure complete capture of the mass coming out of the unsaturated zone and to appropriately distribute that mass among the six stream tubes, based on those six repository regions being modified and applied to the expanded areas addressed by the EIS analysis.

Table I-37 lists the ranges of stochastic parameters that were included in the analysis of saturated zone flow and transport.

Table I-37. Stochastic parameters for saturated zone flow and transport.[a]

Parameter	Distribution type	Distribution statistics [bounds]
Effective porosity, alluvium	Truncated normal	Mean = 0.25, SD[b] = 0.075 [0, 1.0]
Effective porosity, upper volcanic aquifer	Log triangular	[1×10^{-5}, 0.02, 0.16]
Effective porosity, middle volcanic aquifer	Log triangular	[1×10^{-5}, 0.02, 0.23]
Effective porosity, middle volcanic confining unit	Log triangular	[1×10^{-5}, 0.02, 0.30]
Effective porosity [plutonium], volcanic units	Log uniform	[1×10^{-5}, 1×10^{-3}]
Distribution coefficient K_d (milliliters per gram) for:		
Neptunium (alluvium)	Uniform	[5, 15]
Neptunium (volcanic units)	Beta (approx. exp.)	Mean = 1.5, SD= 1.3 [0, 15]
Protactinium (alluvium)	Uniform	[0, 550]
Protactinium (volcanic units)	Uniform	[0, 100]
Selenium (alluvium)	Uniform	[0, 150]
Selenium (volcanic units)	Beta (approx. exp.)	Mean = 2.0, SD = 1.7, [0, 15]
Uranium (alluvium)	Uniform	[5, 15]
Uranium (volcanic units)	Uniform	[0, 4.]
Plutonium (all units)	Log uniform	[1×10^{-5}, 10]
Longitudinal dispersivity, all units (meters)	Log-normal	Log(mean) = 2.0, log(SD) = 0.753
Fraction of flow path in alluvium	Discrete CDF[c]	[0, 0.3] (see text)

a. Source: DOE (1998a, Volume 3, Table 3-20, page 3-140).
b. SD = standard deviation.
c. CDF = cumulative distribution function.

I.4.5.6 Biosphere Dose Conversion Factors for Waterborne Radionuclides

A biosphere dose conversion factor for groundwater is a number used to convert the annual average concentration of a radionuclide in the groundwater to an annual radiological dose for humans. The calculation of a biosphere dose conversion factor requires knowledge about the pathway the radionuclide would follow from the well to humans and the lifestyle and eating habits of humans. Figure I-36 illustrates the biosphere modeling components.

The approach used in this long-term performance assessment calculated the health consequences for a reference person living in the Amargosa Valley. The reference person would be an adult who lived year-round on a farm in the Amargosa Valley, grew a garden, raised livestock, and ate locally grown food. Because future human technologies, lifestyles, and activities are inherently unpredictable, the analysis assumed that the future inhabitants of the region would be similar to present-day inhabitants. This assumption has been accepted in similar international efforts at biosphere modeling and is preferable to developing a model for a future society (National Research Council 1995, all).

A lifestyle survey of people living in the area was completed in 1997 (TRW 1998i, Section 9.4, pages 9-25 to 9-35). Among other functions, the survey was intended to give an accurate representation of dietary patterns and lifestyle characteristics of residents within 80 kilometers (50 miles) of the Yucca

Mountain site. Of special interest was the proportion of locally grown foodstuff consumed by local residents and details about regularly consumed food types.

The Amargosa Valley region is primarily rural agrarian in nature and the local vegetation is primarily desert scrub and grasses. Agriculture consists mainly of growing livestock feed (for example, alfalfa); however, gardening and animal husbandry are common. Water for household uses, agriculture, horticulture, and animal husbandry is primarily from local wells.

Another component of the dose to people would be the inadvertent ingestion of contaminated soil, usually from vegetables. The inhalation pathways would include breathing small soil particles that became airborne during outdoor activities, especially farming, mining, and construction activities that would disturb the soil or bedrock. Proximity to a radiation source external to the body would result in an external pathway. This pathway is called "groundshine" when the contaminants are on the ground, "submersion" when they are in the atmosphere, and "immersion" when they are in water.

The analysis calculated biosphere dose conversion factors for the exposure pathways described above. Although many of the input parameters were derived from site-specific data obtained from the Yucca Mountain regional survey and weather data tabulations, some were from other published sources. The input parameters used in the biosphere modeling are described in the Viability Assessment (DOE 1998a, Volume 3, Section 3.8). The estimated consumption rates for vegetables, fruits, grains, beef, poultry, milk, eggs, and water were from the results of the survey (TRW 1998i, Tables 9-14 through 9-20, pages T9-20 to T9-26). Generic food-transfer factors were from IAEA (1994, pages 5 to 58). The amount of plant uptake of radionuclides used in the calculations was taken from LaPlante and Poor (1997, pages 2-12 to 2-14).

The analysis calculated the dose from each radionuclide that would reach the reference person by multiplying the amount of radionuclide ingested, inhaled, or deposited near that person by the dose conversion factor for that radionuclide. Dose conversion factors have important uncertainties associated with them. However (as is customary for radiological compliance evaluations and EISs), this analysis used only fixed values derived by methods from the *International Commission on Radiological Protection Publication 30* (ICRP 1979, all). These methods are similar to those specified by the Environmental Protection Agency (Eckerman, Wolbarst, and Richardson 1988, all).

The long-term performance assessment calculations used the statistical distributions of biosphere dose conversion factors. When the postulated climate change occurred during the model run, the biosphere dose conversion factors changed to reflect the precipitation patterns associated with the new climate. The major impact of a wetter climate would be to reduce the amount of well water required for irrigation. The analysis did not consider other climate-related effects such as the appearance of springs, seeps, or other surface water, because they would be unlikely to cause a large change in the consequences for a maximally exposed individual. The result was the annual dose rate that the reference person would receive from that radionuclide at a given time. The reference person (referred to in this EIS as a maximally exposed individual) was developed from a series of lifestyle assumptions based on the surveys of lifestyles in the region. Details on the reference person development are in the Viability Assessment (DOE 1998a, Volume 3, pages 3-150 to 3-155).

In the analyses for this EIS, the same biosphere dose conversion factors were used for the four locations considered [5, 20, 30, and 80 kilometers (3, 12, 19, and 50 miles)]. The biosphere dose conversion factors are appropriate for the 30-kilometer location due to its similarity to the 20-kilometer location. However, using the same factors for the other locations resulted in a systematic dose overestimation at 5 and 80 kilometers. This overestimate resulted because not all of the exposure pathways considered in the calculation of biosphere dose conversion factors for the 20-kilometer location were appropriate for the 5-and 80-kilometer locations. The 5-kilometer location would be a drinking-water-only pathway (ingestion dose only) because this location is not suitable to irrigation or farming. The 80-kilometer location is a lake playa, where evaporating contaminated water would result in deposits of contaminated dust. Resuspension of the contaminated dust present the only exposure pathway for this location (that is, drinking water and irrigation water pathways would not be relevant). However, development and use of location-specific biosphere dose conversion factors for 5 and 80 kilometers would only serve to reduce the calculated impacts reported in this EIS. Therefore, using the biosphere dose conversion factors developed for the Viability Assessment (DOE 1998a, Volume 3, pages 3-158 to 3-161) for the 20-kilometer location at all other locations evaluated in this EIS is considered conservative.

I.5 Waterborne Radioactive Material Impacts

This section presents the total radiological dose to maximally exposed individuals, as calculated by the RIP model, at the following four groundwater withdrawal or discharge locations downgradient from the Yucca Mountain site where contaminated water could reach the accessible environment:

- A potential well 5 kilometers (3 miles) from the repository
- A potential well 20 kilometers (12 miles) from the repository
- A potential well 30 kilometers (19 miles) from the repository
- Franklin Lake Playa, the closest potential groundwater discharge point downstream from the repository [80 kilometers (50 miles)]

The total radiological dose was calculated from repository closure to 10,000 years following closure and at a time when the peak radiological dose would be observable. RIP model simulations carried out to 1 million years after repository closure also will include the peak radiological dose. These results are provided in Section I.5.1.

Apparent anomalous behavior of total radiological dose results predicted by the RIP model for the low and intermediate thermal load scenario under the Proposed Action inventory is explained in Section I.5.2.

The sensitivity of the estimates of waterborne radioactive material impacts to the fuel cladding model is examined in Section I.5.3.

I.5.1 TOTAL RELEASES DURING 10,000 YEARS AND 1 MILLION YEARS

The RIP model calculated radionuclide releases and radiological doses from individual nuclides and the total radiological dose due to all nine modeled radionuclides released from the repository from failed waste packages. The model calculated total radiological dose in either of two ways: as a single run using expected values of variable parameters, or in multiple realizations (runs) using randomly selected values for distributed parameters. The model can calculate the total radiological dose as the expected value of individual nuclides or the sum of all nuclides, for which sum the model chooses the mean value of all distributed parameters. In addition, the model can use the *Monte Carlo* code to stochastically, or randomly, perform any number of realizations or runs to select values of the distributed parameters. The stochastic nature of the predictions is shown by the complementary cumulative distribution function of the total radiological dose rate (that is, the sum of doses over all radionuclides) for 10,000 or 1 million years. The total radiological dose represents the radiological dose to a maximally exposed individual at the accessible environment using potentially affected groundwater for drinking water. The complementary cumulative distribution functions discussed in this section represent the result of 100 realizations of the RIP model.

The number of realizations used for a Monte Carlo simulation is an important issue with respect to the reliability of analysis results and proper allocation of resources. The number of runs required to reliably predict peak dose rates was examined (DOE 1998a, Volume 3, page 4-71). To verify that 100 realizations would be sufficient, 10,000-year and 100,000-year simulations for the high thermal load scenario with Proposed Action inventory were carried out with 1,000 and 300 realizations, respectively. The resulting distributions of peak individual radiological dose rates were compared with the 100-realization base case results for both periods. The complementary cumulative distribution functions for each time period were found to nearly match. The 100-realization complementary cumulative distribution functions did not go below a probability of 0.01 because each predicted dose rate has a probability of occurrence of one one-hundredth, or 0.01. Similarly, the 1,000- and 300-realization distributions display minimum probabilities of 0.001 and 0.003, respectively. Peak dose rates did continue to increase as probability decreased. Increased dose rates at these low probabilities were caused by combinations of extremely uncertain parameter values sampled from the tails of the parameter probability distributions. However, 100 realizations appear to be sufficient for a good compromise between cost and precision.

Figures I-37 through I-39 show the 10,000-year and 1-million-year complementary cumulative distribution functions of total peak radiological dose for the Proposed Action inventory (see Section I.3.1.2) at 5, 20, 30, and 80 kilometers (3, 12, 19, and 50 miles). In sequence, these figures show the total radiological dose at human exposure locations for the high, intermediate, and low thermal load scenarios and show that the maximum peak radiological dose (total for all nuclides) would occur well after 10,000 years. Further, the 10,000-year complementary cumulative distribution functions show that the distance (of the four distances analyzed) at which the highest total radiological dose would occur is 5 kilometers from the repository. As groundwater moves downgradient from the Yucca Mountain site, it flows from tuffaceous rocks to an alluvial aquifer. The pattern of the complementary cumulative distribution reflects the fact that there would be greater natural retardation in the alluvium than in the tuff portions of the hydrostratigraphic units.

Figures I-40 through I-42 show the 10,000-year and 1-million-year complementary cumulative distribution functions of total peak radiological doses for the Inventory Module 1 inventory at 5, 20, 30, and 80 kilometers (3, 12, 19, and 50 miles). In sequence, these figures show the total radiological doses at human exposure locations for the high, intermediate, and low thermal load scenarios. As for the Proposed Action inventory, these figures show that the maximum peak radiological dose (total, all nuclides) would occur well after 10,000 years. Again, the 10,000-year complementary cumulative distribution functions show that the distance (of the four distances analyzed) at which the highest total radiological dose would occur is 5 kilometers from the repository.

For the Viability Assessment and this EIS, the mean peak dose is the average peak dose of the 100 realizations of radiological dose to a maximally exposed individual (that is, the peak for each realization is determined and all peaks are averaged). The 95th-percentile peak dose is the average of the 95th- and 96th-highest ranked peak doses of the 100 realizations of radiological dose to a maximally exposed individual (that is, the peak for each realization is determined, those peaks are ordered from lowest to highest, and the average of the 95th- and 96th-highest is computed).

I.5.2 APPARENT ANOMALOUS BEHAVIOR BETWEEN LOW AND INTERMEDIATE THERMAL LOAD RESULTS FOR PROPOSED ACTION INVENTORY

Comparison of the expected-value simulations for the different thermal load scenarios at the same distance from the repository reveals apparent anomalous behavior. The differences between the scenarios involving low and intermediate thermal loads under the Proposed Action inventory, which show that the low thermal load curve crosses over the intermediate thermal load curve, require further explanation.

The analysis of three thermal load scenarios revealed some differences in performance as measured by the calculation of total radiological dose to maximally exposed individuals at various distances from the repository. In particular, there is an apparent inconsistent relationship between the total dose-rate history curves for the low and intermediate thermal load scenarios at 20 kilometers (12 miles) from the repository. The apparent differences can be explained by the following factors:

- The effect of repository-area shape on the calculation of the dilution factor using the 3DADE analytical solution (Leij, Scaggs, and van Genuchten 1991, all)
- Waste package degradation differences resulting in the solubility-limited transport, among the different repository blocks being considered for disposal, of neptunium-237 from waste-form degradation
- The correlative differences in the percolation flux

I.5.2.1 Effect of the Dilution Factor

The saturated zone dilution factors were presented and discussed in Section I.4.5.4. As noted in that section, the major influences on the calculated dilution factors were the geometry of the total repository, the orientation of the repository relative to the direction of groundwater flow, and the average estimated infiltration for each repository block. The important finding was that for each repository configuration, extension of the repository area in a direction orthogonal to that of groundwater flow had little effect on the calculated dilution factor. However, when calculated for an enlargement parallel to groundwater flow, there were changes in the range of two to three times the dilution factors.

Thus, the intermediate thermal load Proposed Action scenario for the 20-kilometer (12-mile) distance had the same dilution factor as the high thermal load Proposed Action scenario, because the repository shape was relatively similar with essentially no change orthogonally to the flow direction. In contrast, the low thermal load Proposed Action scenario for the 20-kilometer distance has almost double the area of the intermediate thermal load Proposed Action scenario. Moreover, the repository is approximately twice as long in the direction parallel to groundwater flow, resulting in a dilution factor almost two times less than that of the intermediate thermal load Proposed Action scenario. Thus, because of the repository geometry, the dilution factors between the low and intermediate thermal load Proposed Action scenarios would result in less dilution under the low thermal load scenario.

I.5.2.2 Effect of Waste Package Degradation

Figure I-43 shows the total-radiological-dose-history curve for the Proposed Action inventory for the intermediate and low thermal load scenarios. The peak radiological dose from the low thermal load scenario is slightly delayed compared to the intermediate thermal load scenario, due to the delay in package failure initiation for the low thermal load scenario. An examination of the waste package failure distribution between these two scenarios (Figure I-44) shows that after the initial juvenile package failure (one package fails early for every case) stipulated by the Viability Assessment analysis, the first failure of the intermediate thermal load scenario is about 9,000 years after repository closure, whereas the first failure of the low thermal load scenario is about 27,000 years after repository closure. Thus, the amount of neptunium-237 available for removal from the repository is less for the low thermal load scenario than for the intermediate thermal load scenario.

The disparity in amount of neptunium-237 available for removal persists until the time of the super-pluvial climate. Figure I-43 shows that until the super-pluvial climate cycle (about 300,000 years after repository closure) the low thermal load total radiological dose history curve lies below and later than the intermediate thermal load total radiological dose history curve. Essentially, the peak radiological doses occur at different times by that same amount of material removed. At this time, the number of waste package failures has increased to allow differences in removal rates from the repository due to the solubility limitations of neptunium-237. A larger proportion of the neptunium-237 is removed under the intermediate thermal load conditions because of the relatively higher amount of percolation flux and larger number of waste packages for the upper block for this scenario. However, more of the neptunium-237 remains in the repository under the low thermal load case because it can not all be removed from the larger repository area due to the reduced amount of water. The total-radiological-dose-to-receptor curve then crosses over the intermediate thermal load curve at about 300,000 years after closure. Thereafter, the two curves slowly approach one another during the remainder of the simulation but never recross during the simulated period.

I.5.2.3 Effect of Percolation Flux Distribution

The percolation flux differs across Yucca Mountain, especially in relation to the proposed areas. Figure I-45 shows the average percolation flux for the different repository areas. Note that Block 5 has the lowest percolation flux and Block 8 has the largest percolation flux. The intermediate thermal load Proposed Action scenario includes only the upper block (Block 1) and the capture areas are similar to the high thermal load Proposed Action scenario. The average infiltration flux for the upper block is larger than that for Block 8.

A sensitivity analysis using only the long-term average climate shows that the release rate of neptunium-237 at the top of the water table has two peaks. One is influenced by percolation flux in capture regions 1, 2, and 4, and the other is influenced by percolation flux in capture regions 3, 5, and 6. The reason for the two-peak aspect of the total release-rate curve is that neptunium-237 is solubility limited, and the lower percolation flux in the lower block and Block 8 does not completely remove all of the available neptunium-237 from these blocks at the same rate as in areas with greater percolation flux. The comparable curve for the intermediate thermal load Proposed Action scenario shows that all neptunium-237 is released at approximately the same time. Figures I-46 through I-49 show a comparison of the neptunium-237 radiological dose-rate histories for the low and intermediate thermal load scenarios for only the average long-term climate at the engineered barrier system and at the exposure location [20 kilometers (12 miles)]. These figures show that the difference in percolation flux is apparent at the engineered barrier system and accentuated in the saturated zone because of the retarded release of neptunium-237 under lower percolation flux. Because neptunium-237 is the dominant radionuclide contributing to the total radiological dose at times greater than 100,000 years, the curves indicating the low and intermediate thermal load total radiological-dose rate history cross. After crossing, the curves do not maintain their separation but tend to approach one another without recrossing for the remainder of the 1-million-year simulation period. It appears that they would likely cross again between 1 million and 1.5 million years at the observed rate of closure if the simulation were extended.

I.5.2.4 Conclusion

The analysis of the three thermal loads proposed for the planned repository configuration revealed anomalous differences in performance as measured by the calculation of total radiological dose to maximally exposed individuals at various distances from the repository. The apparent differences can be explained by three factors:

- The effect of repository area shape on the calculation of the saturated zone dilution factor using the 3DADE numerical code, based on an analytical solution to flow and transport from the repository
- Differences in waste package failure under the different thermal loads
- Differences in the percolation flux and the correlative neptunium-237 solubility-limited transport among the different repository blocks being considered for disposal

I.5.3 SENSITIVITY TO FUEL CLADDING MODEL

Section 5.4.4 of this EIS describes a sensitivity analysis DOE conducted to assess the importance of fuel pin cladding protection on radiological dose. This section contains additional details for the sensitivity analysis.

The average radionuclide inventory listed in Table I-1 for each commercial spent nuclear fuel waste package was used in the sensitivity analysis. Under the Proposed Action, approximately 1.2 percent of the spent nuclear fuel would have stainless-steel cladding rather than zirconium-alloy cladding. The stainless steel would degrade much faster than zirconium alloy, so the sensitivity analysis neglected stainless-steel cladding as a protective barrier. In addition, approximately 0.1 percent of the fuel pins are proposed to fail in the reactor environment. Thus, under the Proposed Action, 1.3 percent of the radionuclides in every spent nuclear fuel waste package would be available for degradation and transport as soon as the waste package failed.

For the purposes of comparison, the analysis performed additional stochastic runs for 10,000 and 1 million years after repository closure assuming the zirconium-alloy cladding would provide no resistance to water or radionuclide movement after the waste package failed. Table I-38 compares the peak radiological dose rate from groundwater transport of radionuclides for the base case and this case, which assures zirconium-alloy cladding would not be present. The analysis used data representing the high thermal load scenario to calculate individual exposures for a 20-kilometer (12-mile) distance only for purposes of comparison.

Table I-38. Comparison of consequences for a maximally exposed individual from groundwater releases of radionuclides using different fuel rod cladding models under the high thermal load scenario.

	Mean consequence[a]		95th-percentile consequence[b]	
Maximally exposed individual	Dose rate (millirem/year)	Probability of an LCF[c]	Dose rate (millirem/year)	Probability of an LCF
Peak at 20 kilometers[d] within 10,000 years after repository closure with cladding credit	0.22	7.6×10^{-6}	0.58	2.0×10^{-5}
Peak at 20 kilometers within 10,000 years after repository closure without cladding credit	5.4	1.9×10^{-4}	15	5.3×10^{-4}
Peak at 20 kilometers within 1 million years after repository closure with cladding credit	260	9.0×10^{-3}	1,400	5.0×10^{-2}
Peak at 20 kilometers within 1 million years after repository closure without cladding credit	3,000	1.1×10^{-1}	10,800	3.8×10^{-1}

a. Based on sets of 100 simulations of total system performance, each using random samples of uncertain parameters.
b. Represents a value for which 95 out of the 100 simulations yielded a smaller value.
c. LCF = latent cancer fatality; incremental lifetime (70 years) risk of contracting a fatal cancer for individuals, assuming a risk of 0.0005 latent cancer per rem for members of the public (NCRP 1993a, page 31).
d. To convert kilometers to miles, multiply by 0.62137.

Figure I-50 shows complementary cumulative distribution functions of the peak radiological dose rates for the four suites of model runs. Approximately 25 percent of the 10,000-year runs did not show any releases to the locations at a distance of 20 kilometers (12 miles). The zero releases are the reason the 10,000-year curves in Figure I-50 start at an exceedance probability of 0.73 and decrease with increasing radiological dose rate. All of the 1-million-year runs show releases at 20 kilometers.

The analysis assumed that the zirconium-alloy cladding would provide no barrier to water movement and radionuclide mobilization after the failure of the waste package. However, DOE expects that the zirconium alloy would provide some impediment to radionuclide mobilization when the waste package is breached. Therefore, the results for no cladding listed in Table I-38 should be viewed as an upper boundary.

I.6 Waterborne Chemically Toxic Material Impacts

Further transport analysis is warranted because the screening analysis (Section I.3.2.3.3) indicated that the repository could release chromium into groundwater in substantial quantities and thus could represent a human-health impact. Surrogate calculations were performed using the RIP model and inputs based on the radiological materials transport simulations. This approach selected a long-lived unretarded isotope (iodine-129) to serve as a surrogate for chromium. Iodine is highly soluble and exhibits little or no sorption so when corrected for radioactive decay, its movement represents scalar transport. This method avoided the extensive inputs necessary to define a new species for the RIP model and revision of the associated external function modules that the analysis had carefully constructed for the nine modeled radionuclides.

I.6.1 CHROMIUM

The screening analysis for chemically toxic materials (Section I.3.2.3) identified chromium from the waste packaging as a potential impact of concern. This section describes a chromium inventory for use in the RIP model and evaluates chromium impacts.

I.6.1.1 RIP Model Adaptations for Chromium Modeling

The following assumptions were applied to the chromium surrogate calculation approach:

1. Iodine-129 will serve adequately as a surrogate for chromium because it has a long radioactive half-life, lacks decay ingrowth by predecessors in a decay chain in the RIP model calculations, and is not retarded in groundwater (chromate is also unretarded). A small error introduced by the slight radioactive decay of iodine-129 during the model simulations can be corrected by an analytical expression as a postprocessing step.

2. Alloy-22 degradation and release is modeled using general corrosion depth of the corrosion-resistant material taken from WAPDEG modeling results (Mon 1999, all) for both dripping and nondripping conditions. The WAPDEG modeled the general corrosion depth (in millimeters per year) of corrosion-resistant material for 400 waste packages were averaged to produce a general degradation rate for dripping and nondripping conditions and converted to a fraction of corrosion-resistant material per year rate for use in the RIP model. The fractional degradation rate curves are show in Figure I-51.

3. Chromium associated with stainless-steel components used in many commercial spent nuclear fuel waste packages would be released proportionately with Alloy-22 chromium. This conservative assumption effectively assumes no credit for the delay of the onset of interior stainless-steel degradation or for the degradation rate of the interior stainless steel itself.

The treatment of Alloy-22 corrosion-resistant material degradation and chromium mobilization required the redefinition of the RIP container model. This calculation used the "Primary Container" in the RIP model to represent only the corrosion-allowance material (outer layer) of the waste package. The "Secondary Container" in the RIP model (used to represent cladding in the radiological material transport simulations) was not used. The waste matrix was used to represent the corrosion-resistant inner layer made of Alloy-22. These steps, with the proper material inventory and degradation coefficients, enabled the use of the current RIP model structure for this calculation.

The following additional changes were made to the radiological RIP model input files to conduct the surrogate chromium mobilization and migration calculation:

1. Iodine-129 solubility was specified as 1,976 grams per cubic meter (0.12 pounds per cubic foot), based on the near-field geochemistry screening study results for chromium (iodine-129 serving as a surrogate for chromium). Section I.3.2.3.1 contains details on determining this solubility limit.

2. For each source term, the inventory of all radionuclides (except iodine-129) received a value of zero.

3. The inventory of iodine-129 in each source term were specified in units of grams (rather than the original units of curies) per waste package using the values in Tables I-18 through I-22 in Section I.3. All inventory was assigned to the RIP model Waste Matrix Fraction (and none to the Primary or Secondary Container Fractions) in each source term.

4. The analysis assumed that mobilized chromium from the corrosion-resistant material would advect directly from the exposed corrosion-resistant material surface onto the invert (drift floor).

5. All secondary container definitions were all changed to a "degenerate" distribution at time zero, to eliminate the effects of any cladding protection from the calculation. A degenerate distribution simply results in all secondary containers failing at the specified time. The Alloy-22 corrosion-resistant material layer would be outside the cladding and, hence, not a barrier from this perspective.

6. The primary container definitions were changed to a "Degenerate" distribution at time zero, to eliminate the effects of corrosion-allowance material protection. This step is necessary because the protective benefits of the corrosion-allowance material are implicit in the WAPDEG results used to directly incorporate corrosion-resistant material degradation into the RIP model.

7. The waste-form-degradation rate for each source term was replaced with new variables representing weight-averaged Alloy-22 degradation. The definition of these degradation rates is detailed below.

8. RIP model output was requested in grams (mass) rather than curies (radioactivity).

To arrive at a weight-averaged fractional corrosion rate to apply to all waste packages of a given category (spent nuclear fuel, high-level radioactive waste, or DOE spent nuclear fuel) in a given repository region, the following steps were taken. The Alloy-22 generalized corrosion depth for dripping and nondripping conditions was converted to a fractional degradation rate, as described above. The Alloy-22 fractional corrosion rate was computed from a weighted average (with respect to the fraction of packages subject to dripping and nondripping conditions in the current climate) of dripping and nondripping generalized corrosion rates. This weight-averaged fractional degradation rate was then used to model the release of chromium from the waste package to the near-field environment.

For the Proposed Action, 30 percent of the chromium inventory would originate from interior stainless-steel components used in some commercial spent nuclear fuel waste packages (see Table I-16). Because the waste package would have to fail before degradation and transport of interior components could begin, simply adding the two chromium inventories together would yield artificially high results.

A two-stage scoping analysis, following the steps outlined above for using the RIP model to calculate chromate migration, was performed for the Proposed Action inventory under the high thermal load scenario to predict chromate concentrations at the 5-kilometer (3-mile) distance. In the first stage, the model was run with only the chromium inventory from the Alloy-22 corrosion-resistant material [904,00(grams (about 2,000 pounds) of chromium per commercial spent nuclear fuel waste package] following the steps outlined above for chromium modeling. In the second stage, the model was run again with only the interior stainless-steel inventory [514,000 grams (about 1,100 pounds) of chromium per commercial spent nuclear fuel waste package] but used the complete WAPDEG waste package model (as used in the Viability Assessment) to represent complete waste package containment. Only the commercial spent nuclear fuel packages would differ; no interior stainless-steel internal components would be used in high-level radioactive waste or DOE spent nuclear fuel containers. Each RIP model run was held to the same random number seed (used to "seed" the random number generator that is used to select random values of stochastic parameters) so the realizations would be replicated. The results of each simulation were summed, with respect to realization and time step, to calculate the total chromium concentration at 5 kilometers (3 miles). The results are listed in Table I-39.

Table I-39. Chromium groundwater concentrations (milligrams per liter)[a] at 5 kilometers (3 miles) under Proposed Action inventory using the high thermal load scenario and a two-stage RIP model.

Model	Peak chromium concentration	
	Mean	95th-percentile
RIP Stage 1: Corrosion-resistant material (Alloy-22) chromium inventory	0.0085	0.037
RIP Stage 2: Interior-to-waste package (SS/B[b] alloy) chromium inventory	0.000000086	0.00000048
Totals (Stage 1 + Stage 2, by realization; time step)	**0.0085**	**0.037**

a. To convert milligrams per liter to pounds per cubic foot, multiply by 0.00000624.
b. SS/B = stainless-steel boron.

The chromium concentrations obtained in this scoping analysis demonstrated that the inventory of chromium associated with interior stainless-steel components, although it would represent 30 percent of the total chromium inventory, would be small with respect to the peak chromium concentration in groundwater at the closest downgradient location considered. Including the interior stainless-steel chromium inventory increased the estimate of the mean peak chromium concentration by 0.00088 percent over modeling the corrosion-resistant material chromium alone. The 95th-percentile peak chromium concentration was increased by 0.000072 percent over modeling the corrosion-resistant material inventory of chromium alone. Therefore, an additional step to model the interior stainless-steel corrosion and transport was unnecessary to predict peak chromate concentrations.

Two factors would contribute to the inconsequential impact of the chromium inventory from the waste package interior. First, the Alloy-22 in the waste package would have to be breached before interior stainless steel was exposed to water and began to degrade. Thus, much of the chromium in the Alloy-22 would already have migrated before the interior stainless-steel chromium began to degrade and migrate. Second, the Alloy-22 degradation would depend strongly on the RIP model parameters controlling the fraction of packages exposed to dripping conditions. Packages that experienced dripping conditions would degrade much faster; only those that experienced dripping conditions would fail within 10,000 years and permit exposure of interior stainless steel. The vast majority of waste packages would not fail, so the interior chromium inventory would never be exposed for degradation and transport.

Based on this demonstration of the relative unimportance of the interior stainless-steel chromate inventory in calculating peak chromium concentrations within 10,000 years, only the corrosion-resistant material (Alloy-22) in the chromium inventory was simulated for analysis of chromium impacts as a waterborne chemically toxic material.

I.6.1.2 Results for the Proposed Action

The chromium-migration calculation was conducted for the Proposed Action inventory under the high, intermediate, and low thermal load scenarios using the same stochastic approach as that used for the waterborne radioactive material assessment. The 100 independent realizations, using randomly selected input parameter values chosen from assigned probability distributions of values, were simulated with the RIP model. Simulations were performed to estimate chromium concentrations at 5, 20, 30, and 80 kilometers (3, 12, 19, and 50 miles) for 10,000 years following closure. The resulting concentrations were decay-corrected to remove the slight radioactive decay calculated by the RIP model for the surrogate constituent, iodine-129.

The mean peak concentrations and 95th-percentile peak concentrations computed with the RIP model, using the surrogate chromium-migration calculation described above, are listed in Table I-40 for all thermal load scenarios under the Proposed Action. Figures I-52 through I-54 show the complementary cumulative distribution function for the 100 realizations of chromium concentration under the Proposed Action at each of the four locations for the low, intermediate, and high thermal load scenarios, respectively.

Table I-40. Peak chromium groundwater concentration (milligrams per liter)[a] under the Proposed Action inventory.[b]

Thermal load	Maximally exposed individual	Mean	95th-percentile
High	At 5 kilometers[c]	0.0085	0.037
	At 20 kilometers	0.0028	0.012
	At 30 kilometers	0.0018	0.0063
	At 80 kilometers	0.00022	0.00061
Intermediate	At 5 kilometers	0.0029	0.0096
	At 20 kilometers	0.0023	0.010
	At 30 kilometers	0.00080	0.0038
	At 80 kilometers	0.000031	0.00015
Low	At 5 kilometers	0.0046	0.016
	At 20 kilometers	0.0018	0.0083
	At 30 kilometers	0.00067	0.0033
	At 80 kilometers	0.000053	0.00034

a. To convert milligrams per liter to pounds per cubic foot, multiply by 0.0000624.
b. Based on 100 repeated simulations of total system performance, each using randomly sampled values of uncertain parameters.
c. To convert kilometers to miles, multiply by 0.62137.

A simple sensitivity run, reducing the solubility limit of the iodine-129 surrogate by one order of magnitude (from 1,976 to 197.6 milligrams per liter), demonstrated that the imposed value of the solubility limit did not affect the resulting concentration at the accessible environment. This demonstration suggests that the chromium degradation rate is a major controlling factor over the release of chromium.

There are two measures for comparing human health effects for chromium. When the Environmental Protection Agency established its Maximum Contaminant Level Goals, it considered safe levels of contaminants in drinking water and the ability to achieve these levels with the best available technology. The Maximum Contaminant Level Goal for chromium is 0.1 milligram per liter (0.0000062 pound per cubic foot) (40 CFR 141.51). The other measure for comparison is the reference dose factor for chromium, which is 0.005 milligram per kilogram (0.0004 ounce per pound) of body mass per day (EPA 1999, all). The reference dose factor represents a level of intake that has no adverse effect on humans. It can be converted to a threshold concentration level for drinking water. The conversion yields essentially the same concentration for the reference dose factor as the Maximum Contaminant Level Goal.

No attempt can be made at present to estimate the groundwater concentrations of hexavalent chromate in Table I-40, in terms of human health effects (for example, latent cancer fatalities). The carcinogenicity of hexavalent chromium by the oral route of exposure cannot be determined because of a lack of sufficient epidemiological or toxicological data (EPA 1999, all; EPA 1998, page 48).

I.6.1.3 Results for Inventory Modules 1 and 2

Chromium impacts were calculated for Inventory Modules 1 and 2 using the same approach as for the Proposed Action. Peak mean and 95th-percentile chromium concentrations for Inventory Modules 1 and 2 are listed in Tables I-41 and Table I-42, respectively. Figures I-55 through I-57 show the complementary cumulative distribution function for the 100 realizations of chromium concentration for Inventory Module 1 at each of the four locations for the low, intermediate, and high thermal load scenarios, respectively.

Table I-41. Peak chromium groundwater concentration (milligrams per liter)[a] for 10,000 years after closure under Inventory Module 1.[b]

Thermal load	Maximally exposed individual	Mean	95th-percentile
High	At 5 kilometers[c]	0.032	0.14
	At 20 kilometers	0.018	0.10
	At 30 kilometers	0.0057	0.027
	At 80 kilometers	0.00029	0.00070
Intermediate	At 5 kilometers	0.023	0.083
	At 20 kilometers	0.0089	0.042
	At 30 kilometers	0.0032	0.017
	At 80 kilometers	0.00019	0.00057
Low	At 5 kilometers	0.0093	0.0353
	At 20 kilometers	0.0050	0.022
	At 30 kilometers	0.0020	0.0084
	At 80 kilometers	0.000074	0.00026

a. To convert milligrams per liter to pounds per cubic foot, multiply by 0.0000624.
b. Based on 100 repeated simulations of total system performance, each using randomly sampled values of uncertain parameters.
c. To convert kilometers to miles, multiply by 0.62137.

Table I-42. Peak chromium groundwater concentration (milligrams per liter)[a] due only to Greater-Than-Class-C and Special-Performance-Assessment-Required wastes for 10,000 years after closure under Inventory Module 2.[b]

Thermal load	Maximally exposed individual	Expected Value
High	At 5 kilometers[c]	0.0014
	At 20 kilometers	0.00058
	At 30 kilometers	0.00021
	At 80 kilometers	0.000000012
Intermediate	At 5 kilometers	0.00080
	At 20 kilometers	0.00033
	At 30 kilometers	0.00012
	At 80 kilometers	0.0000000094
Low	At 5 kilometers	0.00060
	At 20 kilometers	0.00025
	At 30 kilometers	0.000086
	At 80 kilometers	0.000000010

a. To convert milligrams per liter to pounds per cubic foot, multiply by 0.0000624.
b. Based on an expected value simulation using the mean of all stochastic parameters for the additional inventory of Inventory Module 2 over Inventory Module 1.
c. To convert kilometers to miles, multiply by 0.62137.

There are two measures for comparing human health effects for chromium. When the Environmental Protection Agency established its Maximum Contaminant Level Goals, it considered safe levels of contaminants in drinking water and the ability to achieve these levels with the best available technology. The Maximum Contaminant Level Goal for chromium is 0.1 milligram per liter (0.0000062 pound per cubic foot) (40 CFR 141.51). The other measure for comparison is the reference dose factor for chromium, which is 0.005 milligram per kilogram (0.0004 ounce per pound) of body mass per day (EPA 1999, all). The reference dose factor represents a level of intake that has no adverse effect on humans. It can be converted to a threshold concentration level for drinking water. The conversion yields essentially the same concentration for the reference dose factor as the Maximum Contaminant Level Goal.

No attempt can be made at present to express the estimated groundwater concentrations of hexavalent chromate in Table I-42 in terms of human health effects (for example, latent cancer fatalities). The carcinogenicity of hexavalent chromium by the oral route of exposure cannot be determined because of a lack of sufficient epidemiological or toxicological data (EPA 1999, all; EPA 1998, page 48).

I.6.2 MOLYBDENUM

Alloy-22 used as a waste package inner barrier also contains 13.5 percent molybdenum (ASTM 1994, page 2). During the corrosion of Alloy-22, molybdenum behaves almost the same as the chromium. Due to the corrosion conditions, molybdenum also dissolves in a highly soluble hexavalent form. Therefore, the source term for molybdenum will be exactly 13.5/22 times (61.4 percent) the source term for chromium. All the mechanisms and parameters are the same as those used for chromium so modeling is unnecessary. It is reasonable to assume that molybdenum would be present in the water at concentrations 61.4 percent of those reported above for chromium.

There is currently no established toxicity standard for molybdenum (in particular, the Environmental Protection Agency has not established a Maximum Contaminant Level Goal for molybdenum), although this does not mean that molybdenum is not toxic. The concentrations of molybdenum would be very small, so no effect would be likely to result from the molybdenum released to the groundwater.

I.6.3 URANIUM

While the screening analysis indicated that elemental uranium would not pose a health risk as a waterborne chemically toxic material (see Section I.3.2.3.3), it was retained for consideration for other reasons. The total uranium inventory (all uranium isotopes) is listed for the inventory modules in Table I-23.

The reference dose for elemental uranium is 0.003 milligram per kilogram of body mass per day (EPA 1999, all). Assuming that a child would experience the maximum individual exposure from the drinking-water pathway, the analysis used a 1-liter (0.26-gallon) daily intake rate and a 16-kilogram (35-pound) body weight to convert the reference dose to a threshold concentration of 4.8×10^{-2} milligram per liter (2.9×10^{-6} pound per cubic foot).

I.6.3.1 RIP Model Adaptations for Elemental Uranium Modeling

To evaluate the consequences of total uranium migration, the mobilization and transport of the total uranium inventory for the Proposed Action listed in Table I-23 were simulated using the RIP model. The following steps were taken in the RIP model adaptation for the total uranium simulations:

1. The inventory of all radionuclides except uranium was set to zero (as a precaution and to prevent confusion with radiological runs).

2. The inventory of uranium (all isotopes) was changed to 8,119 kilograms (17,900 pounds) for commercial spent nuclear fuel packages, 786 kilograms (1,730 pounds) for DOE spent nuclear fuel packages, and 2,826 kilograms (6,220 pounds) for high-level radioactive waste packages.

3. Output from the RIP model was requested in grams rather than curies.

4. The radiological decay rate of uranium-234 was left to represent all uranium isotopes in the waste packages, although the resulting concentrations obtained from RIP model simulations were decay-corrected to provide undecayed concentrations. Various uranium isotopes have different half-lives, so the analysis ignored decay benefits in reducing impacts.

5. Because the chemical properties (such as sorption rate) are functions of the element and not the isotope, the other transport properties of uranium were left the same as those used for the radiological consequences simulations.

6. Use of the parameter FCSOLU, which is used in the RIP model to partition the solubility coefficient to account for the fact that radionuclide simulations model only one isotope of uranium, was omitted for full uranium elemental simulations.

DOE ran 100 simulations to model the release and transport of uranium. The Proposed Action inventory is approximately 70,000 MTHM (77,000 tons). Although a small percentage of the heavy metal in the spent fuel is not uranium, it was reasonable to assume all of it was because doing so had a very small effect on the result and would make the analysis more conservative. This assumption introduced an approximate 7-percent increase into the result. The runs are based on the high thermal load scenario, and the consequences are computed for 5 kilometers (3 miles) from the repository. In addition, the analysis neglected radioactive decay. Most of the uranium present has a very long half-life compared to the analysis period, so decay would have a very small conservative effect on the result.

I.6.3.2 Results for the Proposed Action

The Proposed Action inventory of elemental uranium would be approximately 65 million kilograms (72,000 tons) (see Table I-23). Total elemental uranium migration calculations were made using the RIP model code for the Proposed Action inventory under the high thermal load scenario for 10,000 years following closure for the 5-kilometer (3-mile) distance. The resulting concentrations of elemental uranium in groundwater at the 5-kilometer (3-mile) discharge location were obtained from the simulation results.

The reference dose for elemental uranium is 3.0×10^{-3} milligram per kilogram (4.8×10^{-8} ounce per pound) of food intake per day (EPA 1999, all). Assuming that a child would experience the maximum individual exposure for the drinking water scenario, the analysis used a 1-liter (0.26-gallon) daily intake rate and a 16-kilogram (35-pound) body weight to convert the reference dose to a threshold concentration. The threshold concentration would be 0.048 milligram per liter (3.0×10^{-6} pound per cubic foot).

The maximum uranium concentration over 10,000 years was extracted for each of the 100 sets of simulation results. The mean peak concentration of uranium would be 6.7×10^{-8} milligram per liter (5.2×10^{-9} pound per cubic foot), and the 95th-percentile peak concentration would be 2.2×10^{-8} milligram per liter (1.7×10^{-9} pound per cubic foot). These concentrations would be six orders of magnitude lower than the threshold concentration for the oral reference dose, so DOE expects no human health effects from the chemical effects of waterborne uranium under the high thermal load scenario.

Figure I-58 shows the complementary cumulative distribution function for elemental uranium concentrations at the 5-kilometer (3-mile) discharge location for 10,000 years following closure under the high thermal load scenario. The groundwater concentration information in this figure shows that uranium, as a chemically toxic material, would be far below the reference dose at any probability level.

Based on trends in waterborne radioactive material results, the concentrations of elemental uranium at locations that were more distant [20, 30, and 80 kilometers (12, 19, and 50 miles)] and for the intermediate and low thermal load scenarios at all distance would be even lower. Because of the extremely low concentrations from these simulations, further simulations were unnecessary to evaluate other thermal loads under the Proposed Action. Elemental uranium would not present a health risk as a chemically toxic material under the Proposed Action for any thermal load scenario.

I.6.4 RESULTS FOR INVENTORY MODULES 1 AND 2

Under Inventory Modules 1 and 2, the total uranium inventory would increase from the Proposed Action total of 70,000 MTHM to 120,000 MTHM (Table I-18). The 70-percent increase in elemental uranium inventory would be likely to increase the groundwater concentration at the discharge location (1) at most, if the percentage of the inventory was increased, or (2) by less, if solubility limits were exceeded along the transport paths in groundwater in any case. Even doubling the groundwater concentrations calculated for the Proposed Action inventory would result in concentration levels that would be several orders of magnitude below the reference dose concentration level. Therefore, elemental uranium would not present a substantial health risk as a chemically toxic material under Inventory Module 1 or 2 for any thermal load scenario.

I.7 Atmospheric Radioactive Material Impacts

After DOE closed the Yucca Mountain Repository, there would be limited potential for releases to the atmosphere because the waste would be isolated far below the ground surface. Still, the rock is porous and does allow gas to flow, so the analysis must consider possible airborne releases. The only radionuclide that would have a relatively large inventory and a potential for gas transport is carbon-14. Iodine-129 can exist in a gas phase, but it is highly soluble and therefore would be more likely to dissolve in groundwater rather than migrate as a gas. Other gas-phase isotopes were eliminated in the screening analysis (Section I.3), usually because of short half-lives and because they are not decay products of long-lived isotopes. After carbon-14 escaped from the waste package, it could flow through the rock in the form of carbon dioxide. Atmospheric pathway models were used to estimate human health impacts to the local population in the 84-kilometer (52-mile) region surrounding the repository.

About 2 percent of the carbon-14 in commercial spent nuclear fuel exists as a gas in the space (or *gap*) between the fuel and the cladding around the fuel (Oversby 1987, page 92). The average carbon-14 inventory in a commercial spent nuclear fuel waste package is approximately 12 curies (see Table I-1), so the analysis used a gas-phase inventory of 0.23 curie of carbon-14 per commercial spent nuclear fuel waste package to calculate impacts from the atmospheric release pathway. The analysis described in Section 5.4 included the entire inventory of the carbon-14 in the repository in the groundwater release models. Thus, the groundwater-based impacts would be overestimated slightly (by 2 percent) by this modeling approach.

Carbon is the second-most abundant element (by mass) in the human body, constituting 23 percent of Reference Man (ICRP 1975, page 377). Ninety-nine percent of the carbon comes from food ingestion (Killough and Rohwer 1978, page 141). Daily carbon intakes are approximately 300 grams (0.7 pound) and losses include 270 grams (0.6 pound) exhaled, 7 grams (0.02 pound) in feces, and 5 grams (0.01 pound) in urine (ICRP 1975, page 377).

Carbon-14 dosimetry can be performed assuming specific-activity equivalence. The primary human-intake pathway of carbon is food ingestion. The carbon-14 in food results from photosynthetic processing of atmospheric carbon dioxide, whether the food is the plant itself or an animal that feeds on the plant. Biotic systems, in general, do not differentiate between carbon isotopes. Therefore, the carbon-14 activity concentration in the atmosphere will be equivalent to the carbon-14 activity concentration in the plant, which in turn will result in an equivalent carbon-14 specific activity in human tissues.

I.7.1 CARBON-14 RELEASES TO THE ATMOSPHERE

The calculation of regional radiological doses requires estimation of the annual release rate of carbon-14. The analysis based the carbon-14 release rate on the predicted timeline of container failures for the high thermal load scenario, using average values for the stochastic parameters that were entered. The expected number of spent nuclear fuel waste package failures in 100-year intervals was used to estimate the carbon-14 release rate after repository closure. The estimated amount of material released from each package as a function of time was reduced to account for radiological decay.

As for the waterborne releases described in Section 5.4, some credit was taken for the intact zirconium-alloy cladding (on approximately 99 percent by volume of the spent nuclear fuel) delaying the release of gas-phase carbon-14. The remaining 1 percent by volume of the spent nuclear fuel either would have stainless-steel cladding (which degrades much more quickly than zirconium alloy) or would already have failed in the reactor. The RIP model uses a waste package failure model that conceptually divides the surface area of the waste packages into many *patches*. A corrosion future for each patch is then calculated. The zirconium-alloy cladding failure model is implemented in the same fashion, with the cladding corrosion rate set to a fraction of the corrosion rate of the Alloy-22 in the inner shell of the waste package. This analysis set the cladding corrosion rate for the zirconium alloy to the same value used in the Viability Assessment (DOE 1998a, Volume 3, page 3-101). A plot of the patch-area fraction of the zirconium-alloy cladding that has failed as a function of time after repository closure is shown in Figure I-59. Although difficult to see on the plot scale, no zirconium-alloy cladding would fail during the first 5,000 years after repository closure.

The amount (in curies) of carbon-14 that would be available for transport from a failed waste package, A_T, is calculated as:

$$A_T = (F_{IF} + F_{FC}) \times 0.23 \text{ curies per package}$$

where:

F_{IF} = fraction immediately failed (fuel with stainless-steel cladding or previously failed fuel pins)

F_{FC} = fraction of failed cladding (if the value shown in Figure I-59 is less than 0.01, then that value is used; if the value shown in Figure I-59 exceeds 0.01, then a value of 0.9875 is used)

The model uses the patch failure rate on the zirconium alloy as the fraction of the failed pins until the patch failure rate reaches 1 percent. After the patch failure rate reaches 1 percent, the release rate is reset to not take further credit for zirconium-alloy cladding reducing the transport rate of gas-phase carbon-14. Rather than conducting a detailed gas-flow model of the mountain, the analysis assumed that the carbon-14 from the failed waste package would be released to the ground surface uniformly over a 100-year interval. Thus, the release rate to the ground surface for a waste package would be A_T divided by 100 (curies per year).

Figure I-60 shows the estimated release rate of carbon-14 from the repository for 50,000 years after repository closure, assuming that the spent nuclear fuel with stainless-steel cladding had failed and released its gas-phase carbon-14 prior to being placed in a waste package. This assumption is represented by F_{IF}=0 in the calculation for A_T. The results in Figure I-60 are based on the Proposed Action inventory. Each symbol in the figure represents the carbon-14 release rate to the ground surface for a period of 100 years. The general downward slope of the symbols is due to radioactive decay (carbon-14 has a half-life of 5,730 years). The symbols marking zero releases (curies per year) indicate that no waste packages failed during some 100-year periods. The jagged nature of the plot indicates a different number of waste packages failing in different 100-year intervals. Only 97 of 7,760 spent nuclear fuel waste packages would have failed during the first 10,000 years after repository closure. By 40,000 years after repository closure, 676 of the 7,760 spent nuclear fuel waste packages would have failed. Using this expected-value representation of waste package lifetime, no more than three waste packages would have failed in any single 100-year interval before 30,000 years after repository closure. Between 30,000 and 50,000 years after repository closure, as many as five waste packages would fail in a single 100-year interval. The maximum release rate would occur about 19,000 years after repository closure. The estimated maximum release rate would be about 0.098 microcurie per year.

I.7.2 ATMOSPHERE CONSEQUENCES TO THE LOCAL POPULATION

DOE used the GENII-S code (Leigh et al. 1993, all) to model the atmospheric transport and human uptake of released carbon-14 for the 84-kilometer (52-mile) population radiological dose calculation. This calculation used 84 kilometers rather than the typical 80 kilometers (50 miles) used in an EIS to include the population of Pahrump, Nevada, in the impact estimate. Radiological doses to the regional population near Yucca Mountain from carbon-14 releases were estimated using the population distribution compiled from DOE (1998a, Volume 3, Figure 3-76), which indicates approximately 28,000 people would live in the region surrounding Yucca Mountain in the year 2000. The population by distance and sector used in the calculations are listed in Table I-43. The computation also used current (1993 to 1996) annual average meteorology. The joint frequency data are listed in Table I-44.

Table I-43. Population by sector and distance from Yucca Mountain used to calculate regional airborne consequences.[a]

	Distance from the repository (kilometers)[b]										
Direction	6[c]	16	24	32	40	48	56	64	72	84	*Totals*[d]
S	0	0	16	238	430	123	0	10	0	0	*817*
SSW	0	0	0	315	38	0	0	7	0	0	*360*
SW	0	0	0	0	0	0	868	0	0	0	*868*
WSW	0	0	0	0	0	0	0	0	87	0	*87*
W	0	0	0	638	17	0	0	0	0	0	*655*
WNW	0	0	0	936	0	0	0	0	0	20	*956*
NW	0	0	0	28	2	0	0	0	33	0	*63*
NNW	0	0	0	0	0	0	0	0	0	0	*0*
N	0	0	0	0	0	0	0	0	0	0	*0*
NNE	0	0	0	0	0	0	0	0	0	0	*0*
NE	0	0	0	0	0	0	0	0	0	0	*0*
ENE	0	0	0	0	0	0	0	0	0	0	*0*
E	0	0	0	0	0	0	0	0	0	0	*0*
ESE	0	0	0	0	0	0	0	0	1,055	0	*1,055*
SE	0	0	0	0	3	0	13	0	0	206	*222*
SSE	0	0	0	0	23	172	6	17	6,117	16,399	*22,734*
Totals	**0**	**0**	**16**	**2,155**	**513**	**295**	**887**	**34**	**7,292**	**16,625**	***27,817***

a. Source: Compiled from DOE (1998a, Volume 3, Figure 3-76).
b. To convert kilometers to miles, multiply by 0.62137.
c. The 80-kilometer (50-mile) distance typically used in an EIS analysis was increased to 84 kilometers (52 miles) in order to include the population of Pahrump in the SSE sector in the calculations.
d. Population figures are estimates for 2000.

A population radiological dose factor of 2.2×10^{-9} person-rem per microcurie per year of release was calculated by the GENII code. For a 0.098-microcurie-per-year release, this corresponds to a 7.8×10^{-15}-rem-per-year average radiological dose to individuals in the population. Thus, a maximum 84-kilometer (52-mile) population radiological dose rate would be 2.2×10^{-10} person-rem per year. This radiological dose rate represents 1.1×10^{-13} latent cancer fatalities in the regional population of 28,000

Table I-44. Meteorologic joint frequency data used for Yucca Mountain atmospheric releases (percent of time).[a]

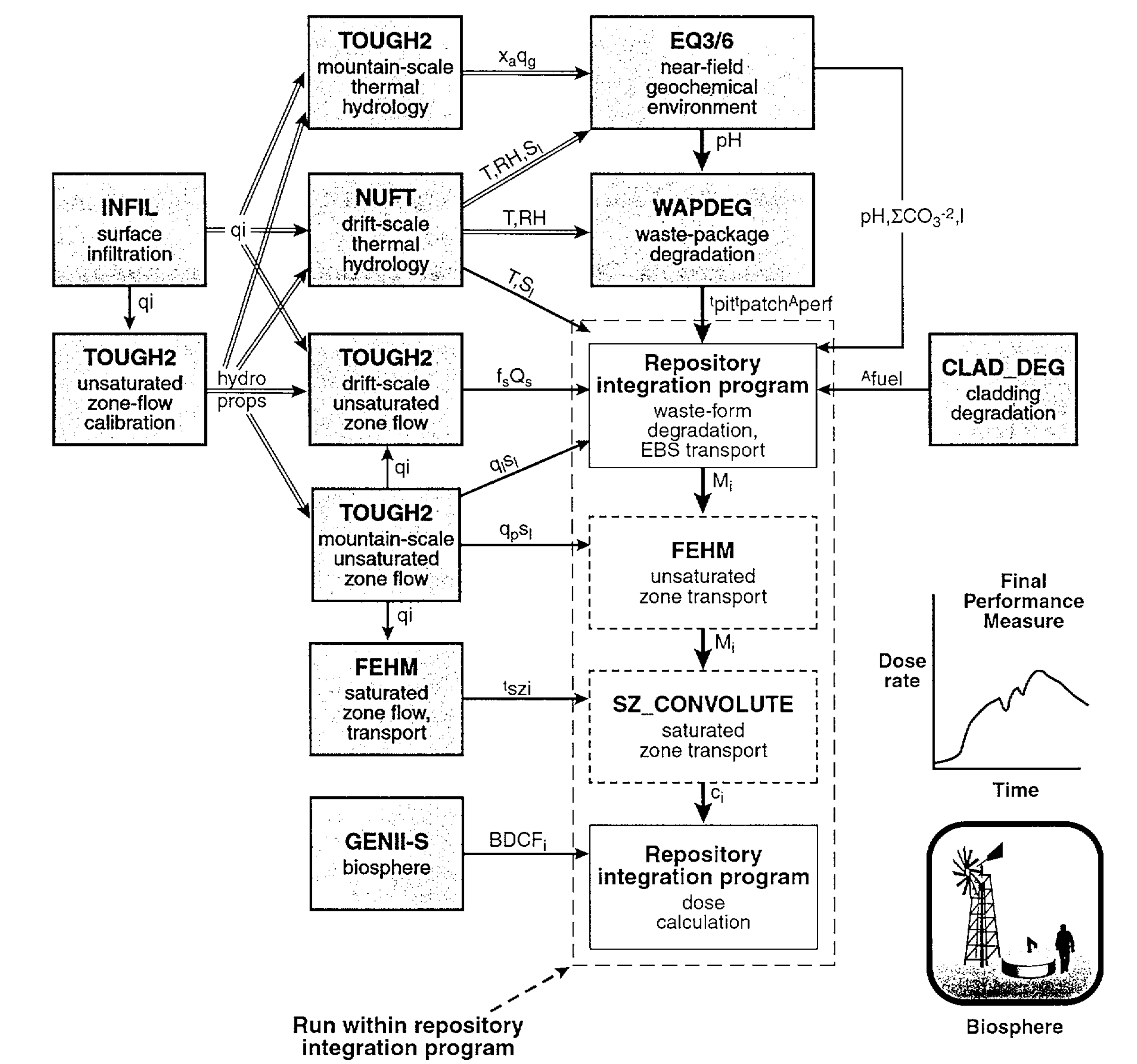
Average wind speed (m/s)[b]	Atmospheric stability class	Direction (wind toward) S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
0.9	A	0.807	0.633	0.613	0.520	0.462	0.604	0.688	0.659	0.467	0.340	0.183	0.200	0.197	0.212	0.412	0.778
	B	0.279	0.479	0.392	0.325	0.372	0.540	1.243	2.279	1.484	0.499	0.290	0.192	0.105	0.070	0.087	0.305
	C	0.113	0.105	0.064	0.017	0.015	0.020	0.041	0.157	0.122	0.067	0.055	0.020	0.012	0.020	0.009	0.032
	D	0.003	0.003	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
2.55	A	0.099	0.073	0.026	0.020	0.026	0.017	0.023	0.061	0.041	0.029	0.023	0.017	0.029	0.029	0.052	0.096
	B	0.058	0.044	0.038	0.026	0.032	0.061	0.125	0.377	0.360	0.070	0.049	0.015	0.009	0	0.009	0.017
	C	0.229	0.267	0.256	0.116	0.110	0.105	0.328	1.193	2.404	0.909	0.671	0.302	0.157	0.142	0.125	0.174
	D	0.105	0.049	0.038	0.003	0.003	0.003	0.006	0.035	0.444	0.290	0.206	0.055	0.035	0.049	0.087	0.099
	E	0.003	0.006	0	0.003	0	0	0.003	0.003	0.003	0.006	0.003	0.003	0.003	0.003	0	0.003
	F	0	0.003	0	0	0	0	0	0	0.003	0.003	0	0	0	0	0	0.003
4.35	A	0.096	0.096	0.041	0.015	0.012	0.009	0.015	0.023	0.058	0.044	0.026	0.023	0.029	0.020	0.020	0.070
	B	0.052	0.087	0.041	0.023	0.006	0.026	0.078	0.261	0.305	0.131	0.076	0.017	0.006	0.003	0.009	0.032
	C	0.142	0.241	0.168	0.070	0.029	0.076	0.131	0.740	1.638	0.308	0.290	0.119	0.049	0.041	0.038	0.102
	D	0.253	0.264	0.163	0.049	0.020	0.020	0.020	0.392	2.375	0.447	0.285	0.081	0.046	0.058	0.139	0.346
	E	0.006	0.017	0	0	0	0	0	0.003	0.006	0.020	0.015	0.006	0.003	0.003	0.012	0.020
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6.95	A	1.568	0.642	0.215	0.038	0.035	0.009	0.023	0.026	0.081	0.142	0.261	0.163	0.209	0.314	0.343	0.819
	B	0.682	0.552	0.067	0.003	0.006	0.006	0.023	0.058	0.348	0.325	0.267	0.131	0.078	0.093	0.078	0.256
	C	0.993	0.560	0.105	0.012	0.009	0.078	0.090	0.244	0.984	0.526	0.337	0.192	0.067	0.076	0.073	0.189
	D	1.594	0.912	0.183	0.020	0.020	0.006	0.035	0.566	3.368	0.430	0.160	0.128	0.035	0.044	0.142	0.598
	E	0.735	0.366	0.067	0.012	0.006	0	0	0.386	2.515	0.192	0.038	0.015	0	0.015	0.064	0.804
	F	0.238	0.096	0.003	0	0.003	0	0	0.142	1.641	0.055	0.032	0	0.003	0.003	0.029	0.796
9.75	A	2.134	0.935	0.218	0.078	0.029	0.041	0.026	0.070	0.163	0.232	0.203	0.232	0.267	0.372	0.587	1.388
	B	0.865	0.627	0.081	0.009	0.003	0.017	0.020	0.046	0.319	0.267	0.154	0.131	0.070	0.052	0.113	0.302
	C	0.720	0.261	0.038	0.012	0.020	0.020	0.009	0.076	0.502	0.299	0.148	0.229	0.078	0.032	0.041	0.157
	D	0.415	0.212	0.020	0.003	0.003	0.003	0.003	0.046	0.627	0.154	0.044	0.032	0.029	0.009	0.026	0.145
	E	0.029	0.006	0	0	0.003	0	0	0	0.006	0.003	0.003	0	0	0.003	0	0.003
	F	0	0.003	0	0	0	0	0	0	0	0	0	0	0	0.003	0	0.003
12.98	A	1.661	0.706	0.418	0.322	0.247	0.244	0.366	0.343	0.407	0.380	0.302	0.299	0.357	0.537	1.083	2.038
	B	0.836	0.668	0.253	0.107	0.157	0.116	0.264	0.499	0.674	0.404	0.270	0.171	0.122	0.096	0.232	0.950
	C	0.322	0.267	0.087	0.017	0.006	0.012	0.026	0.136	0.311	0.107	0.032	0.029	0.020	0.009	0.015	0.038
	D	0.006	0.006	0	0	0	0	0	0.003	0.012	0.003	0	0	0	0	0	0.003
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

a. Source: Adapted from data in TRW (1999d, Appendix B, all)
b. To convert meters per second to feet per second, multiply by 3.2808.

persons each year at the maximum release rate. This annual population radiological dose rate corresponds to a lifetime radiological dose of 1.5×10^{-8} rem over a 70-year lifetime, which corresponds to 7.6×10^{-12} latent cancer fatalities during the 70-year period of the maximum release.

I.7.3 SENSITIVITY TO THE FRACTION OF EARLY-FAILED CLADDING

DOE performed a sensitivity analysis in which all of the cladding on commercial spent nuclear fuel that had stainless-steel cladding (about 1.3 percent of the fuel by volume) was assumed to fail immediately as the waste package failed. The commercial spent nuclear fuel with zirconium-alloy cladding was assumed to fail as shown in Figure I-57. The number of latent cancer fatalities per year in the local population at the time of maximum release would increase from 1.1×10^{-13} to 4.0×10^{-11} under the sensitivity analysis assumptions. The time of maximum release would be 2,000 years after repository closure rather than 19,000 years after repository closure.



OUTPUT Parameters

T	Temperature	Q_s	Seep flow rate
RH	Relative humidity	pH	pH
S_l	Liquid saturation	ΣCO_3^{-2}	Carbonate concentration
X_a	Air mass fraction	I	Ionic strength
q_g	Gas flux	t_{pit}	Initial-pit-penetration time
q_l	Liquid flux	t_{patch}	Initial-patch-penetration time
qi	Infiltration flux	A_{perf}	Perforated container area
M_i	Radionuclide mass flux	A_{fuel}	Exposed fuel area
C_i	Radionuclide concentration	t_{szi}	Saturated zone transport time
f_s	Fraction of WPs with seeps	$BDCF_i$	Biosphere dose conversion factor
		EBS	Engineered Barrier System

Legend

- External Process Model
- Repository Integration Program Cells
- Repository Integration Program Calls External Code
- Response Surface Between Process Models
- Response Surface from Process Model to RIP
- Between RIP Cells and External Code

Source: Modified from DOE (1998a, Volume 3, Figure 2-13).

Figure I-1. Total system performance assessment model.

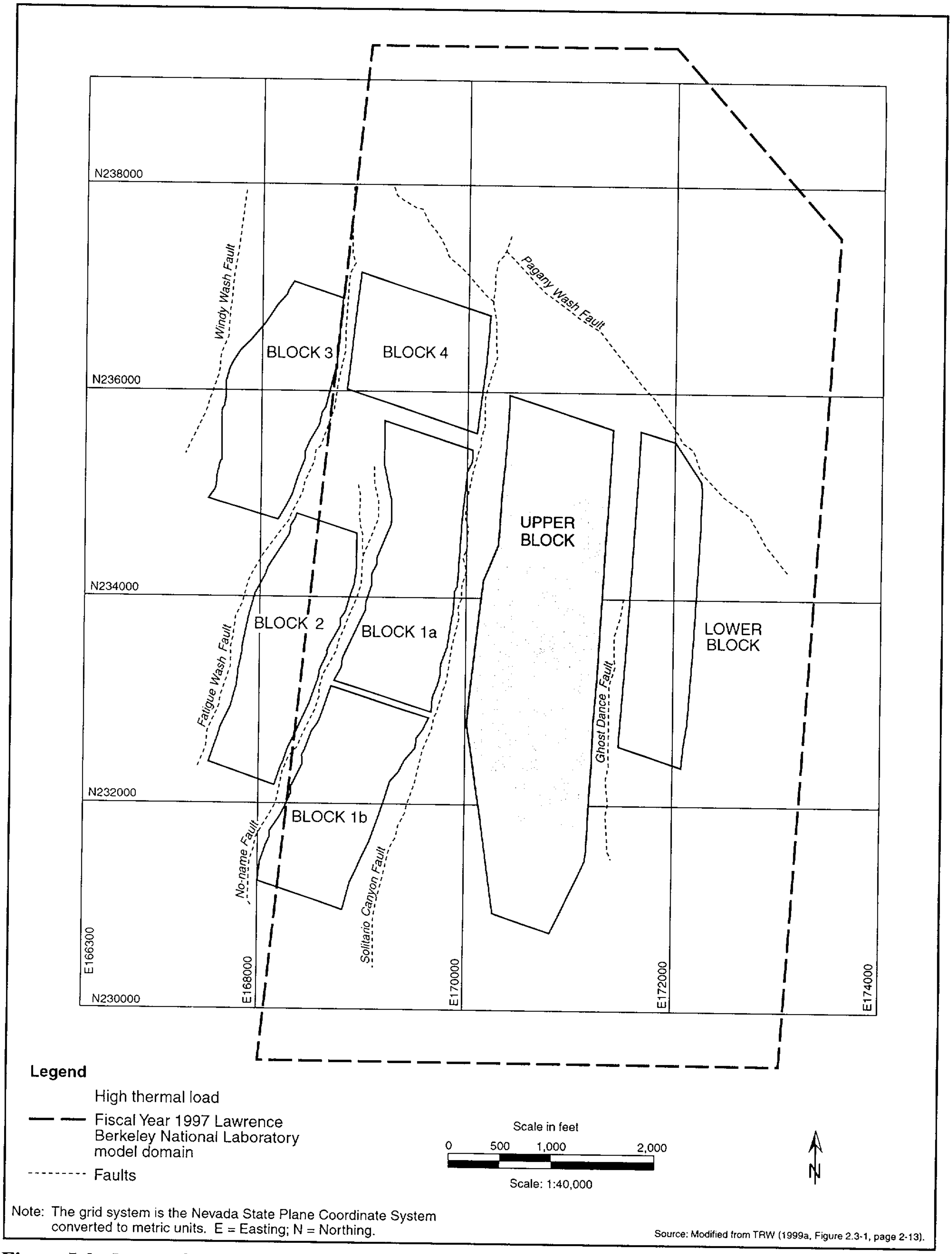


Figure I-2. Layout for Proposed Action inventory for high thermal load (85 MTHM per acre) scenario.

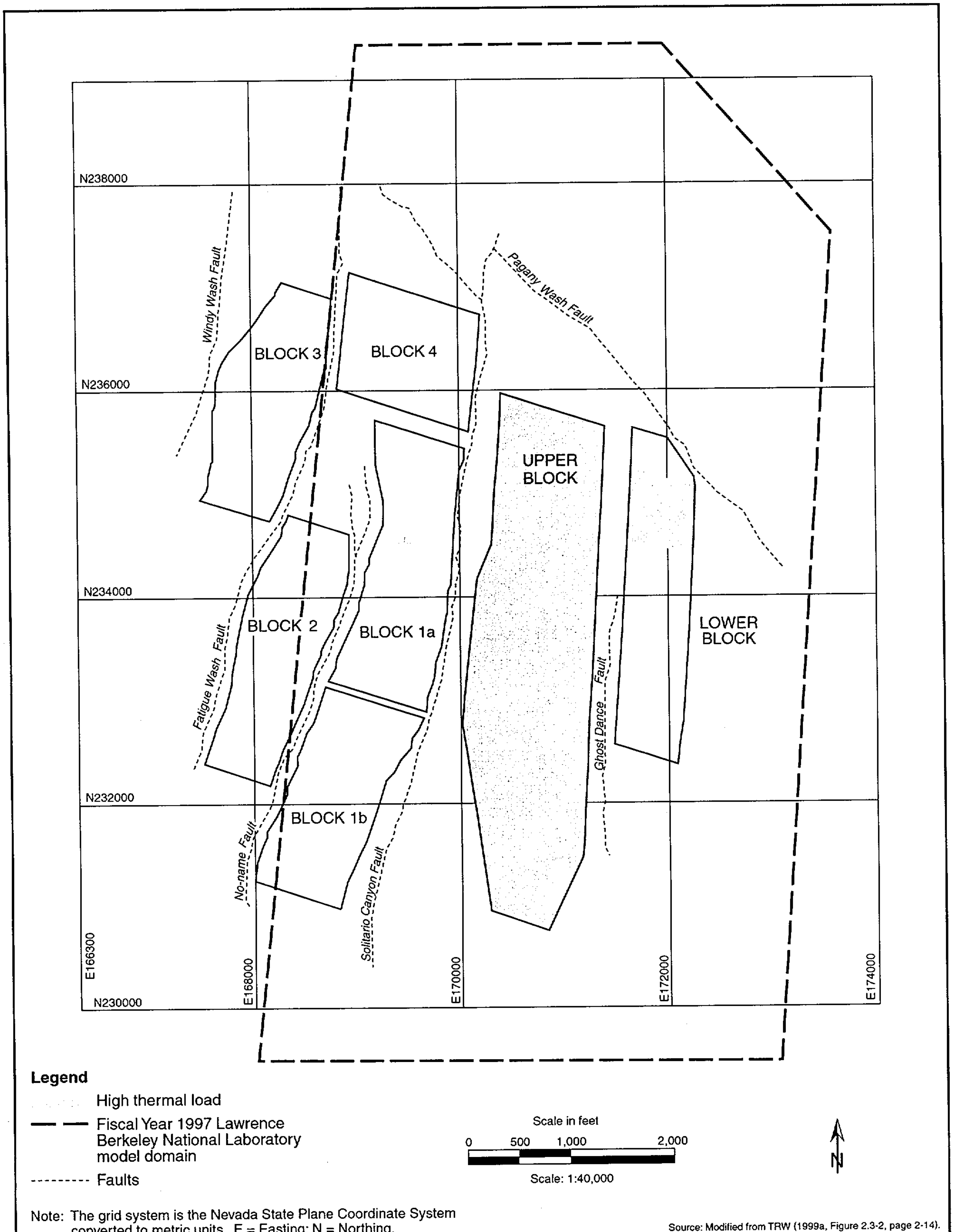


Figure I-3. Layout for Inventory Modules 1 and 2 for high thermal load (85 MTHM per acre) scenario.

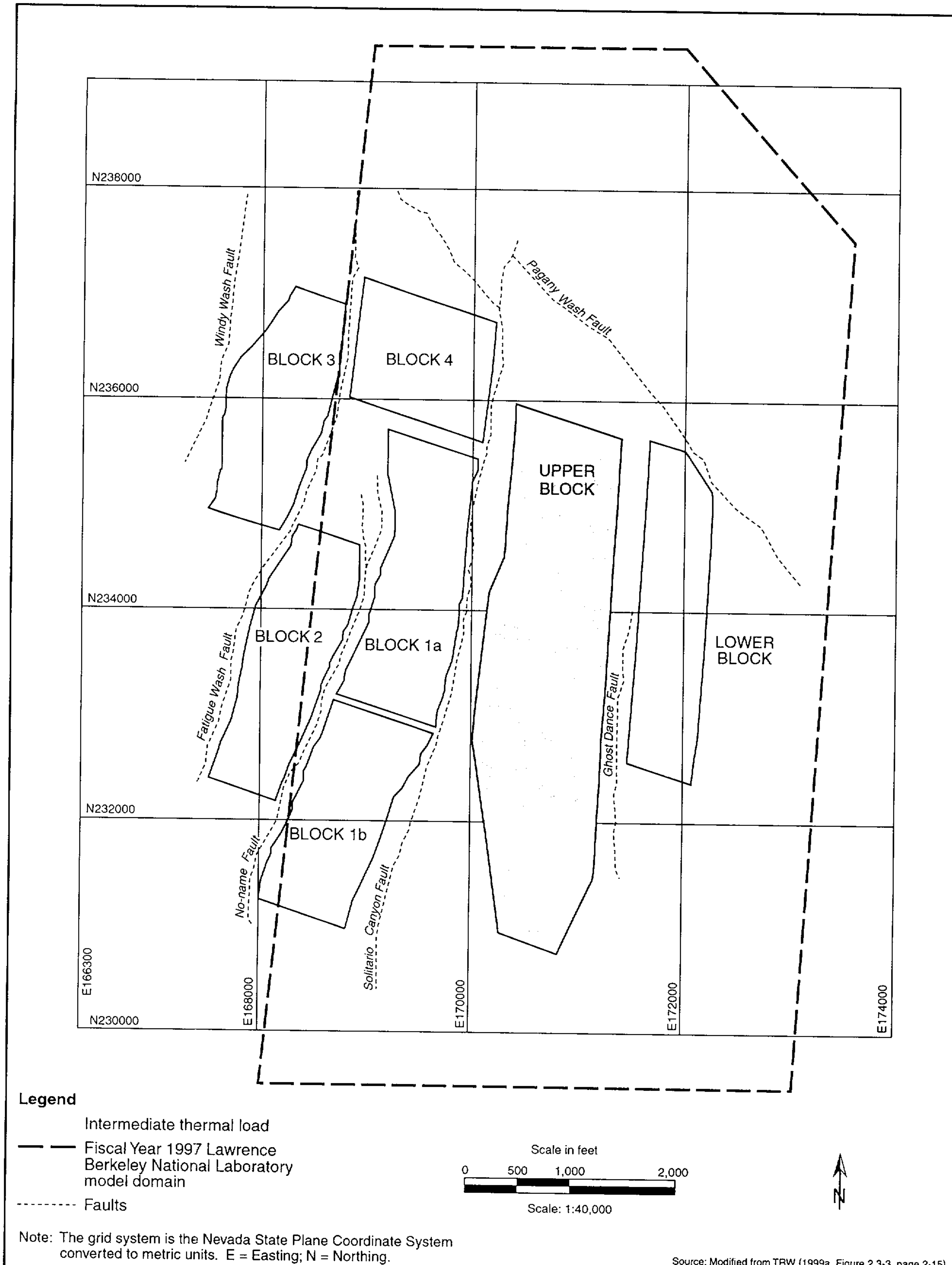


Figure I-4. Layout for Proposed Action inventory for intermediate thermal load (60 MTHM per acre) scenario.

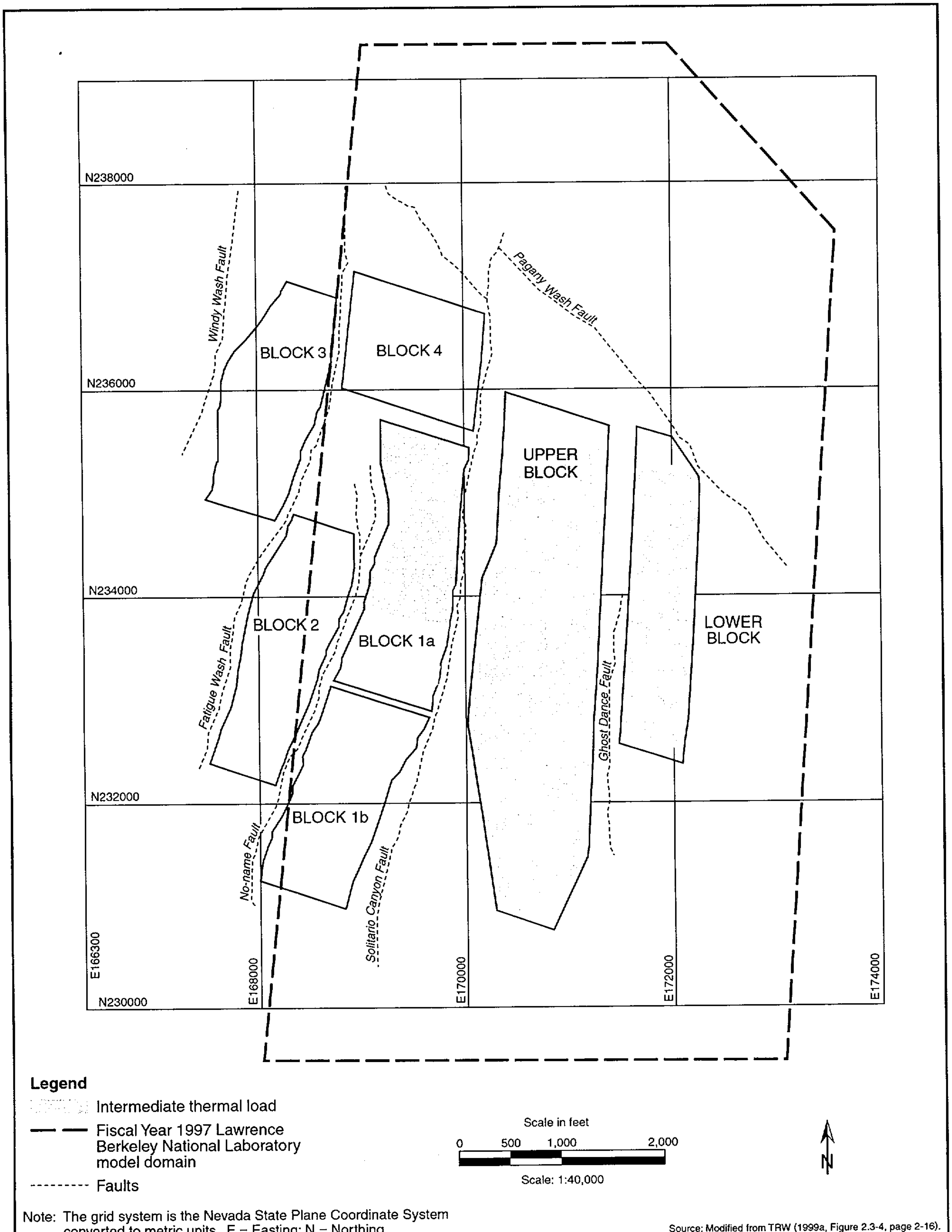
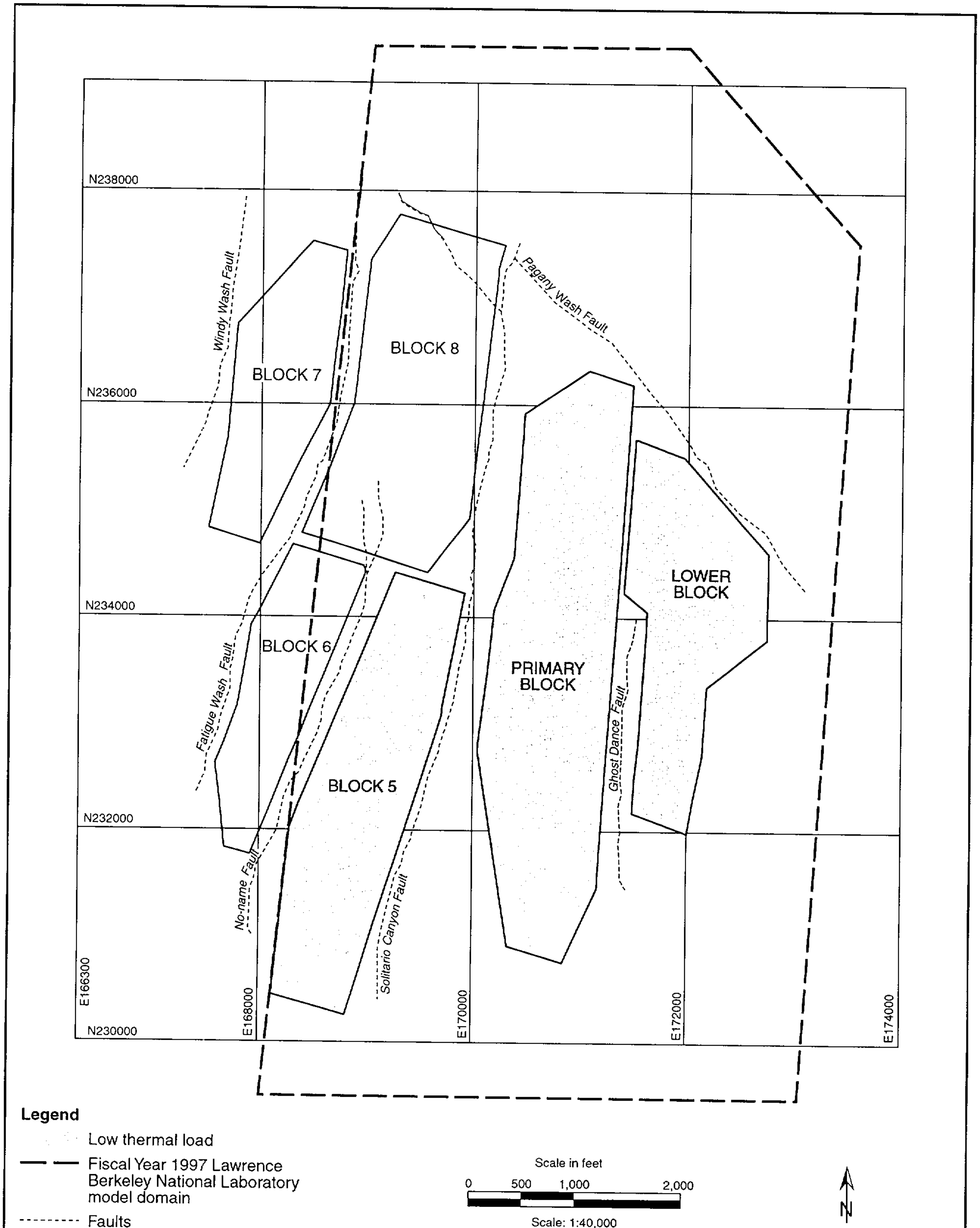


Figure I-5. Layout for Inventory Modules 1 and 2 for intermediate thermal load (60 MTHM per are) scenario.



Note: The grid system is the Nevada State Plane Coordinate System converted to metric units. E = Easting; N = Northing.

Source: Modified from TRW (1999a, Figure 2.3-5, page 2-17).

Figure I-6. Layout for Proposed Action inventory for low thermal load (25 MTHM per acre) scenario.

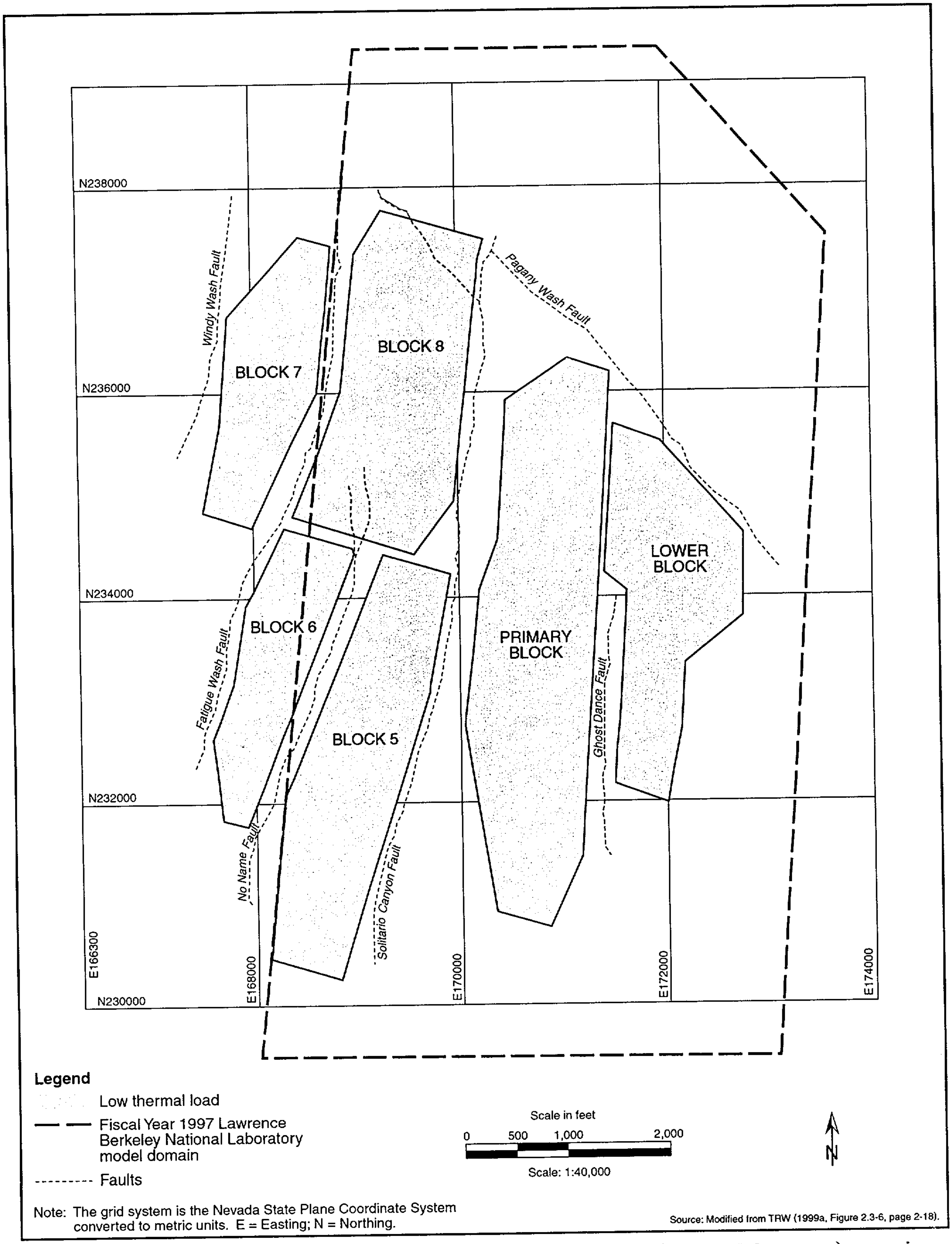


Figure I-7. Layout for Inventory Modules 1 and 2 for low thermal load (25 MTHM per acre) scenario.

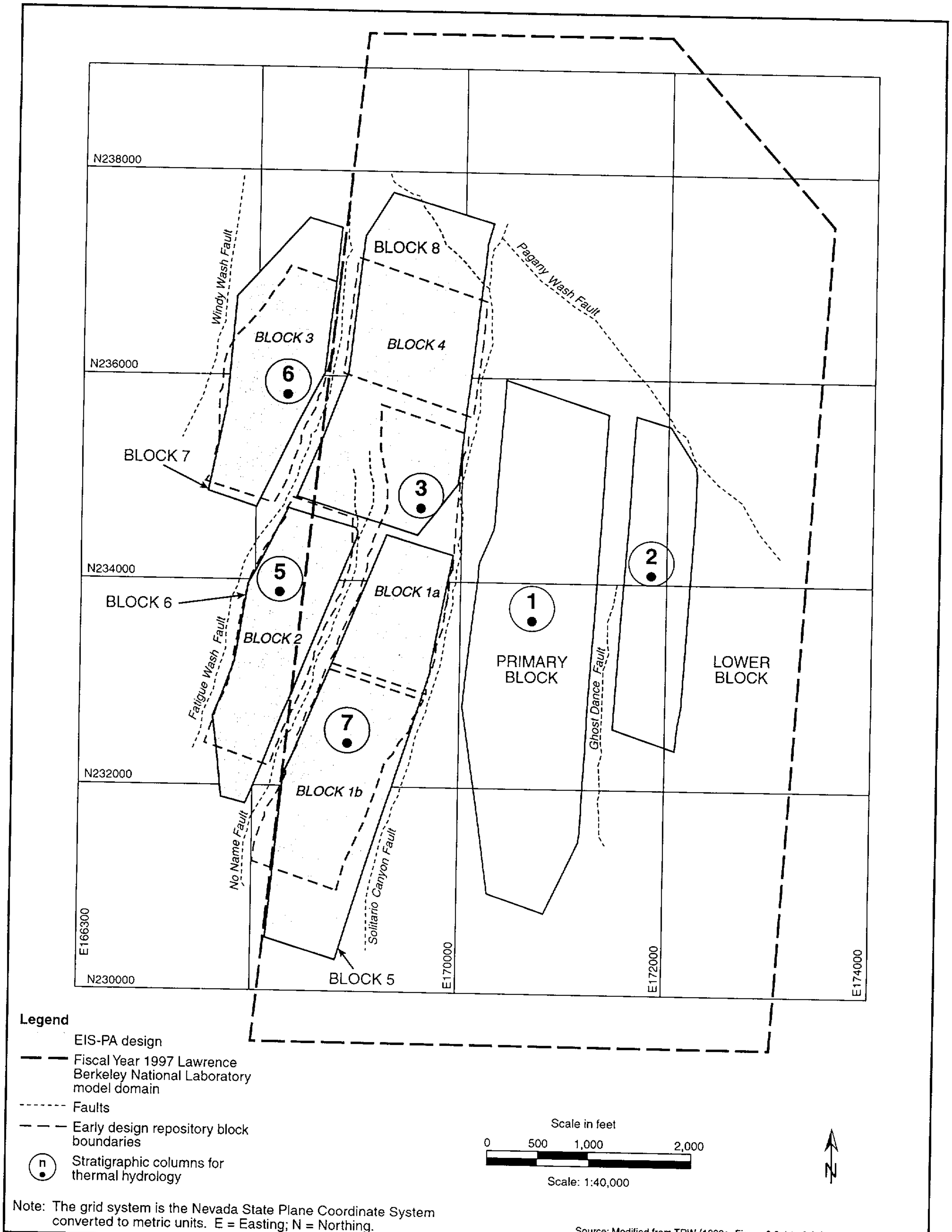


Figure I-8. Relationship between the early performance assessment design and emplacement block layout considered in this EIS performance assessment analysis.

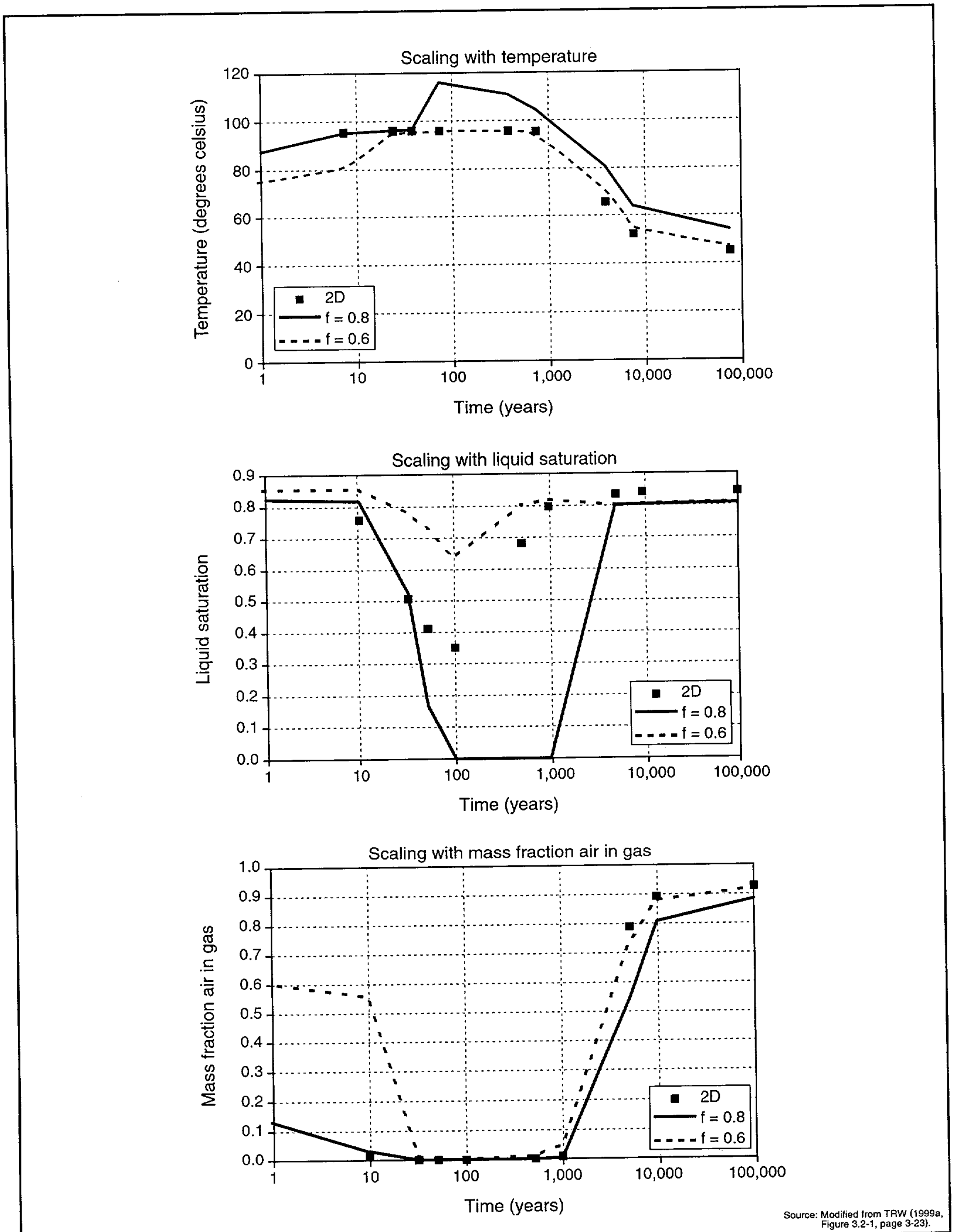
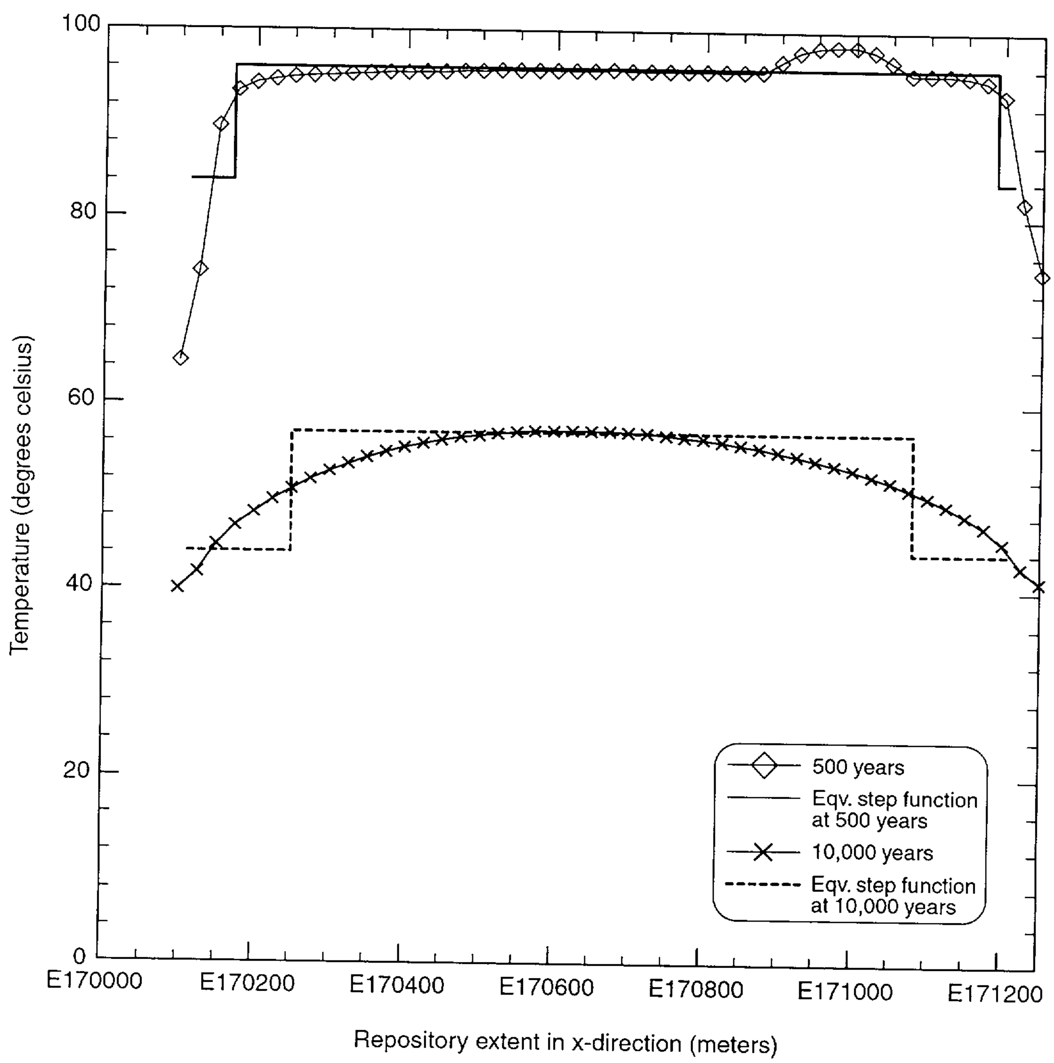


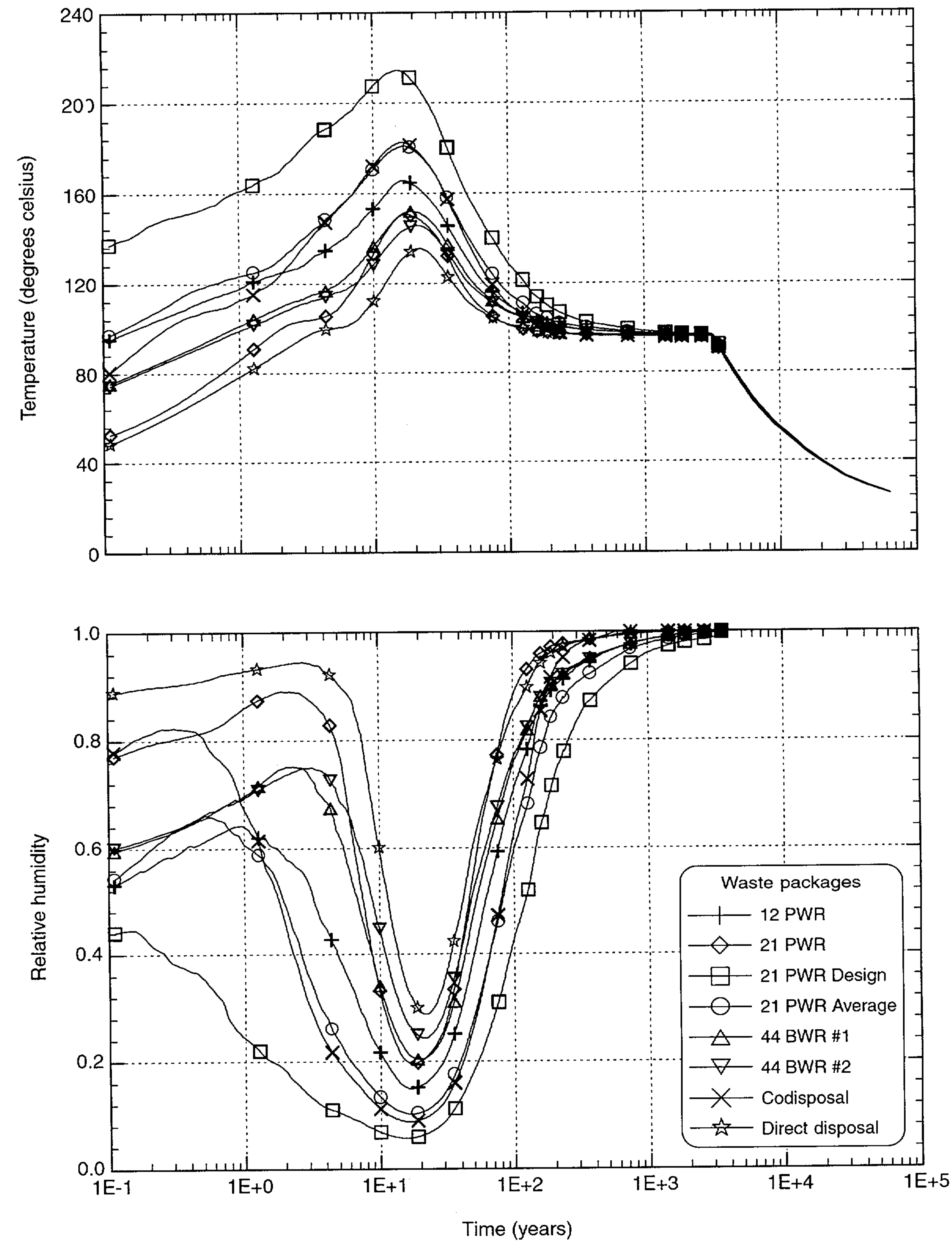
Figure I-9. Development of thermal load scale factors on the basis of two-dimensional and one-dimensional model comparisons using time history of temperature, liquid saturation, and air mass fraction.



Note: The grid system is the Nevada State Plane Coordinate System converted to metric units. E = Easting; N = Northing.

Source: Modified from TRW (1999a, Figure 3.2-2, page 3-24).

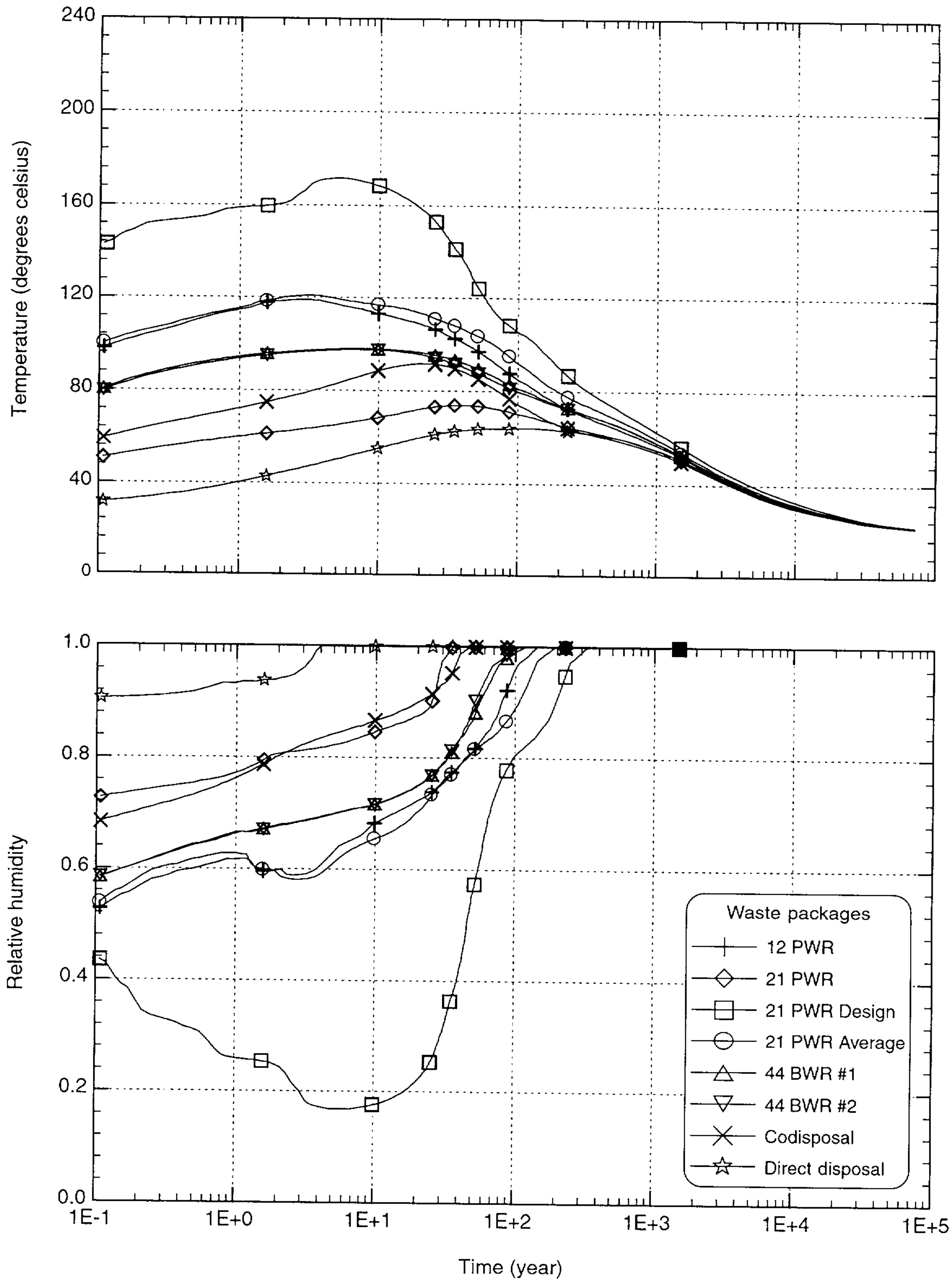
Figure I-10. Partition of repository area between center and edge regions.



BWR = boiling-water reactor; PWR = presurized-water reactor.

Source: Modified from TRW (1999a, Figure 3.2-3, page 3-25).

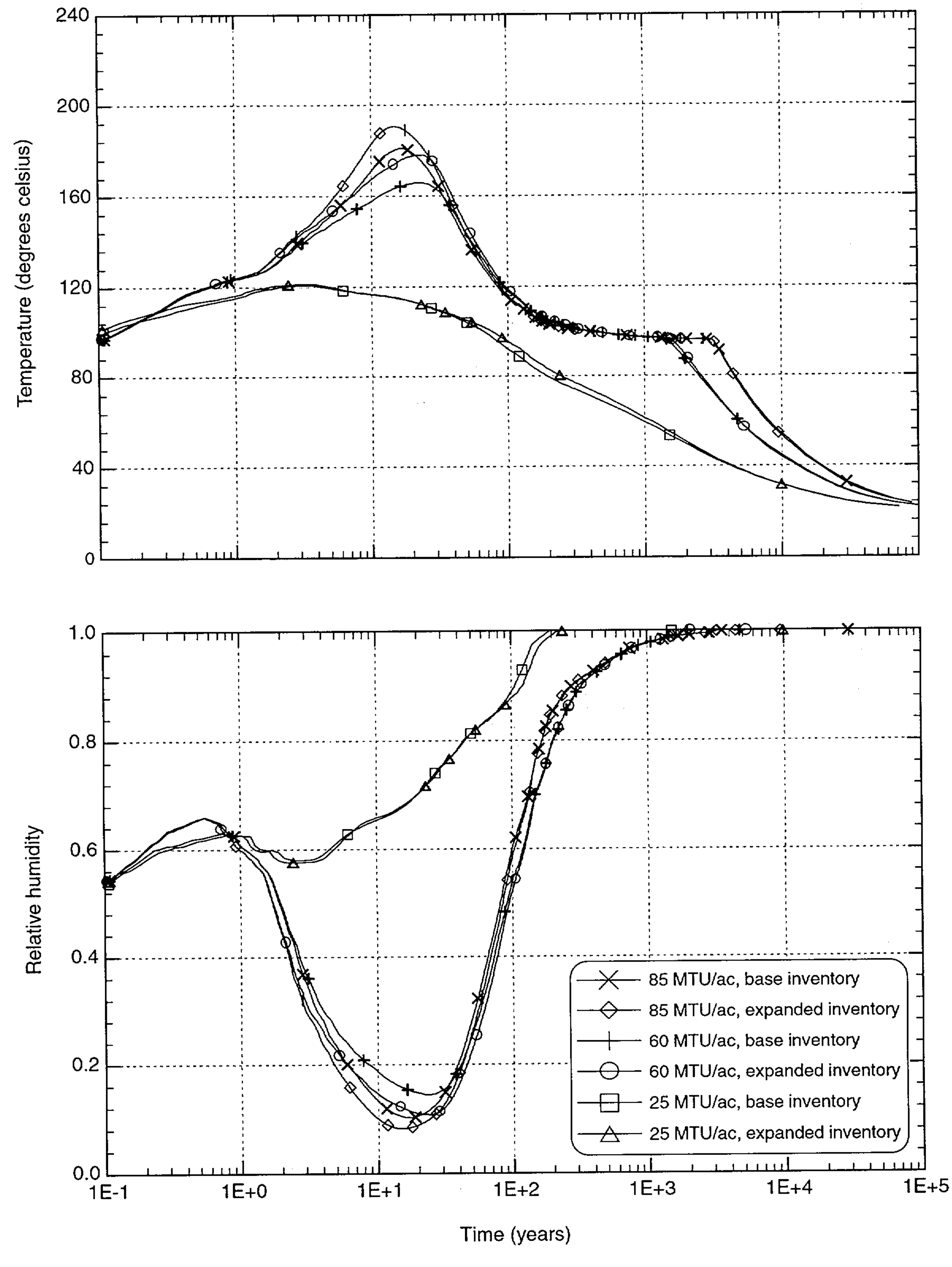
Figure I-11. Temperature and relative humidity histories for all waste packages for high thermal load scenario, Proposed Action inventory, and long-term average climate.



BWR = boiling-water reactor; PWR = presurized-water reactor.

Source: Modified from TRW (1999a, Figure 3.2-4, page 3-26).

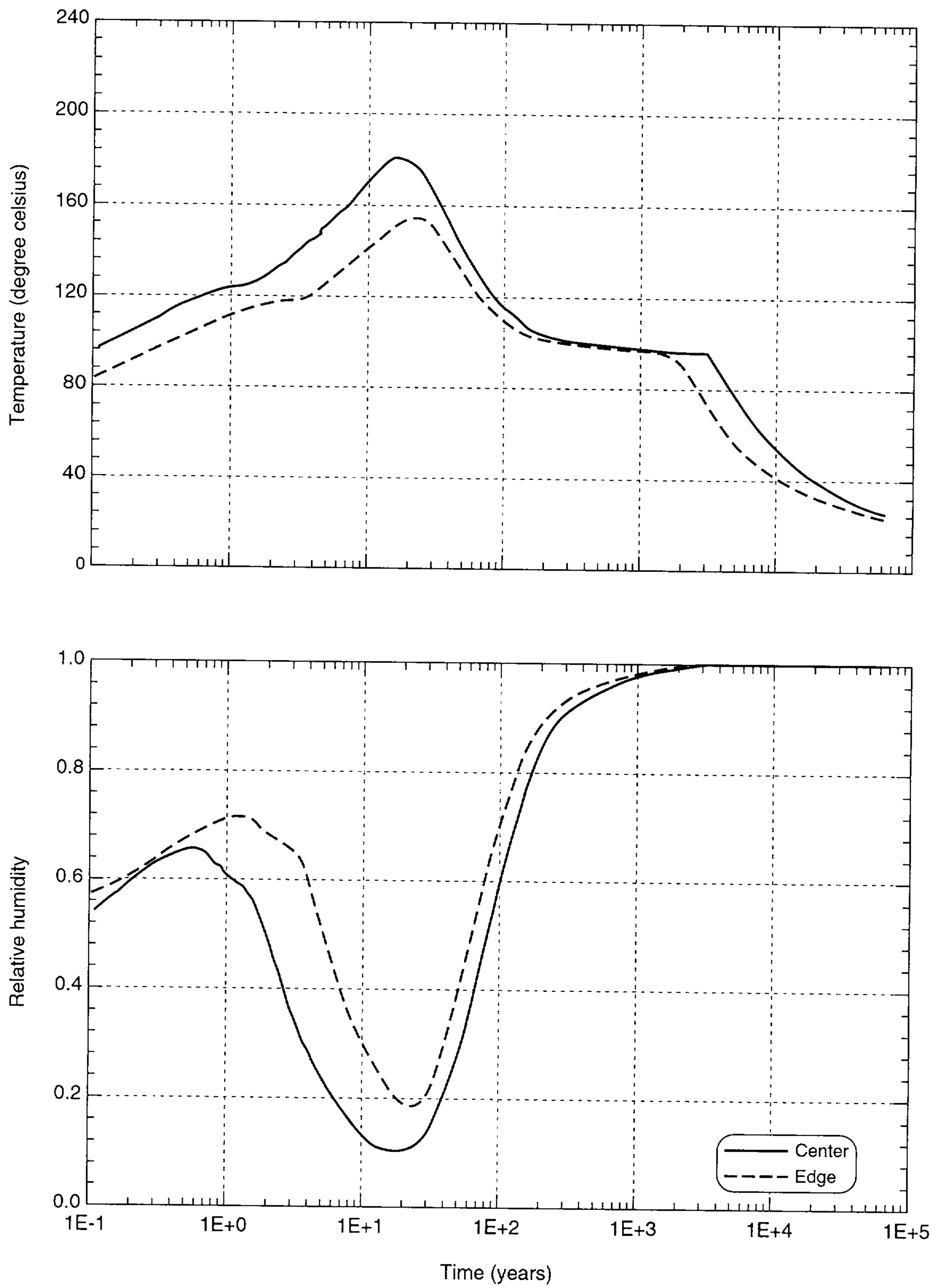
Figure I-12. Temperature and relative humidity histories for all waste packages, low thermal load scenario, Proposed Action inventory, and long-term average climate.



MTU/ac = metric tons of uranium per acre.

Source: Modified from TRW (1999a, Figure 3.2-5, page 3-27).

Figure I-13. Temperature and relative humidity histories for the 21 pressurized-water-reactor average waste packages, long-term average climate scenario, showing sensitivity to waste inventory.



Source: Modified from TRW (1999a, Figure 3.2-6, page 3-28).

Figure I-14. Temperature and relative humidity histories for the 21 pressurized-water-reactor average waste packages, high thermal load scenario, Proposed Action inventory, long-term average climate scenario, comparing the center and edge scenarios.

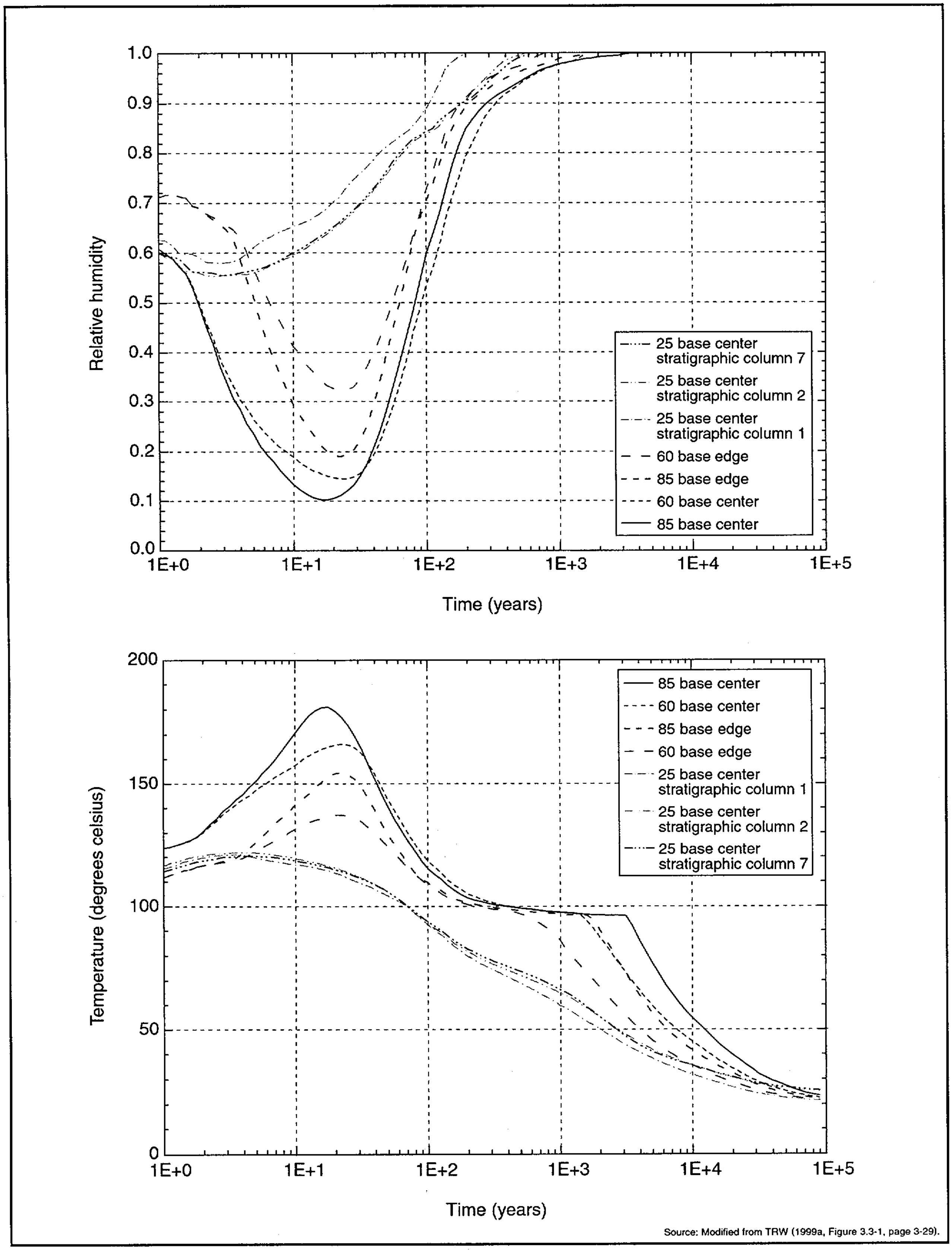


Figure I-15. WAPDEG input temperature and relative humidity histories for all thermal loads with Proposed Action inventory.

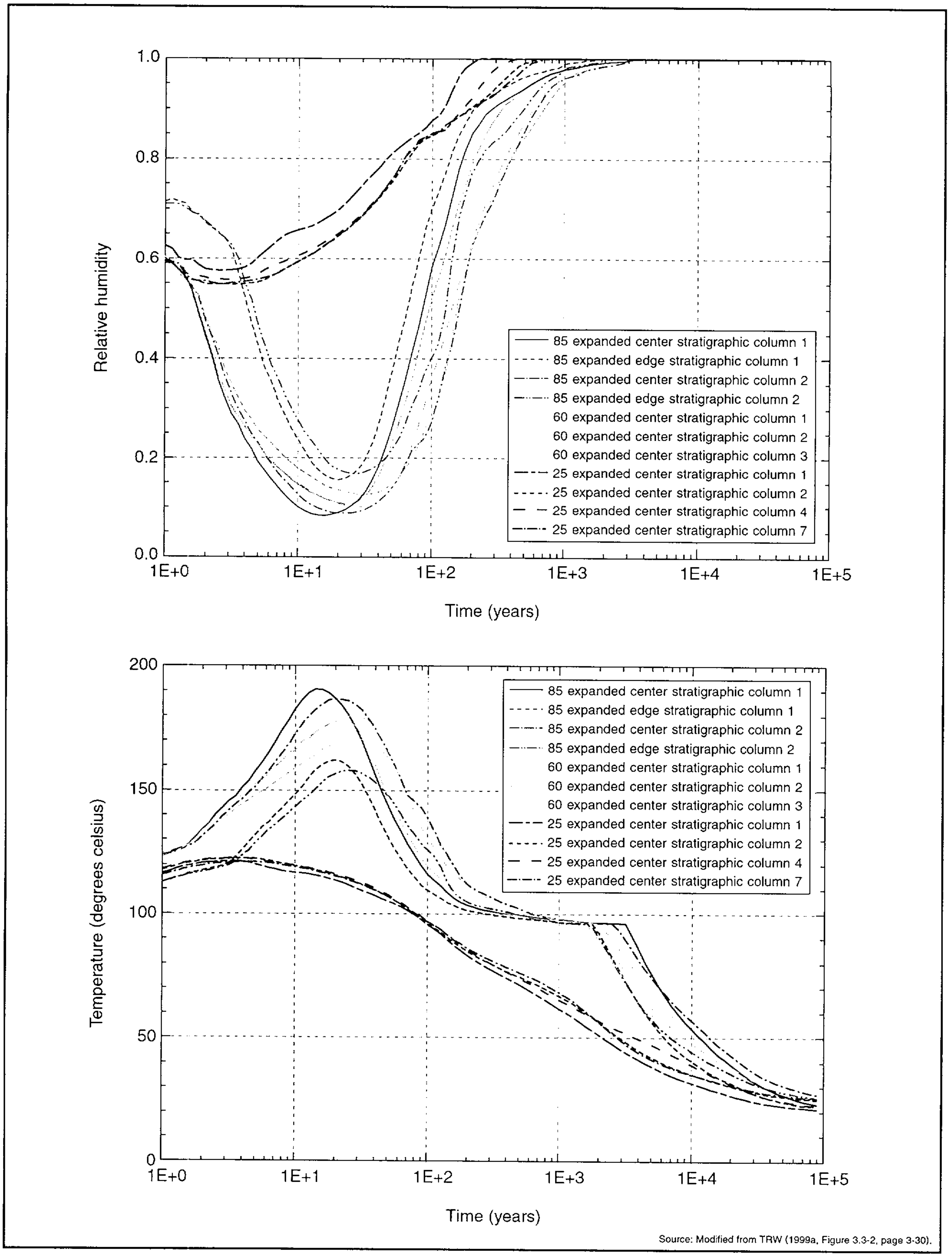
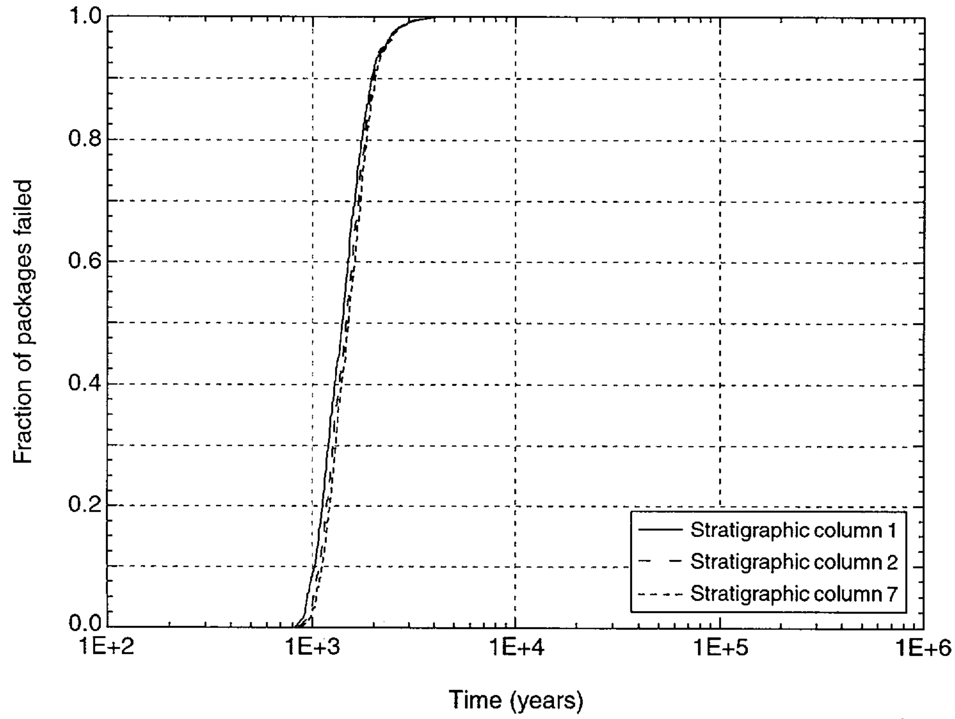
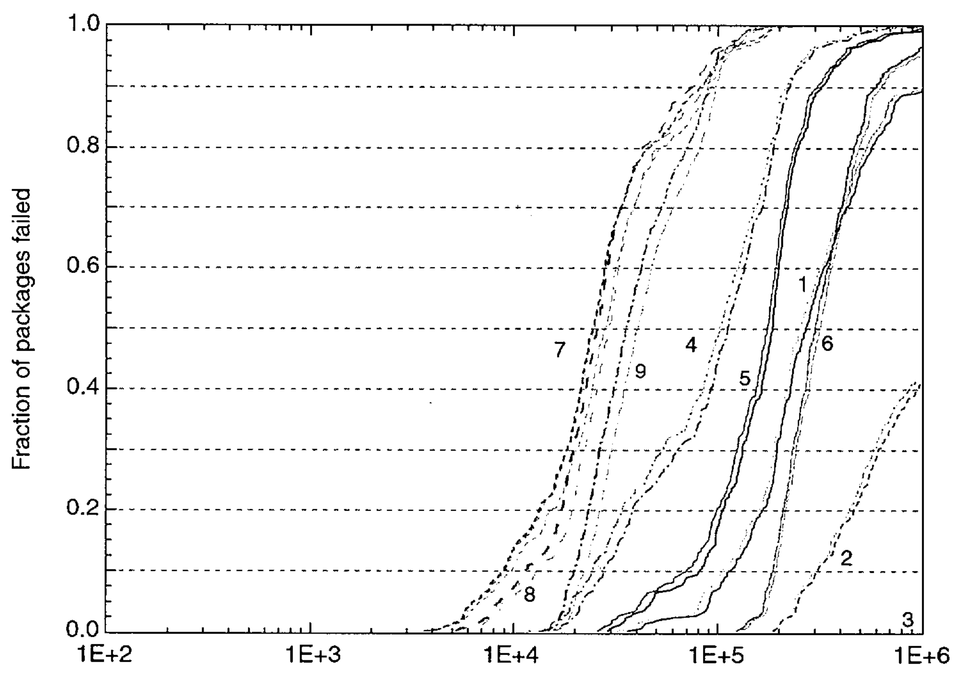


Figure I-16. WAPDEG input temperature and relative humidity histories for all thermal loads with Inventory Modules 1 and 2.



Source: Modified from TRW (1999a, Figure 3.3-3, page 3-31).

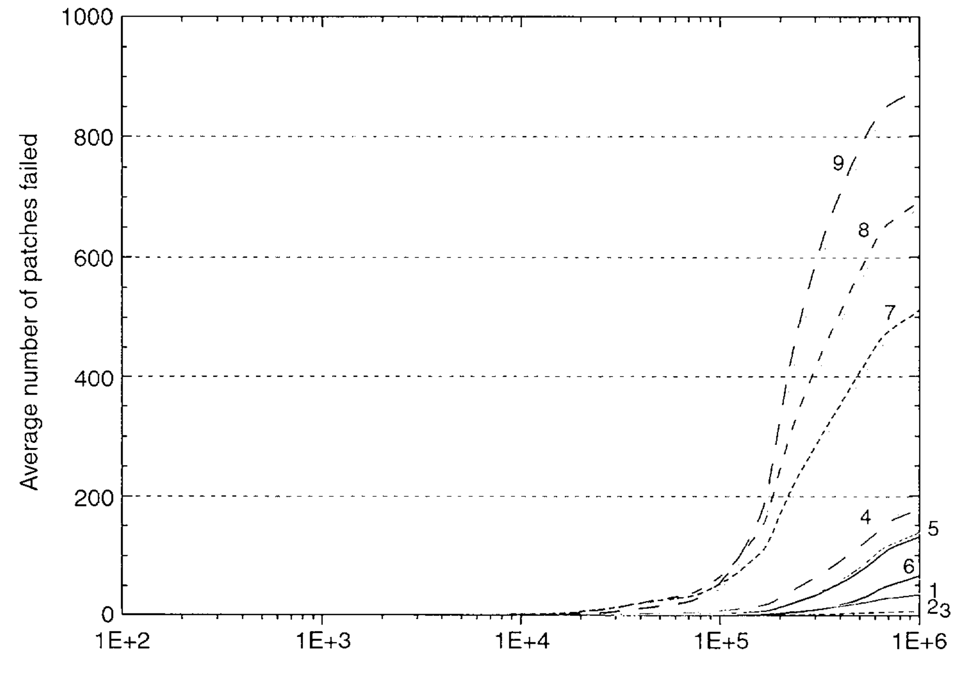
Figure I-17. Time to first breach of the corrosion-allowance material for low thermal load scenario, Proposed Action inventory, all three stratigraphic columns, always-dripping waste packages.



Numbers on plot refer to uncertainty/variability splitting sets shown on Table I-29.

Source: Modified from TRW (1999a, Figure 3.3-4, page 3-32).

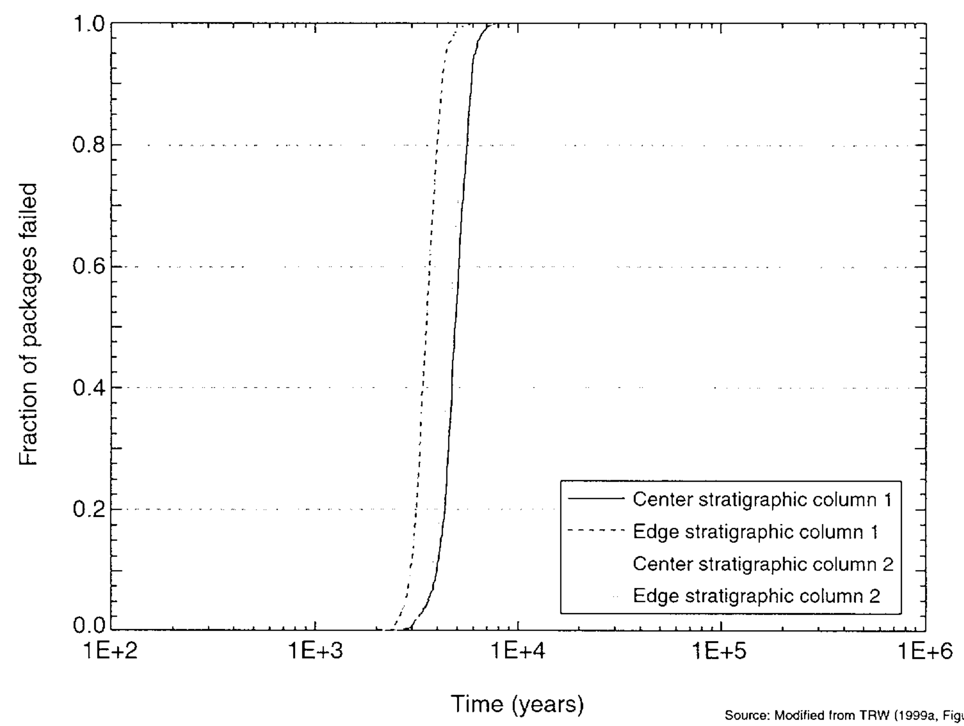
Figure I-18. Time to first breach of the corrosion-resistant material for low thermal load scenario, Proposed Action inventory, all three stratigraphic columns, always-dripping waste packages, and all nine uncertainty/variability splitting sets.



Numbers on plot refer to uncertainty/variability splitting sets shown on Table I-29.

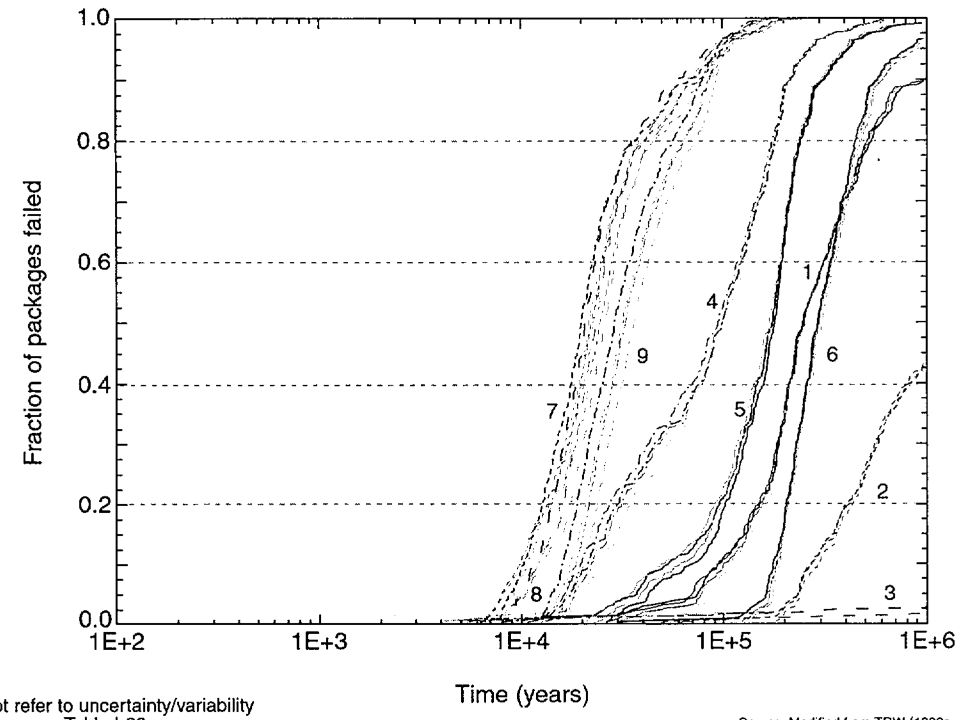
Source: Modified from TRW (1999a, Figure 3.3-5, page 3-33).

Figure I-19. Average number of patches failed per waste package as a function of time for low thermal load scenario, Proposed Action inventory, all three stratigraphic columns, always-dripping waste packages, and all nine uncertainty/variability splitting sets.



Source: Modified from TRW (1999a, Figure 3.3-6, page 3-34).

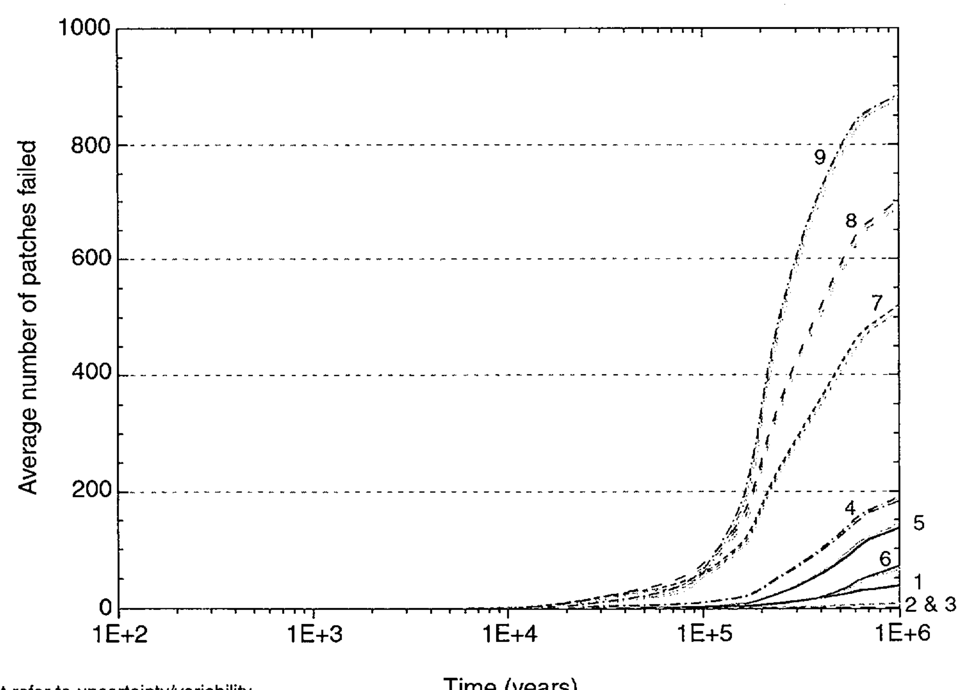
Figure I-20. Time to first breach of the corrosion-allowance material for high thermal load scenario, Inventory Modules 1 and 2, center and edge regions for both stratigraphic columns, always-dripping waste packages.



Numbers on plot refer to uncertainty/variability splitting sets shown on Table I-29.

Source: Modified from TRW (1999a, Figure 3.3-7, page 3-35).

Figure I-21. Time to first breach of the corrosion-resistant material for high thermal load scenario, Inventory Modules 1 and 2, center and edge regions for both stratigraphic columns, always-dripping waste packages, and all nine uncertainty/variability splitting sets.



Numbers on plot refer to uncertainty/variability splitting sets shown on Table I-29.

Source: Modified from TRW (1999a, Figure 3.3-8, page 3-36).

Figure I-22. Average number of patches failed per package as a function of time for high thermal load scenario, Inventory Modules 1 and 2, center and edge regions for both stratigraphic columns, always-dripping waste packages, and all nine uncertainty/variability splitting sets.

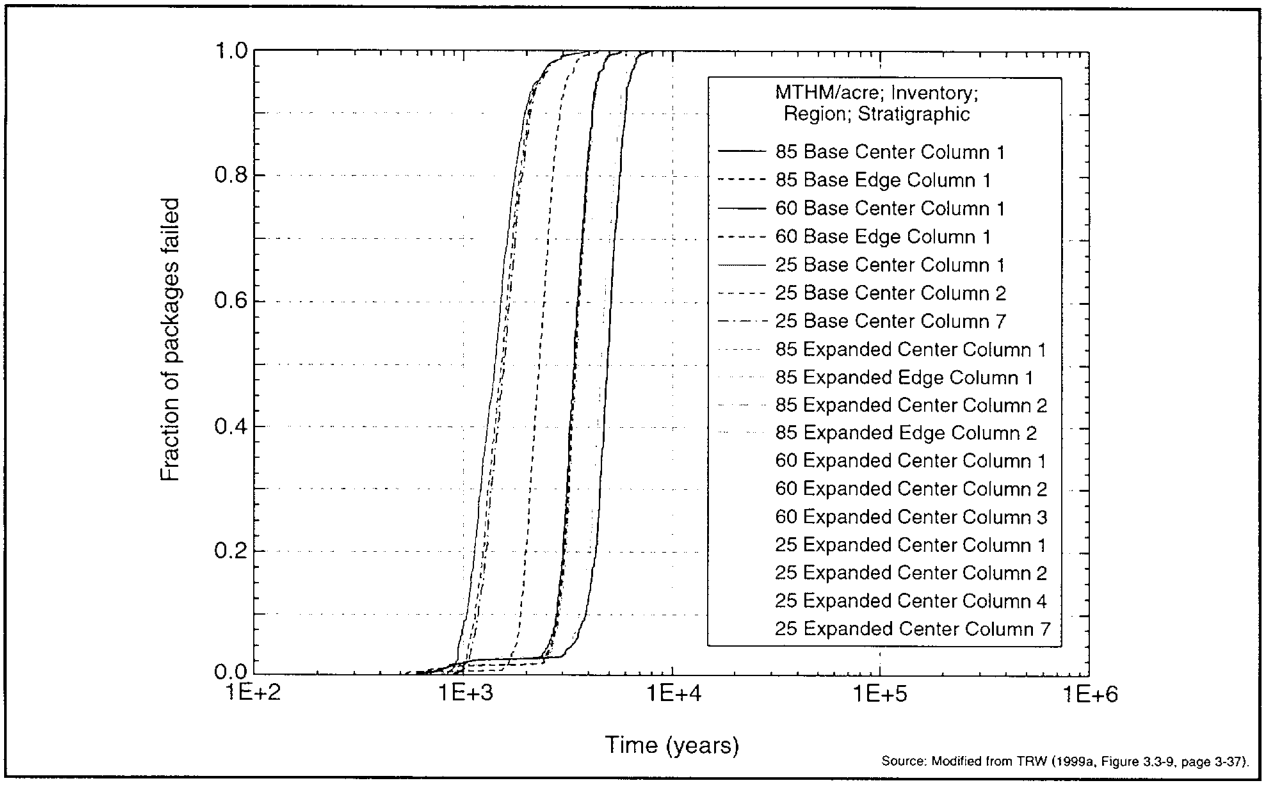


Figure I-23. Time to first breach of the corrosion-allowance material for all thermal loads and inventories, all regions, always-dripping waste packages, uncertainty/variability splitting set 5.

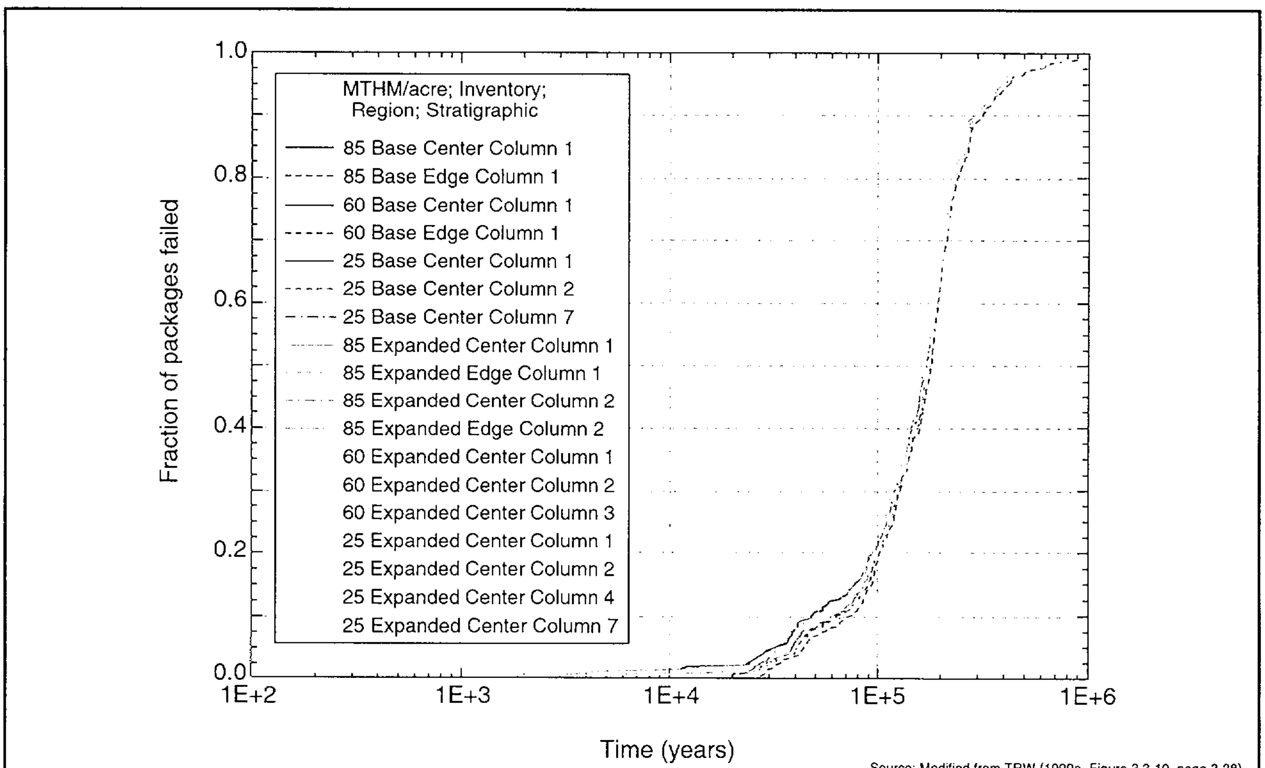


Figure I-24. Time to first breach of the corrosion-resistant material for all thermal loads and inventories, all regions, always-dripping waste packages, uncertainty/variability splitting set 5.

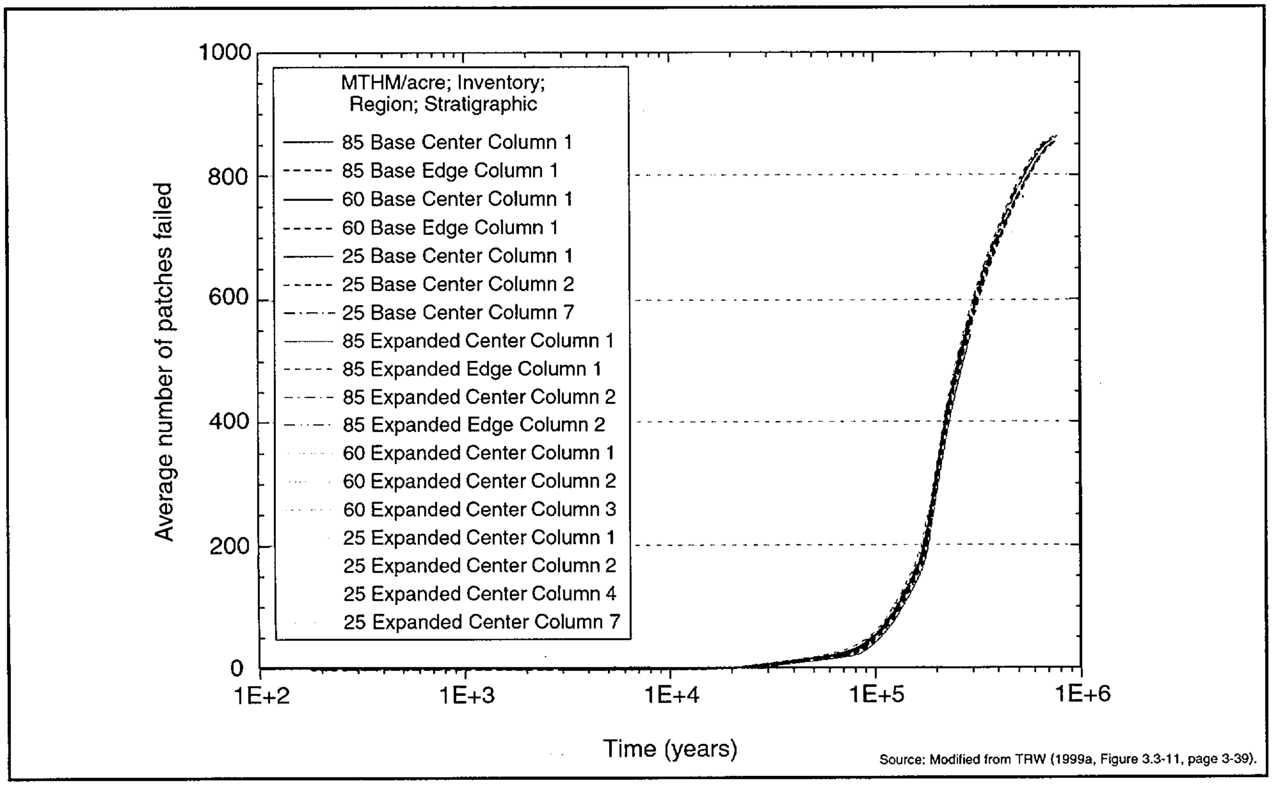


Figure I-25. Average number of patches failed per waste package as a function of time for all thermal loads and inventories, all regions, always-dripping waste packages, uncertainty/variability splitting set 9.

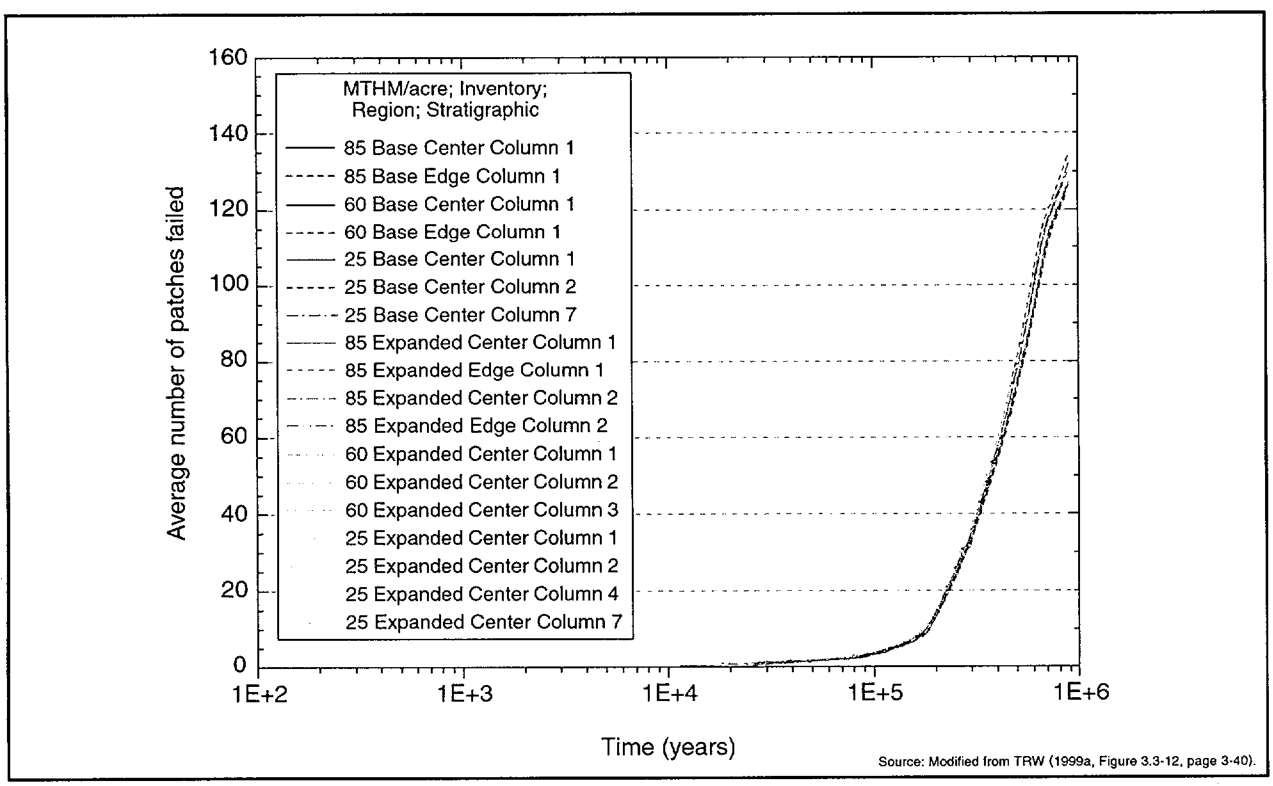


Figure I-26. Average number of patches failed per waste package as a function of time for all thermal loads and inventories, all regions, always-dripping waste packages, uncertainty/variability splitting set 5.

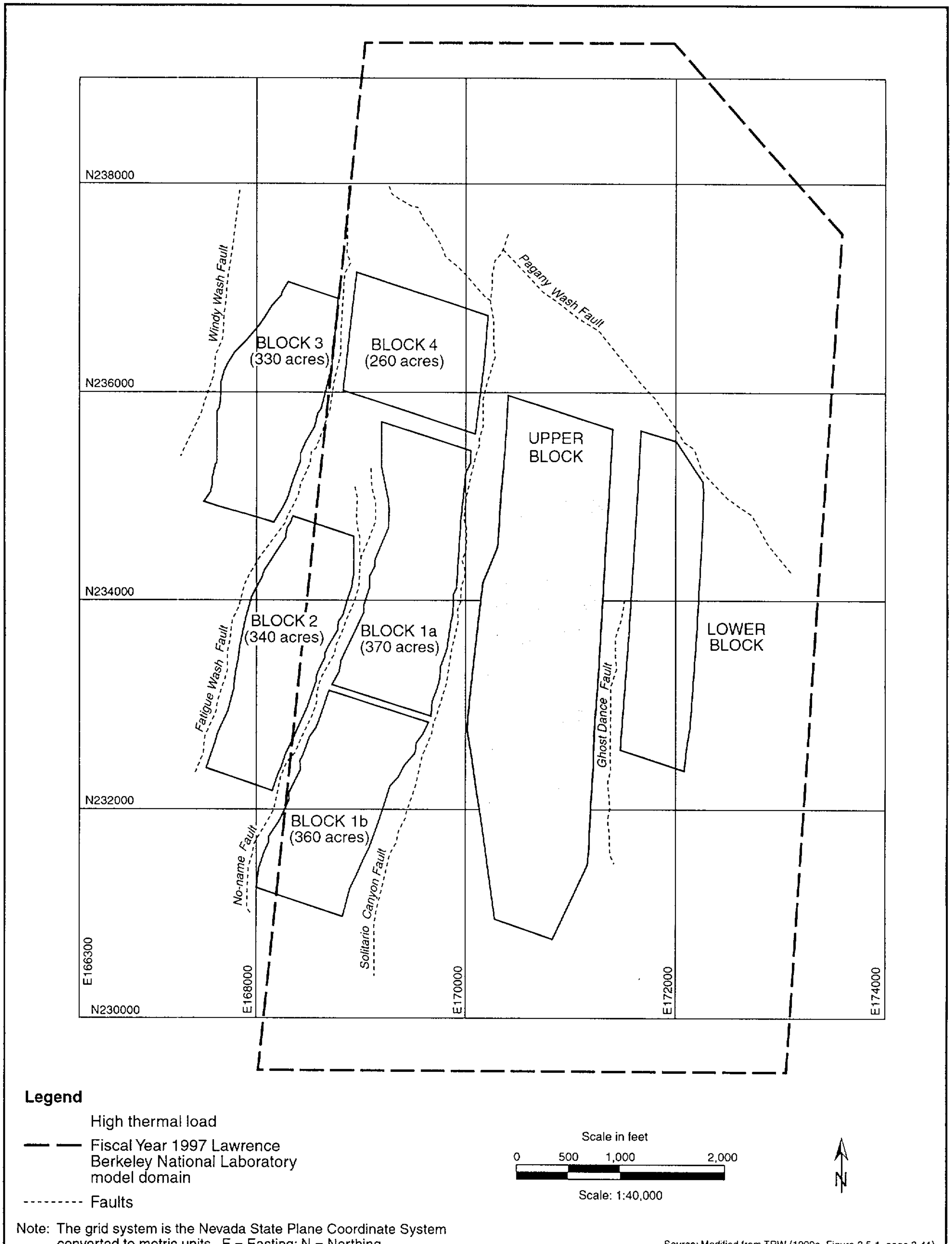


Figure I-27. Regions for performance assessment modeling, Option 1, high thermal load scenario, Proposed Action inventory.

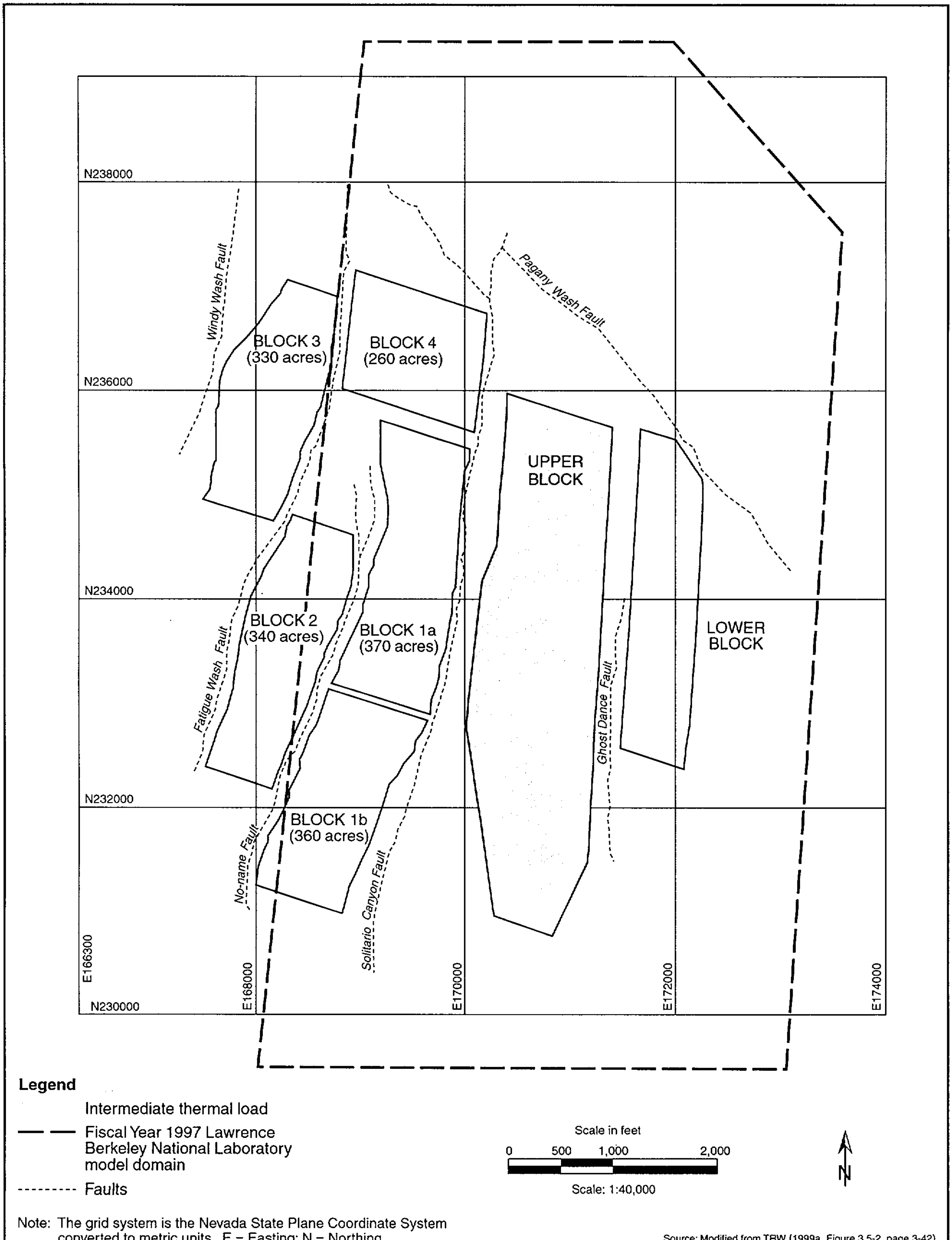


Figure I-28. Regions for performance assessment modeling, Option 2, intermediate thermal load scenario, Proposed Action inventory.

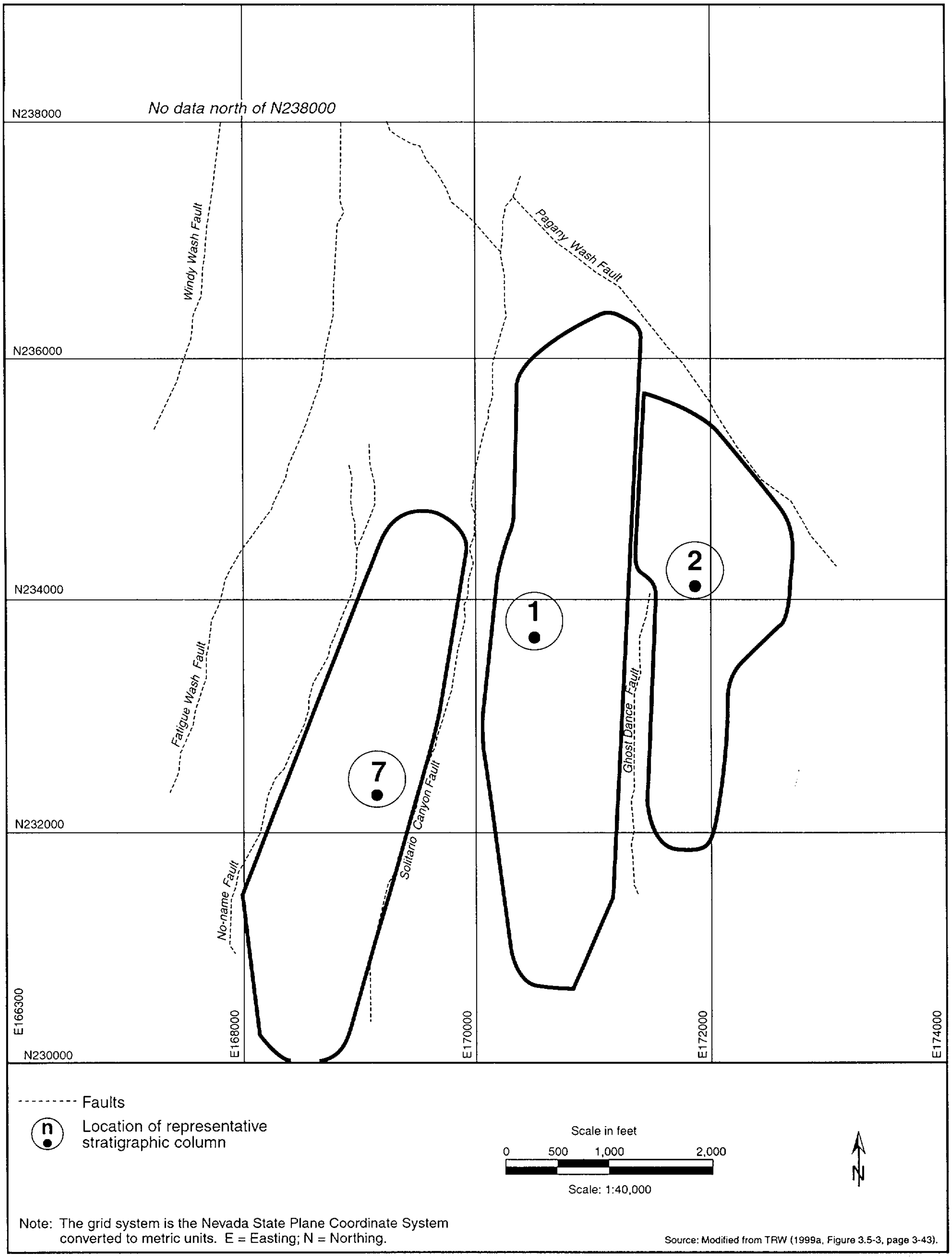


Figure I-29. Repository block areas for performance assessment modeling, Option 3, low thermal load scenario with Inventory Module 1, and intermediate thermal load scenario with Inventory Module 1 cases.

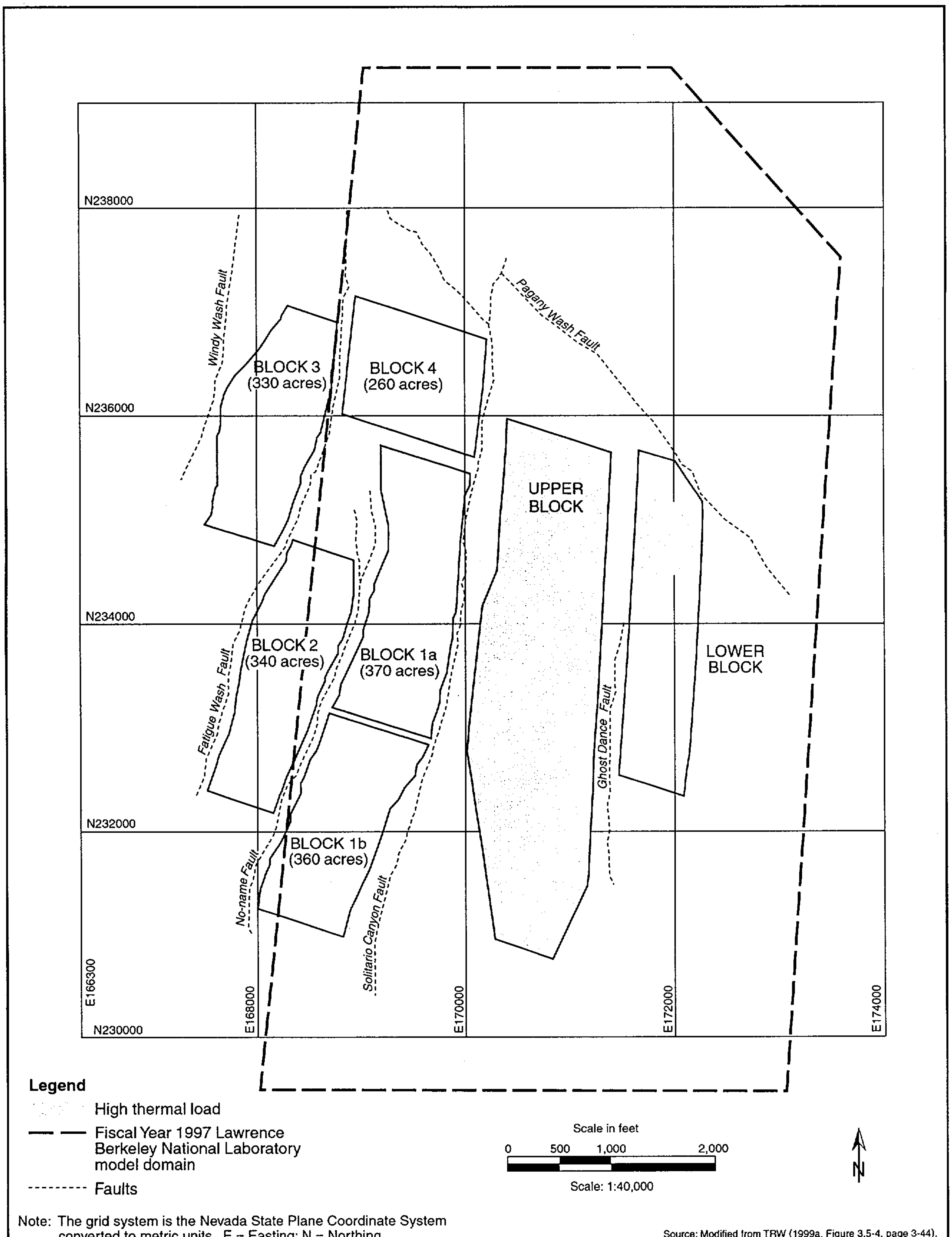


Figure I-30. Regions for performance assessment modeling, Option 4, high thermal load scenario, Proposed Action inventory.

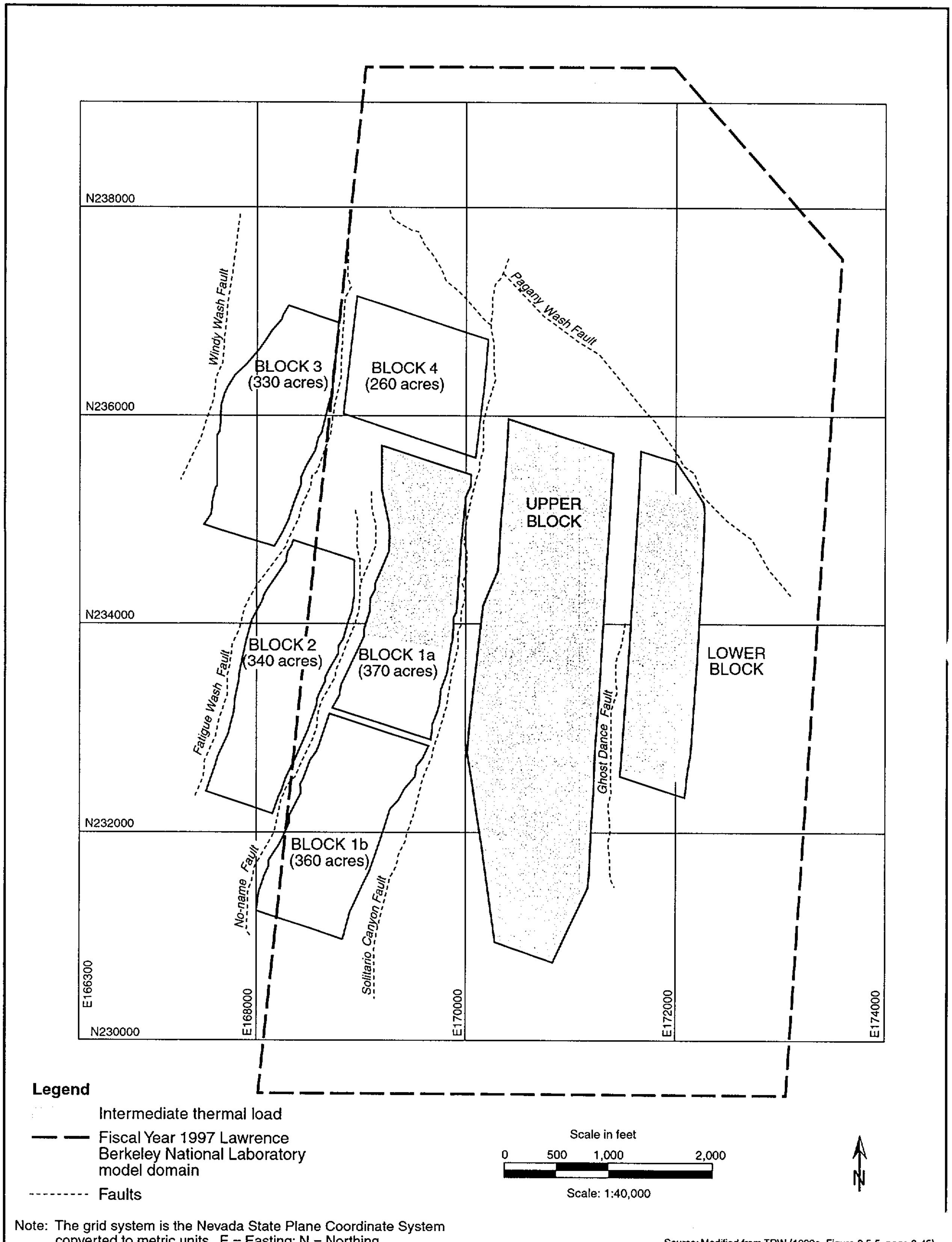


Figure I-31. Regions for performance assessment modeling, Option 5, intermediate thermal load scenario, Inventory Module 1.

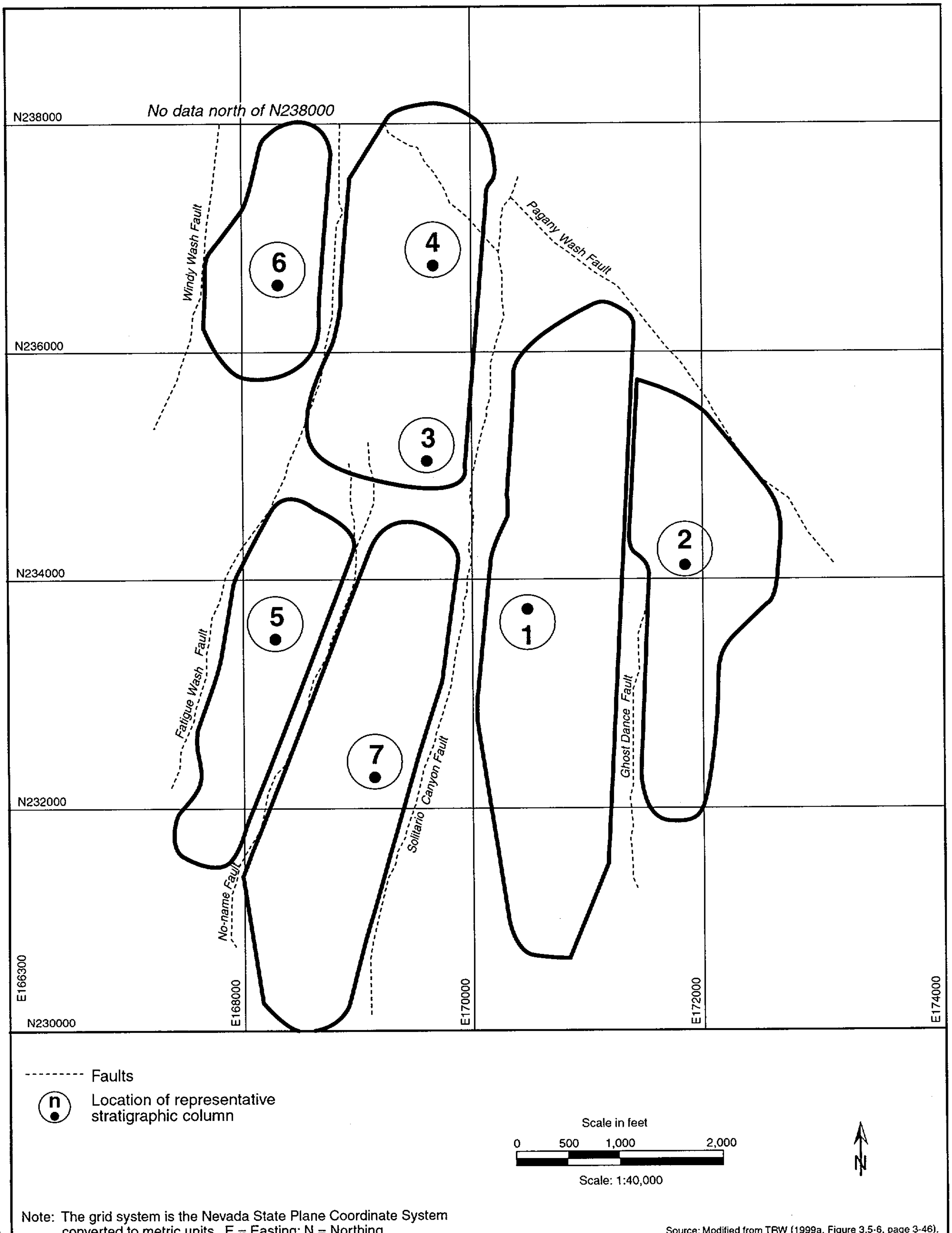


Figure I-32. Repository block areas for performance assessment modeling, Option 6, low thermal load scenario, Inventory Module 1.

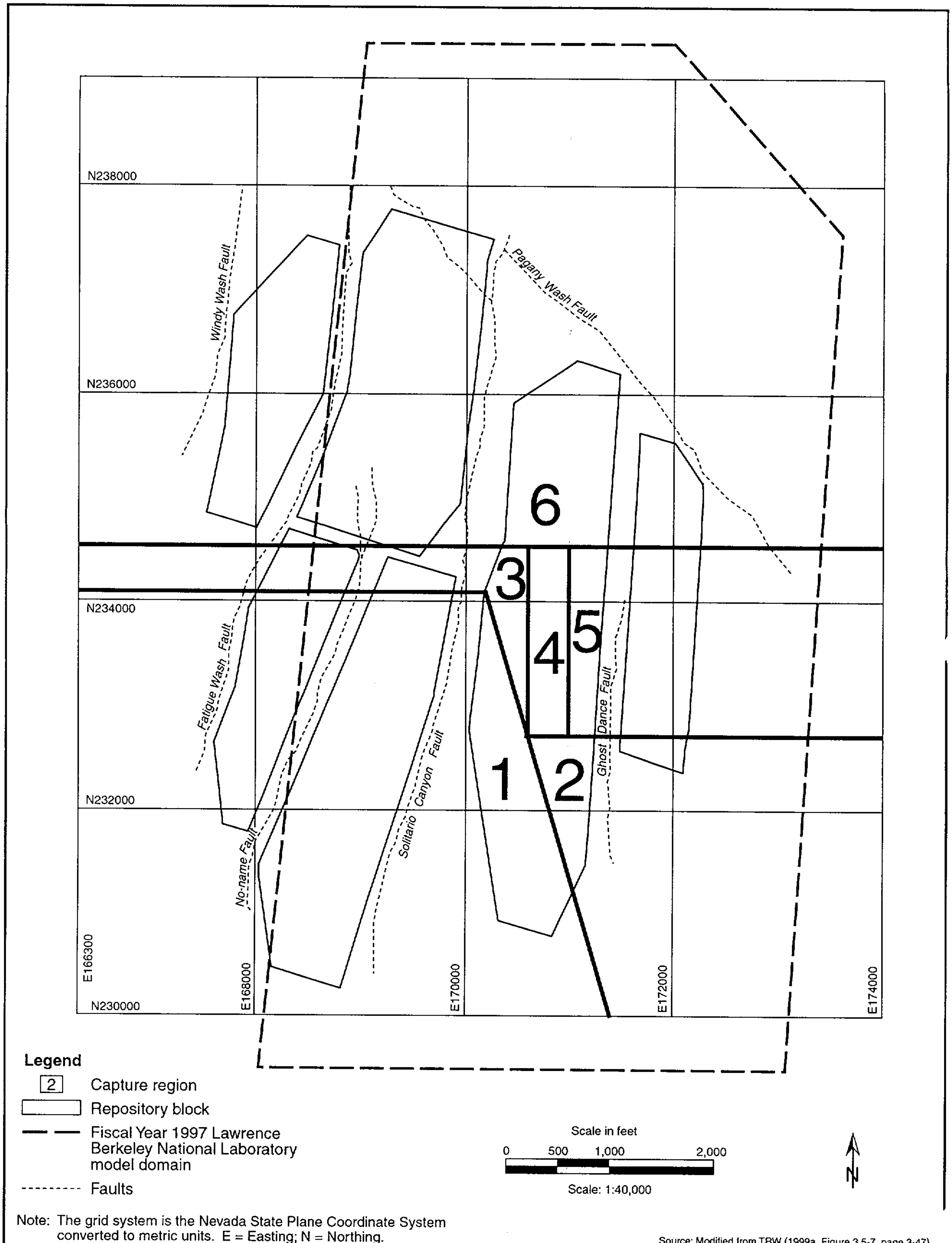


Figure I-33. Capture regions for high and intermediate thermal load scenarios with Proposed Action inventory.

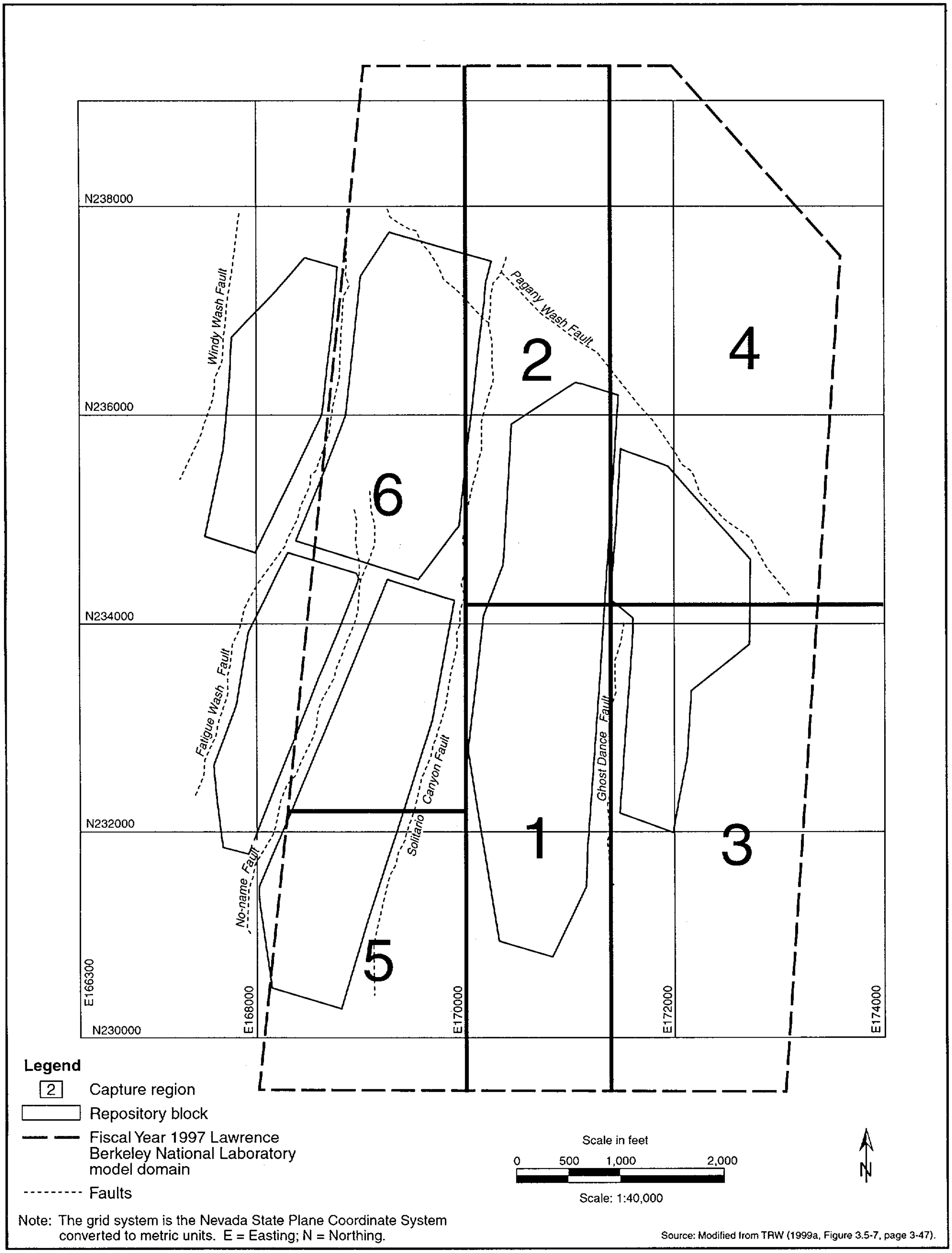


Figure I-34. Capture regions for low thermal load scenario with Proposed Action Inventory and low and intermediate thermal load scenarios with Inventory Modules 1 and 2.

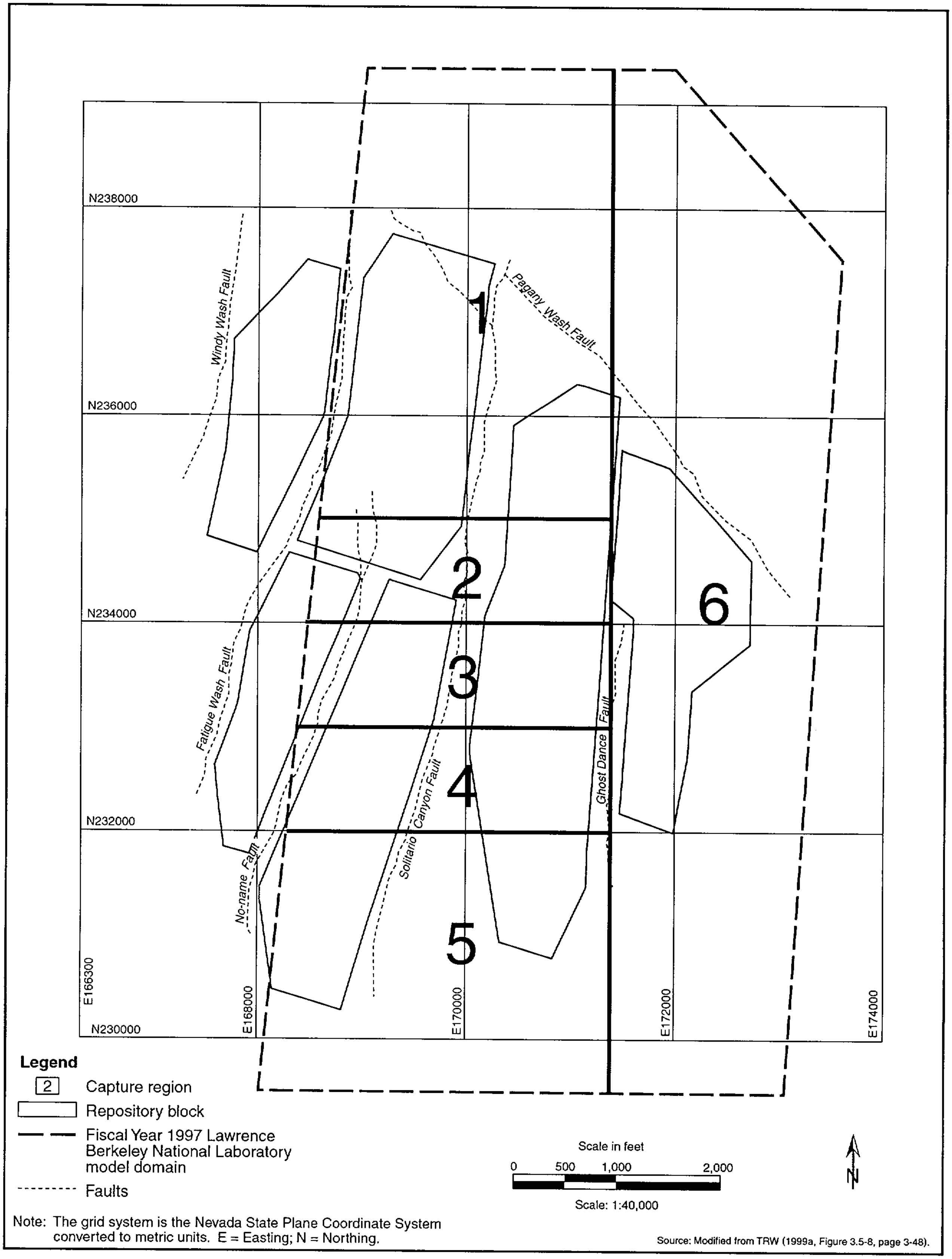
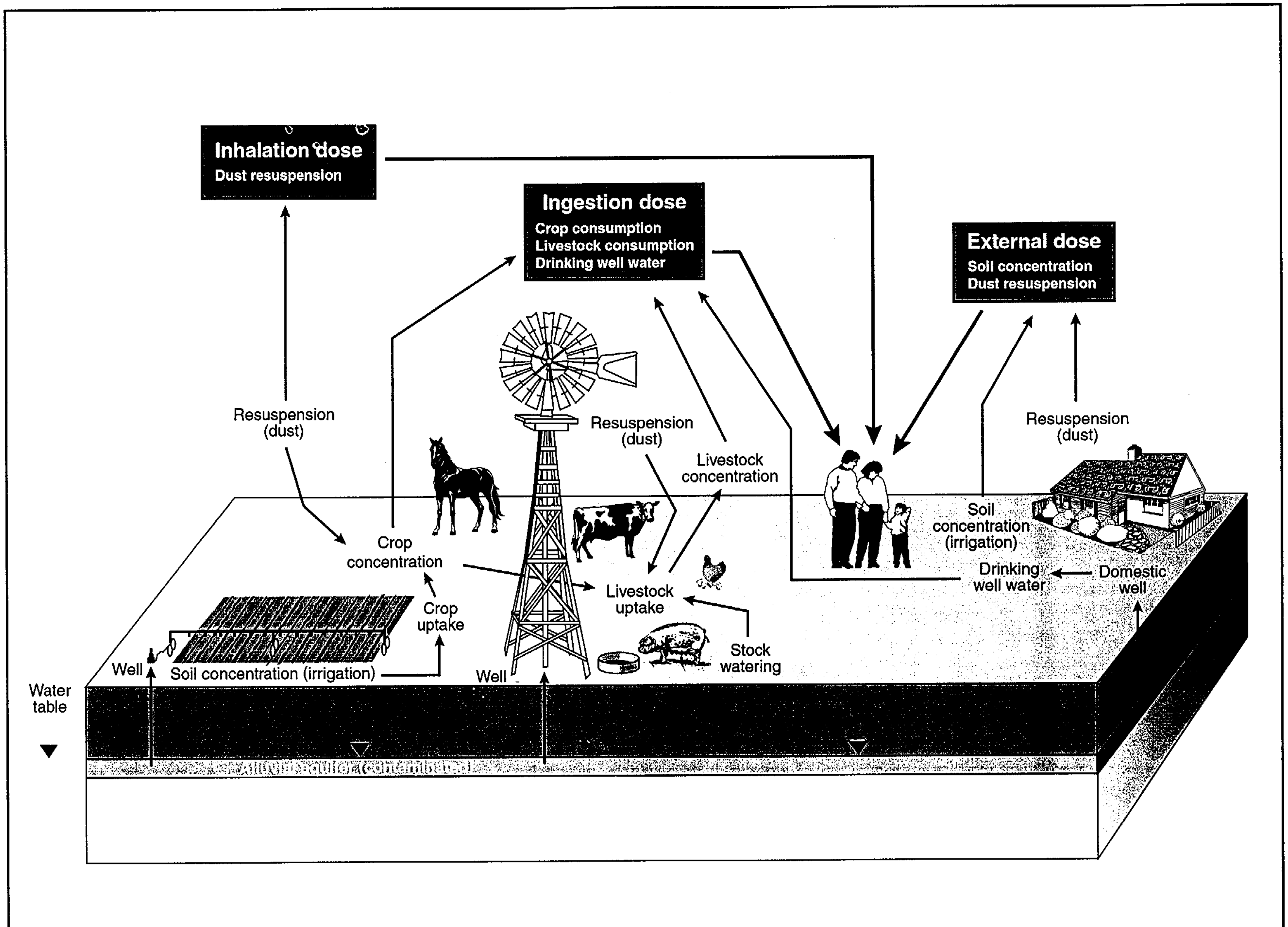
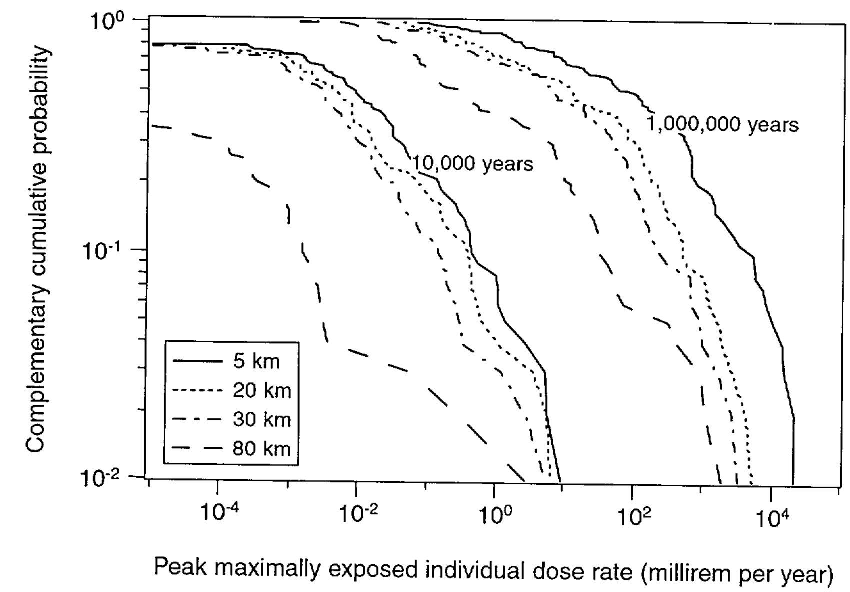


Figure I-35. Capture regions for high thermal load scenario with Inventory Modules 1 and 2.



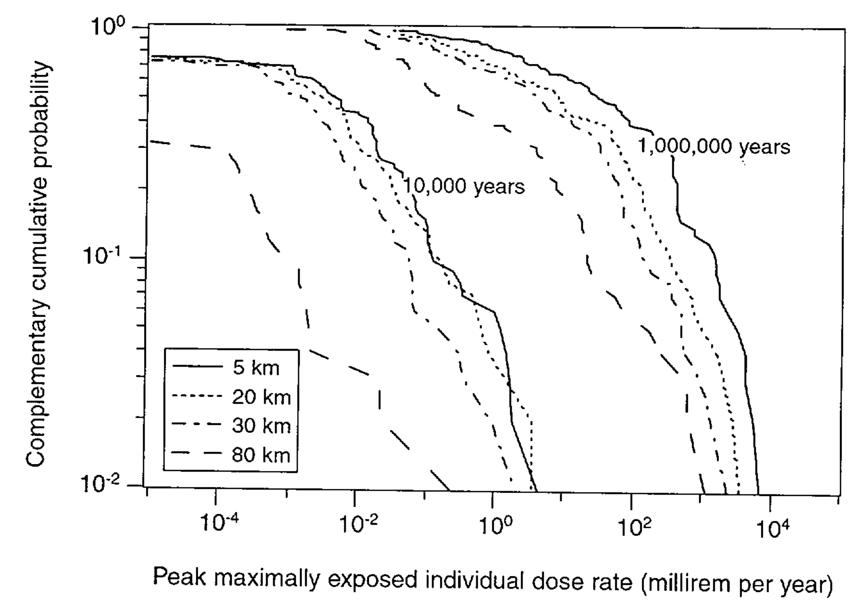
Source: Modified from TRW (1998i, Figure 9-2, page F9-2).

Figure I-36. Biosphere modeling components, including ingestion of contaminated food and water, inhalation of contaminated air, and exposure to direct external radiation.



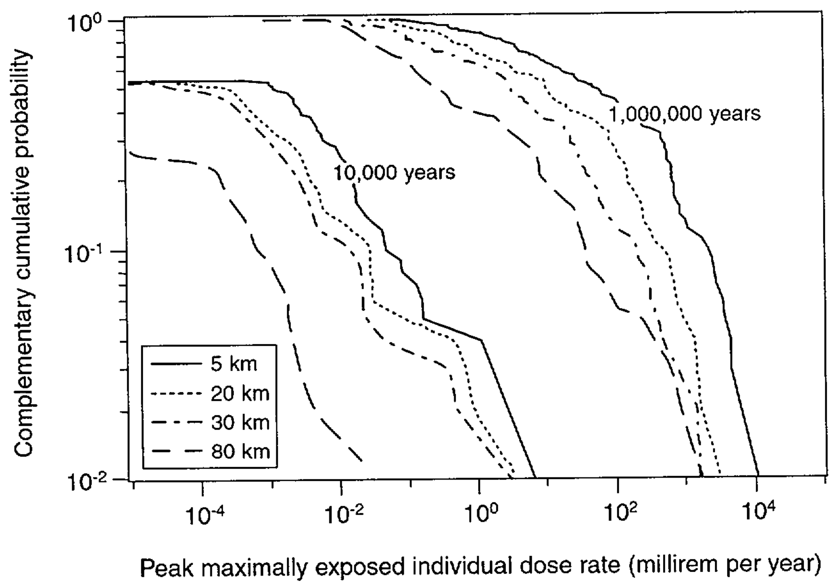
km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-37. Complementary cumulative distribution function of peak maximally exposed individual radiological dose rates during 10,000 and 1 million years following closure for high thermal load scenario with Proposed Action inventory (100 realizations, all pathways, all distances).



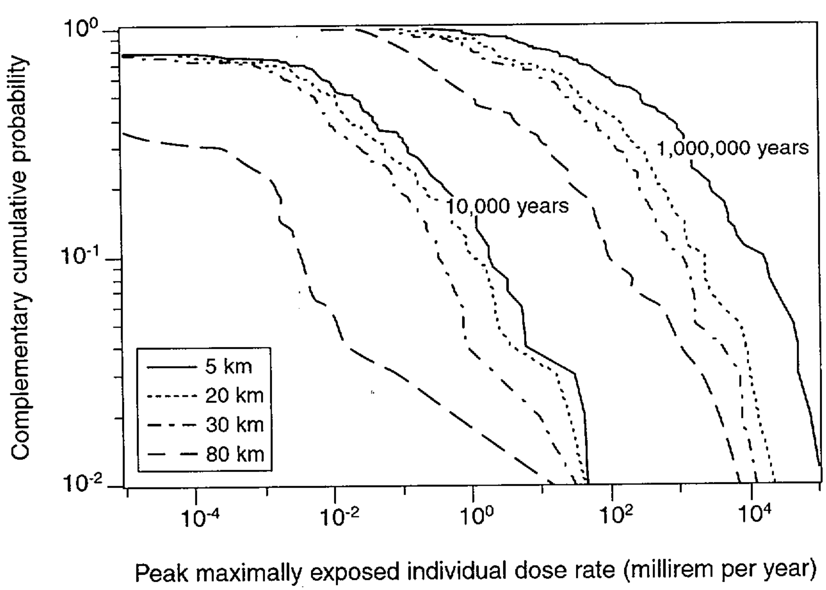
km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-38. Complementary cumulative distribution function of peak maximally exposed individual radiological dose rates during 10,000 and 1 million years following closure for intermediate thermal load scenario with Proposed Action inventory (100 realizations, all pathways, all distances).



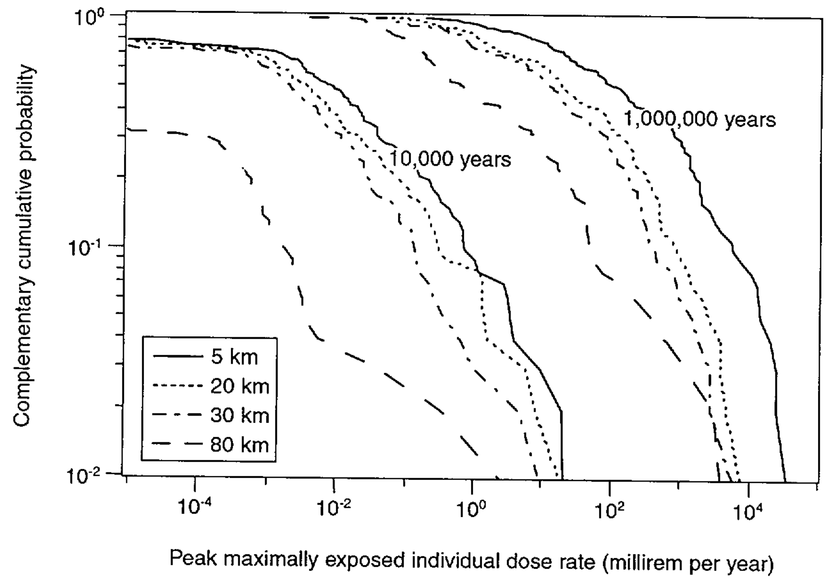
km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-39. Complementary cumulative distribution function of peak maximally exposed individual radiological dose rates during 10,000 and 1 million years following closure for low thermal load scenario with Proposed Action inventory (100 realizations, all pathways, all distances).



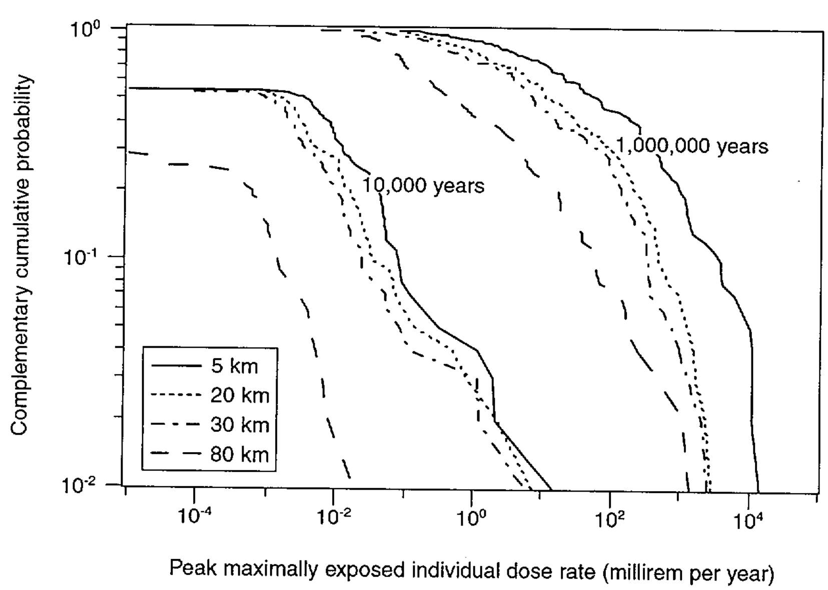
km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-40. Complementary cumulative distribution function of peak maximally exposed individual radiological dose rates during 10,000 and 1 million years following closure for high thermal load scenario with Inventory Module 1 (100 realizations, all pathways, all distances).



km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-41. Complementary cumulative distribution function of peak maximally exposed individual radiological dose rates during 10,000 and 1 million years following closure for intermediate thermal load scenario with Inventory Module 1 (100 realizations, all pathways, all distances).



km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-42. Complementary cumulative distribution function of peak maximally exposed individual radiological dose rates during 10,000 and 1 million years following closure for low thermal load scenario with Inventory Module 1 (100 realizations, all pathways, all distances).

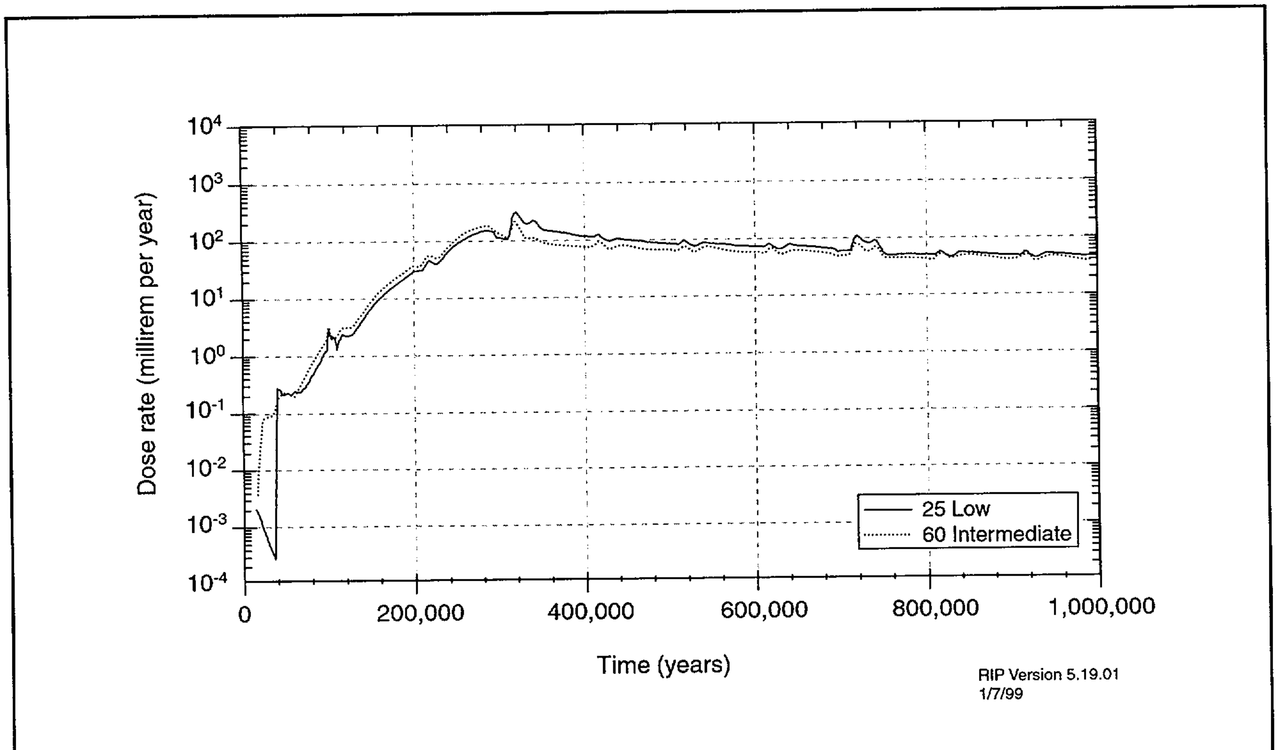


Figure I-43. Comparison of low and intermediate thermal load scenarios total radiological dose histories for the Proposed Action inventory 20 kilometers (12 miles) from the repository.

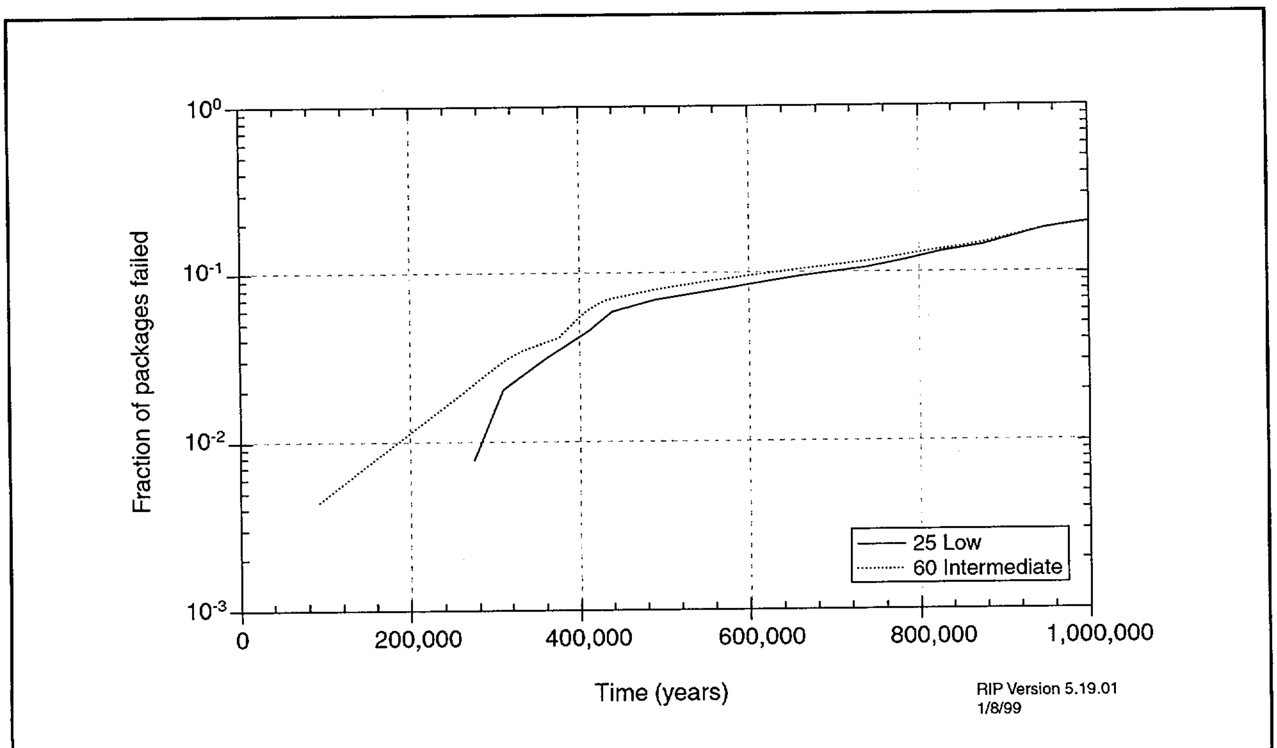


Figure I-44. Waste package failure curves for low and intermediate thermal load scenarios.

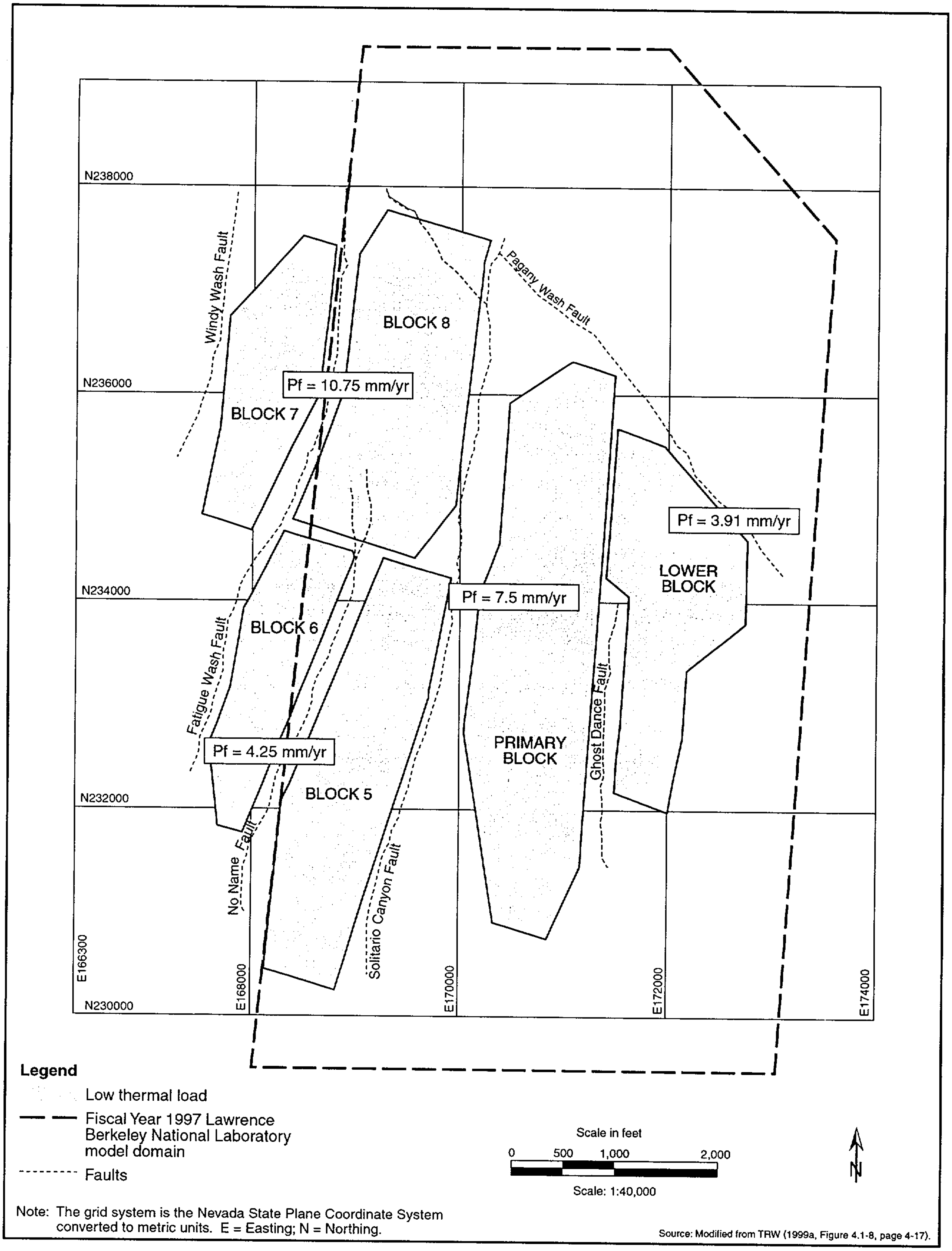


Figure I-45. Average percolation flux for repository blocks.

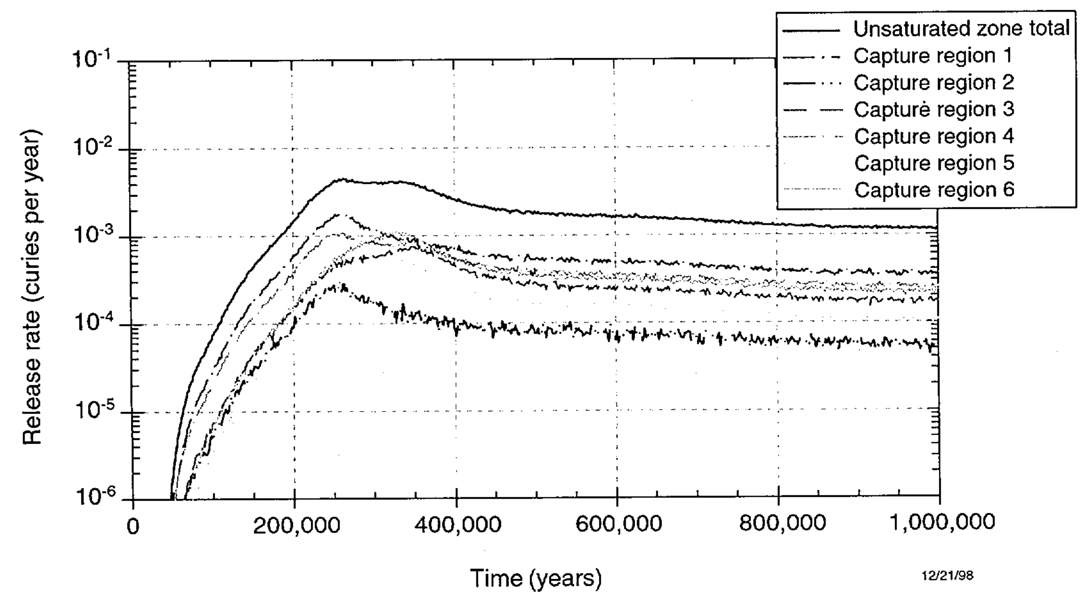


Figure I-46. Neptunium-237 release rate at the water table for fixed long-term average climate for low thermal load scenario during the first 1 million years following repository closure.

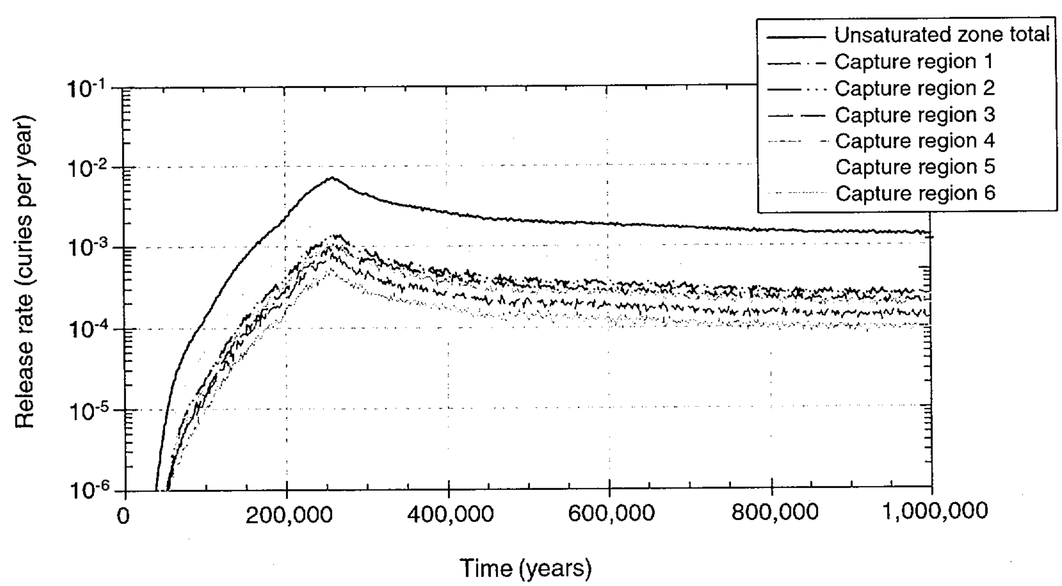


Figure I-47. Neptunium-237 release rate at the water table for fixed long-term average climate for intermediate thermal load scenario during the first 1 million years following repository closure.

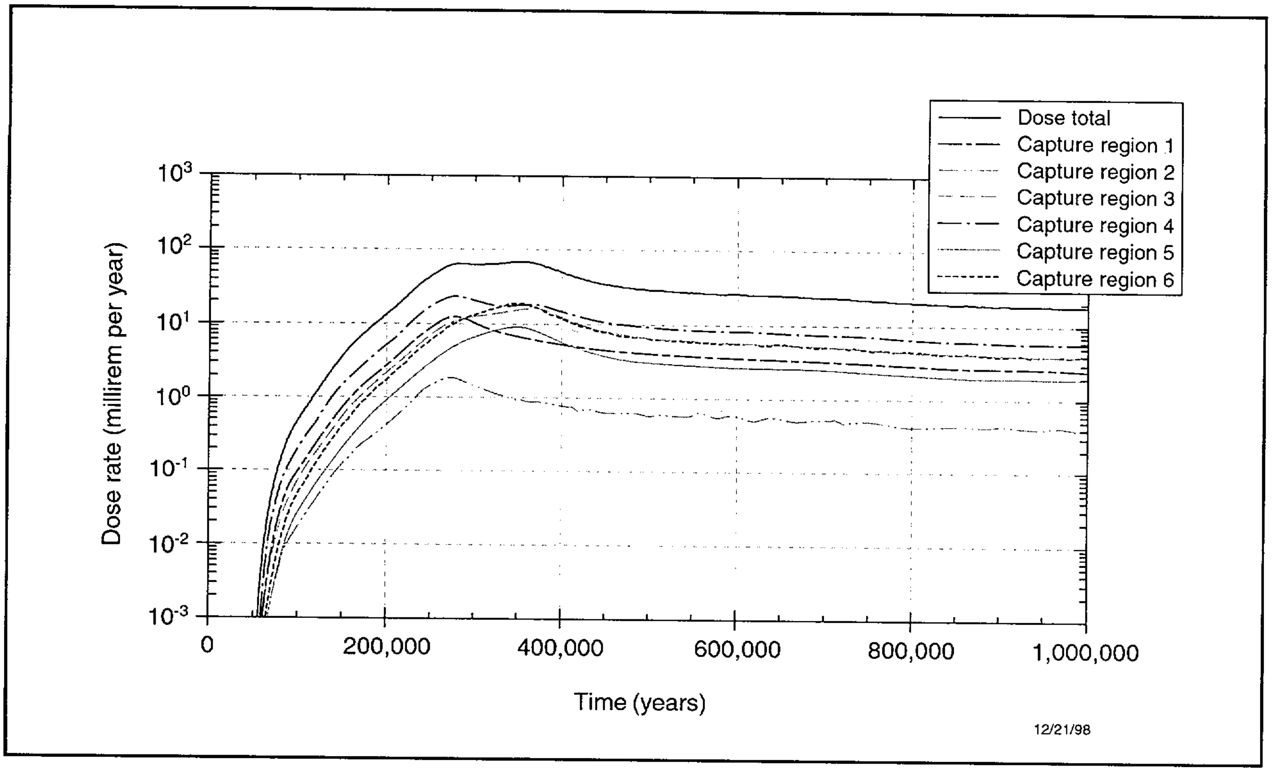


Figure I-48. Neptunium-237 release rate at the end of the saturated zone for fixed long-term average climate for low thermal load scenario during the first 1 million years following repository closure.

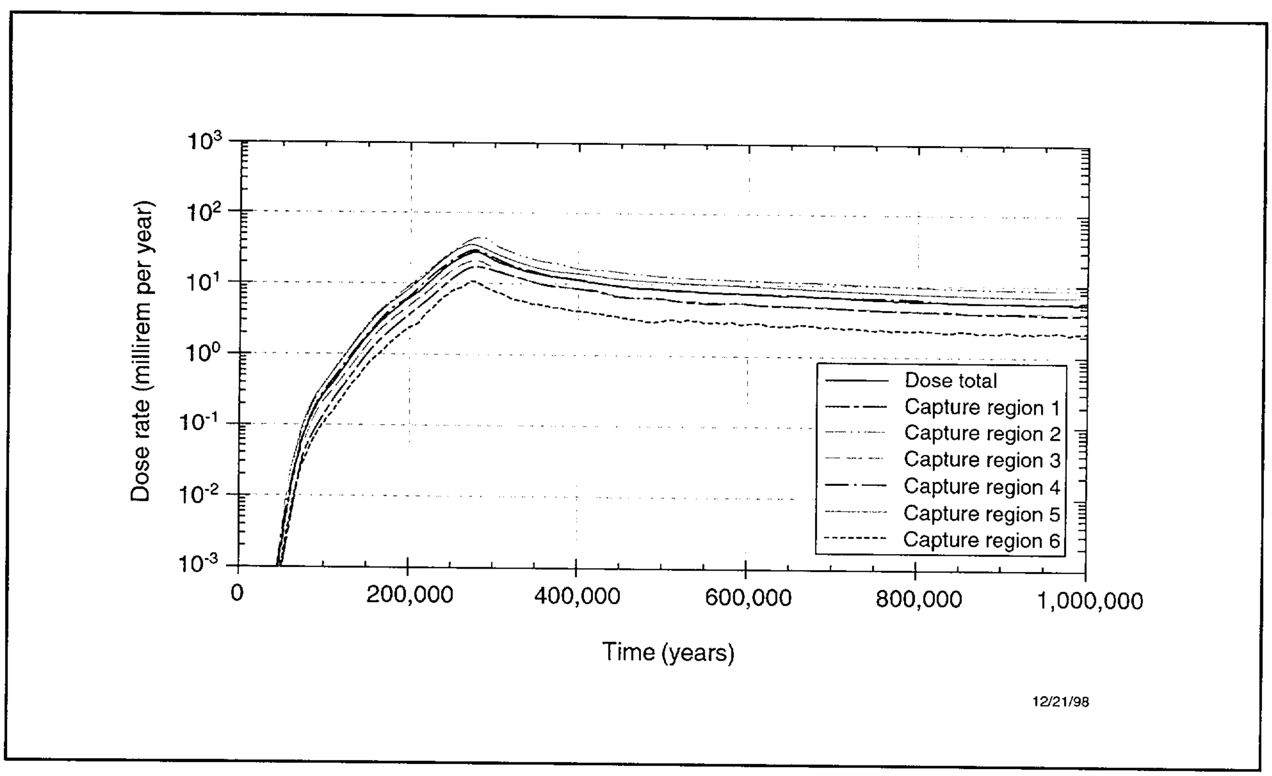


Figure I-49. Neptunium-237 release rate at the end of the saturated zone for fixed long-term average climate for intermediate thermal load scenario during the first 1 million years following repository closure.

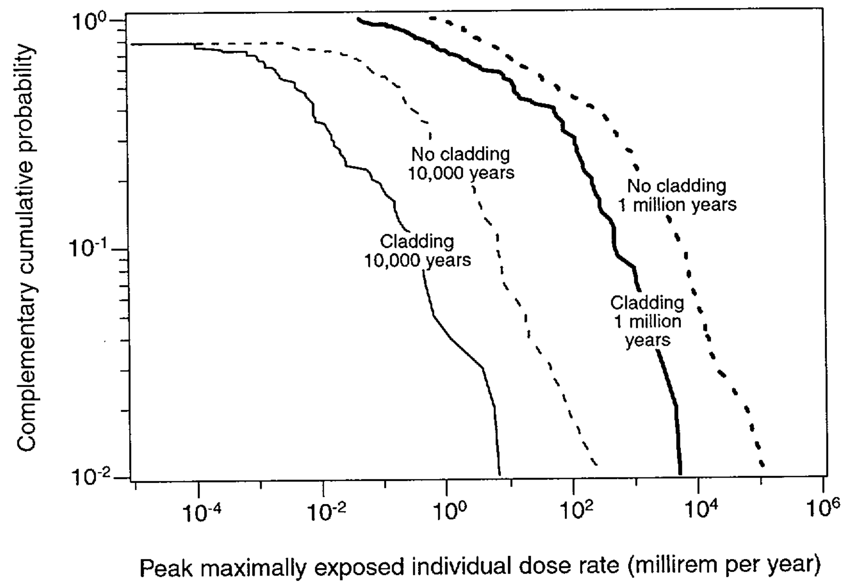


Figure I-50. Complementary cumulative distribution function of radiological doses with and without cladding for a maximally exposed individual at 20 kilometers (12 miles) under the Proposed Action 10,000 and 1 million years after repository closure.

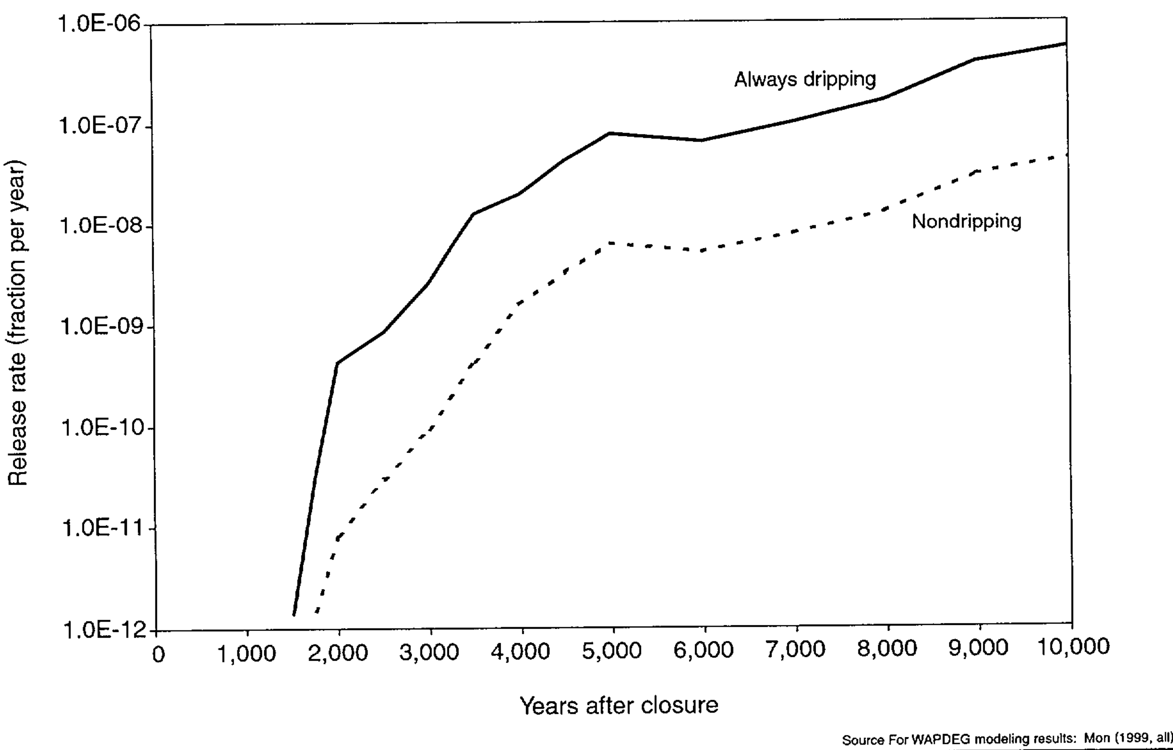
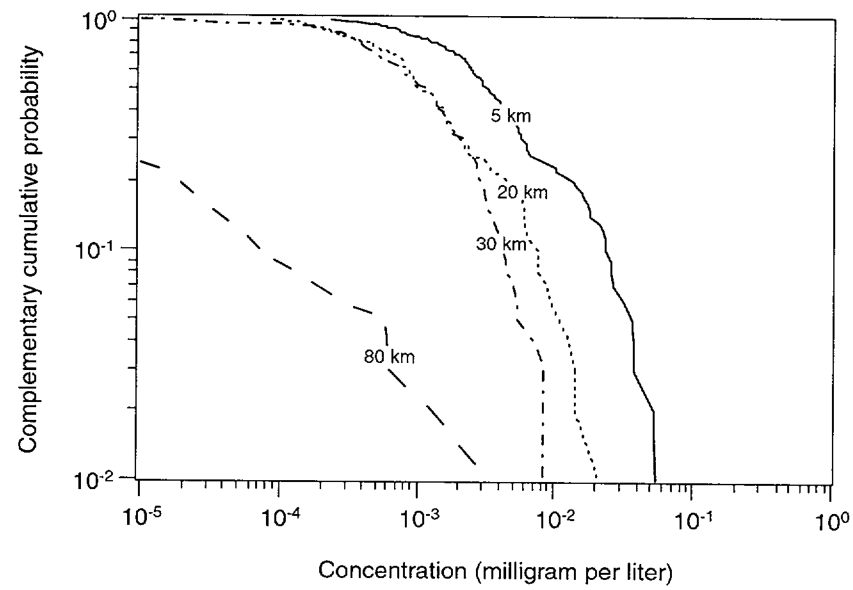
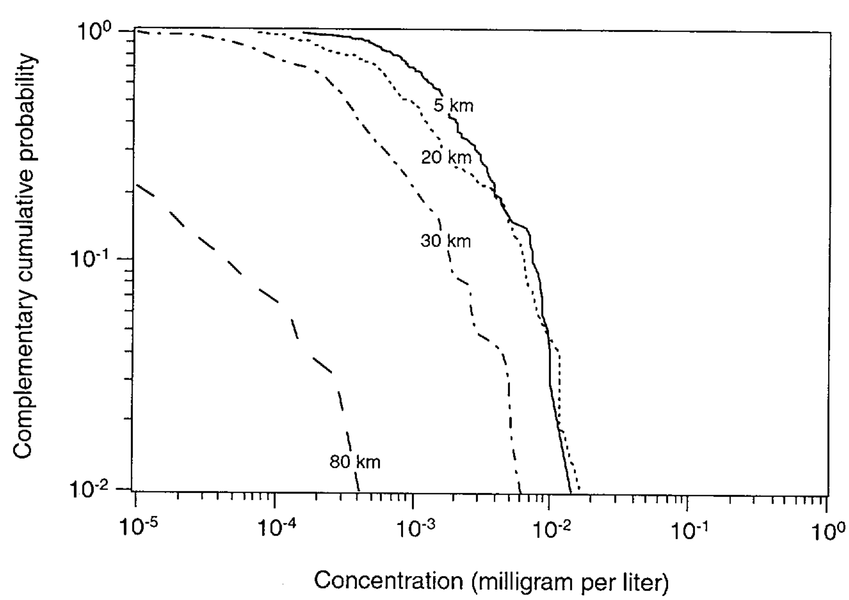


Figure I-51. Average fractional release rate of corrosion-resistant material (Alloy-22) for continually dripping and nondripping conditions computed from WAPDEG modeling results for 400 simulated waste packages.



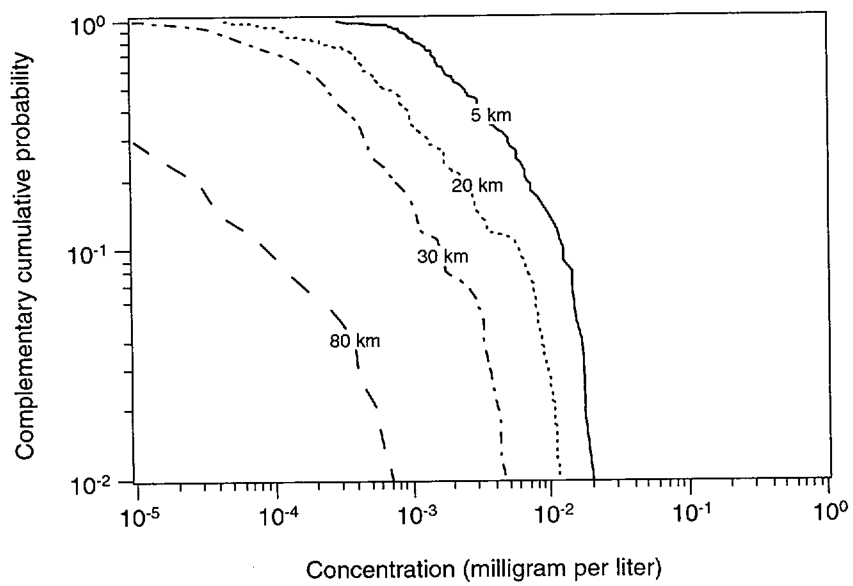
km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-52. Complementary cumulative distribution function of mean peak groundwater concentrations of chromium during 10,000 years following closure under high thermal load scenario with Proposed Action inventory.



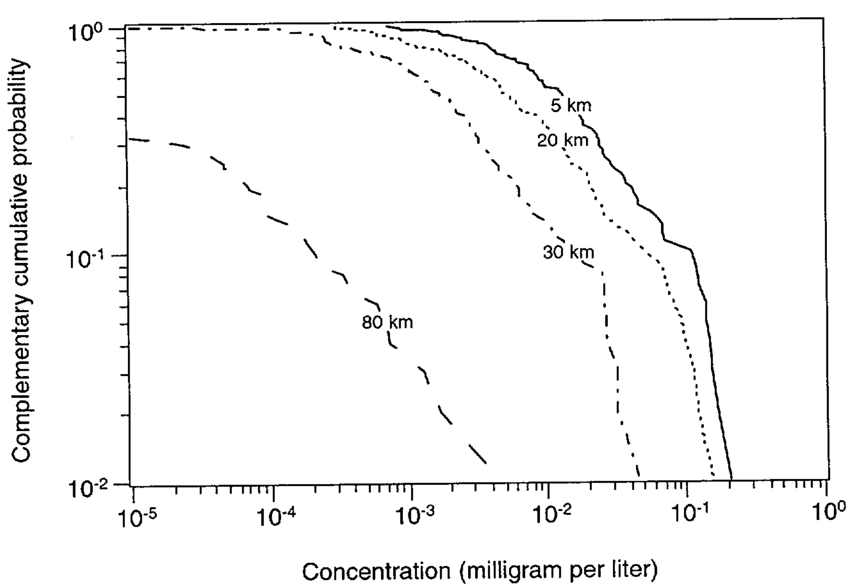
km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-53. Complementary cumulative distribution function of mean peak groundwater concentrations of chromium during 10,000 years following closure under intermediate thermal load scenario with Proposed Action inventory.



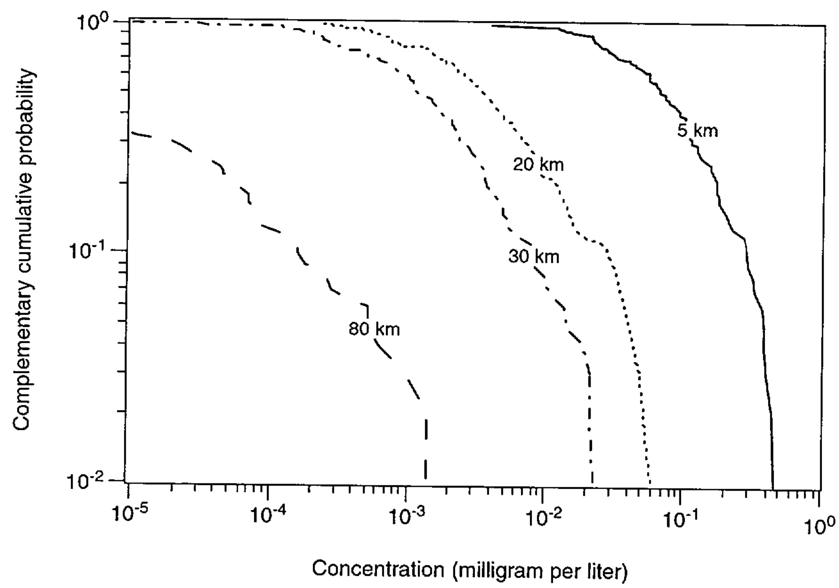
km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-54. Complementary cumulative distribution function of mean peak groundwater concentration of chromium during 10,000 years following closure under low thermal load scenario with Proposed Action inventory.



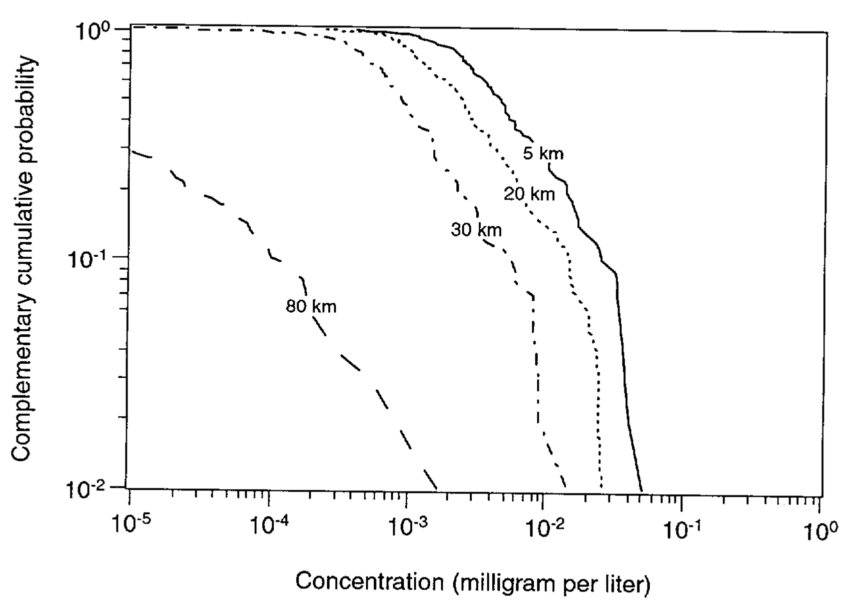
km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-55. Complementary cumulative distribution function of mean peak groundwater concentration of chromium during 10,000 years following closure under high thermal load scenario with Inventory Module 1.



km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-56. Complementary cumulative distribution function of mean peak groundwater concentration of chromium during 10,000 years following closure under intermediate thermal load scenario with Inventory Module 1.



km = kilometer. To convert kilometers to miles, multiply by 0.62137.

Figure I-57. Complementary cumulative distribution function of mean peak groundwater concentration of chromium during 10,000 years following closure under low thermal load scenario with Inventory Module 1.

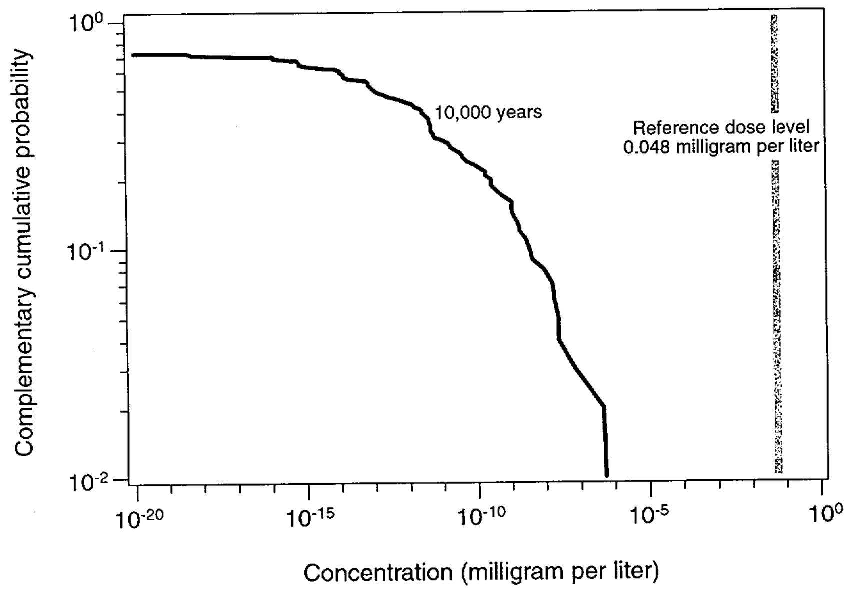


Figure I-58. Complementary cumulative distribution function of mean peak groundwater concentration of elemental uranium in water at 5 kilometers (3 miles) during 10,000 years following closure under high thermal load scenario with Proposed Action inventory.

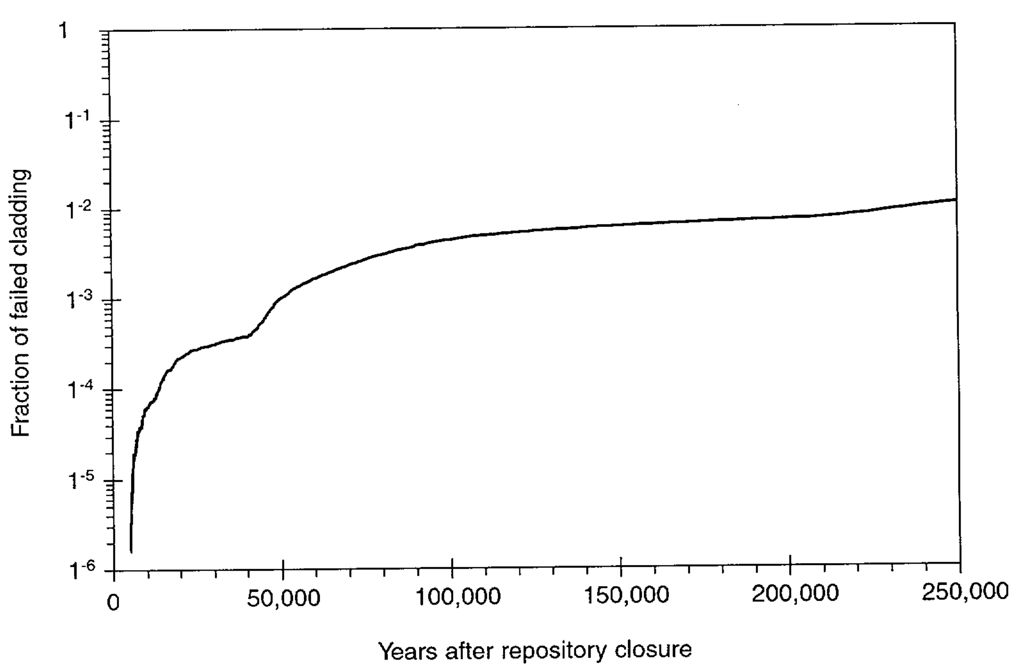


Figure I-59. Fraction (patch area) of cladding that would fail using a zirconium-alloy corrosion rate equal to 1.0 percent of that of Alloy-22.

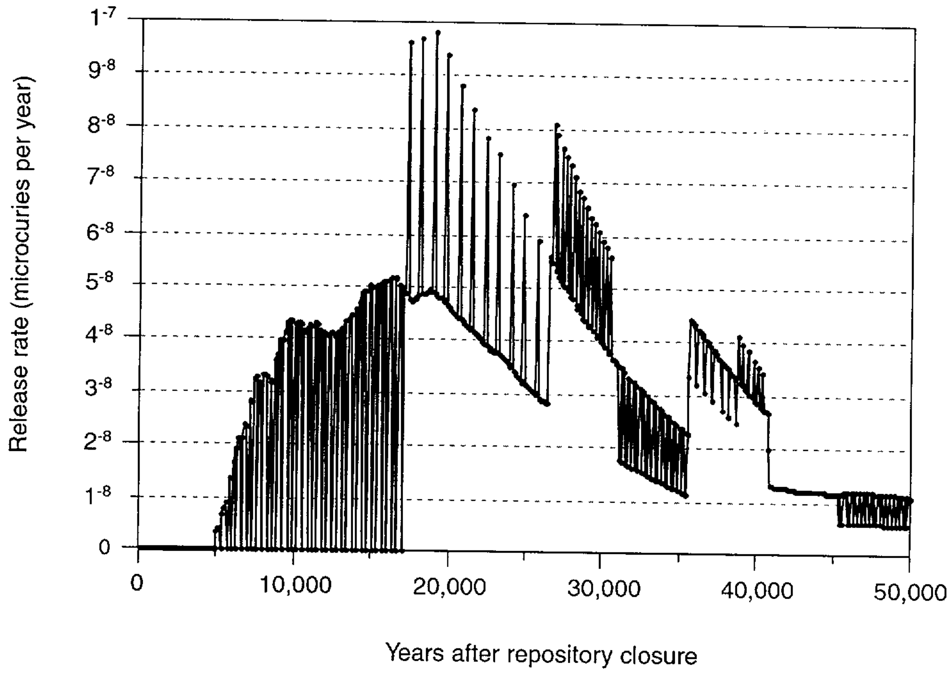


Figure I-60. Release rate of carbon-14 from the repository to the ground surface.

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Appendix J

Transportation

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APPENDIX J. TRANSPORTATION

This appendix provides additional information for readers who wish to gain a better understanding of the methods and analyses the U.S. Department of Energy (DOE) used to determine the human health impacts of transportation for the Proposed Action and Inventory Modules 1 and 2 discussed in this environmental impact statement (EIS). The materials included in Module 1 are the 70,000 metric tons of heavy metal (MTHM) for the Proposed Action and additional quantities of spent nuclear fuel and high-level radioactive waste that DOE could dispose of in the repository as part of a reasonably foreseeable future action. The materials included in Module 2 include the materials in Module 1 and other highly radioactive materials. Appendix A describes materials included in Modules 1 and 2. This appendix also provides the information DOE used to estimate traffic fatalities that would be associated with the long-term maintenance of storage facilities at 72 commercial sites and 5 DOE sites.

The appendix describes the key data and assumptions DOE used in the analyses and the analysis tools and methods the Department used to estimate impacts of loading operations at 72 commercial and 5 DOE sites; incident-free transportation by highway, rail and barge; intermodal transfer; and transportation accidents. The references listed at the end of this appendix contain additional information.

This appendix presents information on analyses of the impacts of national transportation and on analyses of the impacts that could occur in Nevada. Section J.1 presents information on the analysis of occupational and public health and safety impacts for the transportation of spent nuclear fuel and high-level radioactive waste from the 77 sites to the repository. Section J.2 presents information on the analysis of rail and intermodal transportation options. Section J.3 presents information on the analysis of transportation in Nevada. Section J.4 presents a summary assessment of the Nevada transportation implementing alternatives.

J.1 Methods Used To Estimate Potential Impacts of National Transportation

This section provides information on the methods and data DOE used to estimate impacts from shipping spent nuclear fuel and high-level radioactive waste from 72 commercial sites and 5 DOE sites throughout the United States to the Yucca Mountain Repository.

MOSTLY LEGAL-WEIGHT TRUCK AND MOSTLY RAIL SCENARIOS

The Department does not anticipate that either the mostly legal-weight truck or the mostly rail scenario represents the actual mix of truck or rail transportation modes it would use. Nonetheless, DOE used these scenarios as a basis for the analysis of potential impacts to ensure the analysis addressed the range of possible transportation impacts. Thus, the estimated numbers of shipments for the mostly legal-weight truck and mostly rail scenarios represent only the two extremes in the possible mix of transportation modes. Therefore, the analysis provides estimates that cover the range of potential impacts to human health and safety and to the environment for the transportation modes DOE could use for the Proposed Action.

J.1.1 ANALYSIS APPROACH AND METHODS

Three types of impacts could occur to the public and workers from transportation activities associated with the Proposed Action. These would be a result of the transportation of spent nuclear fuel and high-level radioactive waste and of the personnel, equipment, materials, and supplies needed to construct, operate and monitor, and close the proposed Yucca Mountain Repository. The first type, radiological impacts, would be measured by radiological dose to populations and individuals and the resulting estimated number of latent cancer fatalities that would be caused by radiation from shipments of spent nuclear fuel and high-level radioactive waste from the 77 sites under normal and accident transport conditions. The second and third types would be nonradiological impacts—fatalities caused by vehicle emissions and fatalities caused by vehicle accidents. The analysis also estimated impacts due to the characteristics of hazardous cargoes from accidents during the transportation of nonradioactive hazardous materials to support repository construction, operation and monitoring, and closure. For perspective, about 10 fatalities resulting from hazardous material occur each year during the transportation of more than 300 million shipments of hazardous materials in the United States (DOT 1998a, Table 1). Therefore, DOE expects that the risks from exposure to hazardous materials that could be released during shipments to and from the repository sites would be very small (see Section J.1.4.2.4). The analysis evaluated the impacts of traffic accidents and vehicle emissions arising from these shipments.

The analysis used a step-wise process to estimate impacts to the public and workers. The process used the best available information from various sources and computer programs and associated data to accomplish the steps. Figures J-1 and J-2 show the steps followed in using data and computer programs. DOE has determined that the computer programs identified in the figure are suitable, and provide results in the appropriate measures, for the analysis of impacts performed for this EIS.

The CALVIN computer program (TRW 1998, all) is used to estimate the numbers of shipments of spent nuclear fuel from commercial sites. This program uses information on spent nuclear fuel stored at each site and an assumed scenario for picking up the spent fuel from each site. The program also uses information on the capacity of shipping casks that could be used.

The HIGHWAY computer program (Johnson et al. 1993a, all) is a routing tool used to select existing highway routes that would satisfy Department of Transportation route selection regulations and that DOE could use to ship spent nuclear fuel and high-level radioactive waste from the 77 sites to the repository.

The INTERLINE computer program (Johnson et al. 1993b, all) is a routing tool used to select existing rail routes that railroads would be likely to use to ship spent nuclear fuel and high-level radioactive waste from the 77 sites to the repository.

The RADTRAN4 computer program (Neuhauser and Kanipe 1992, all) is used to estimate the radiological dose risks to populations and transportation workers of incident-free transportation and to the general population from accident scenarios. For the analysis of incident-free transportation risks, the code uses scenarios for persons who would share transportation routes with shipments—called *onlink populations*, persons who live along the route of travel—*offlink populations,* and persons exposed at stops. For accident risks, the code evaluates the range of possible accident scenarios from high probability and low consequence to low probability and high consequence.

The RISKIND computer program (Yuan et al. 1995, all) is used to estimate radiological doses to maximally exposed individuals for incident-free transportation and to populations and maximally exposed individuals for accident scenarios. To estimate incident-free doses to maximally exposed individuals, RISKIND uses geometry to calculate the dose rate at specified locations that would arise from a source of radiation. RISKIND is also used to calculate the radiation dose to a population and hypothetical maximally exposed individuals from releases of radioactive materials that are postulated to occur in maximum reasonably foreseeable accident scenarios.

The following sections describe these programs in detail.

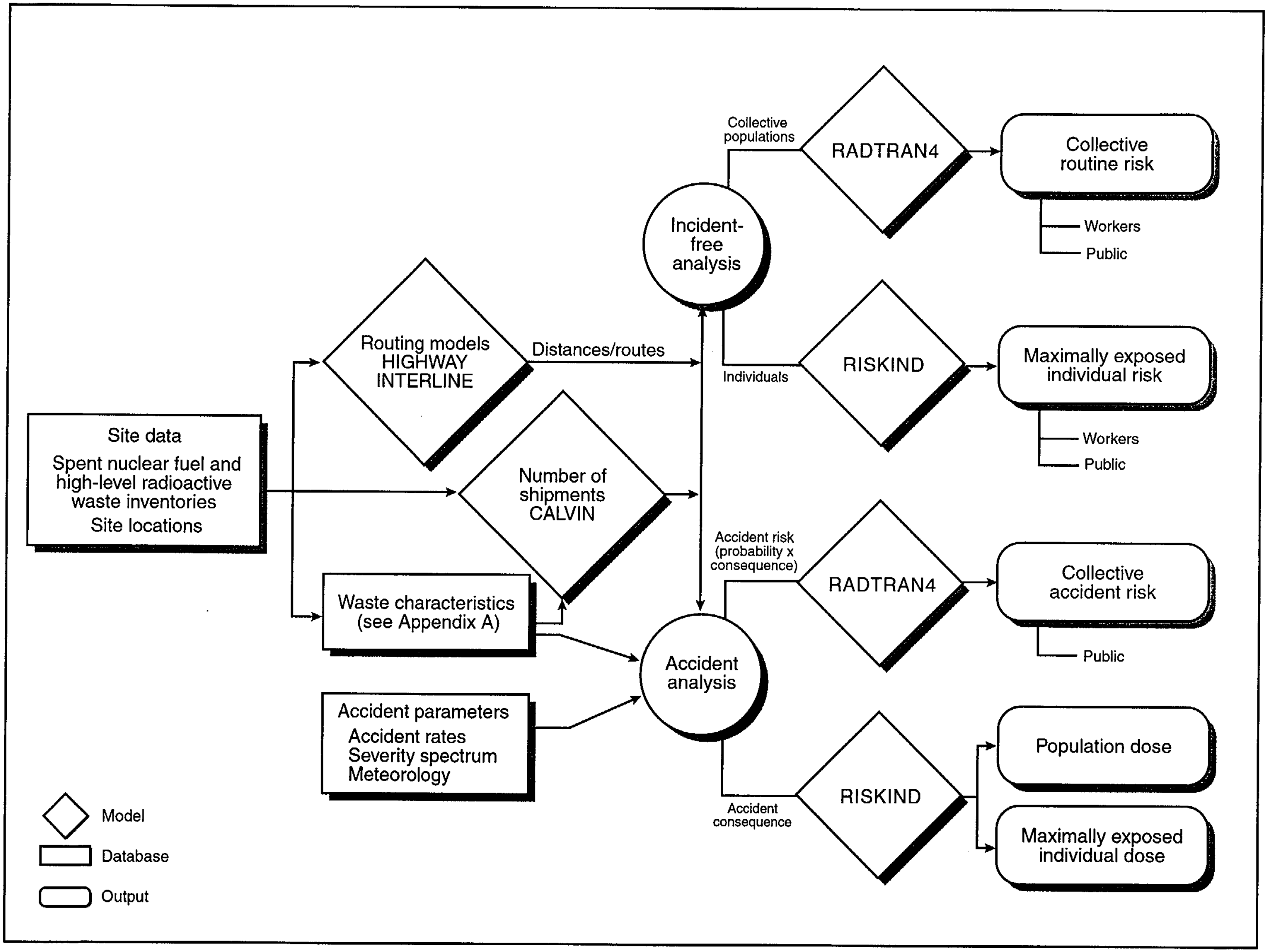
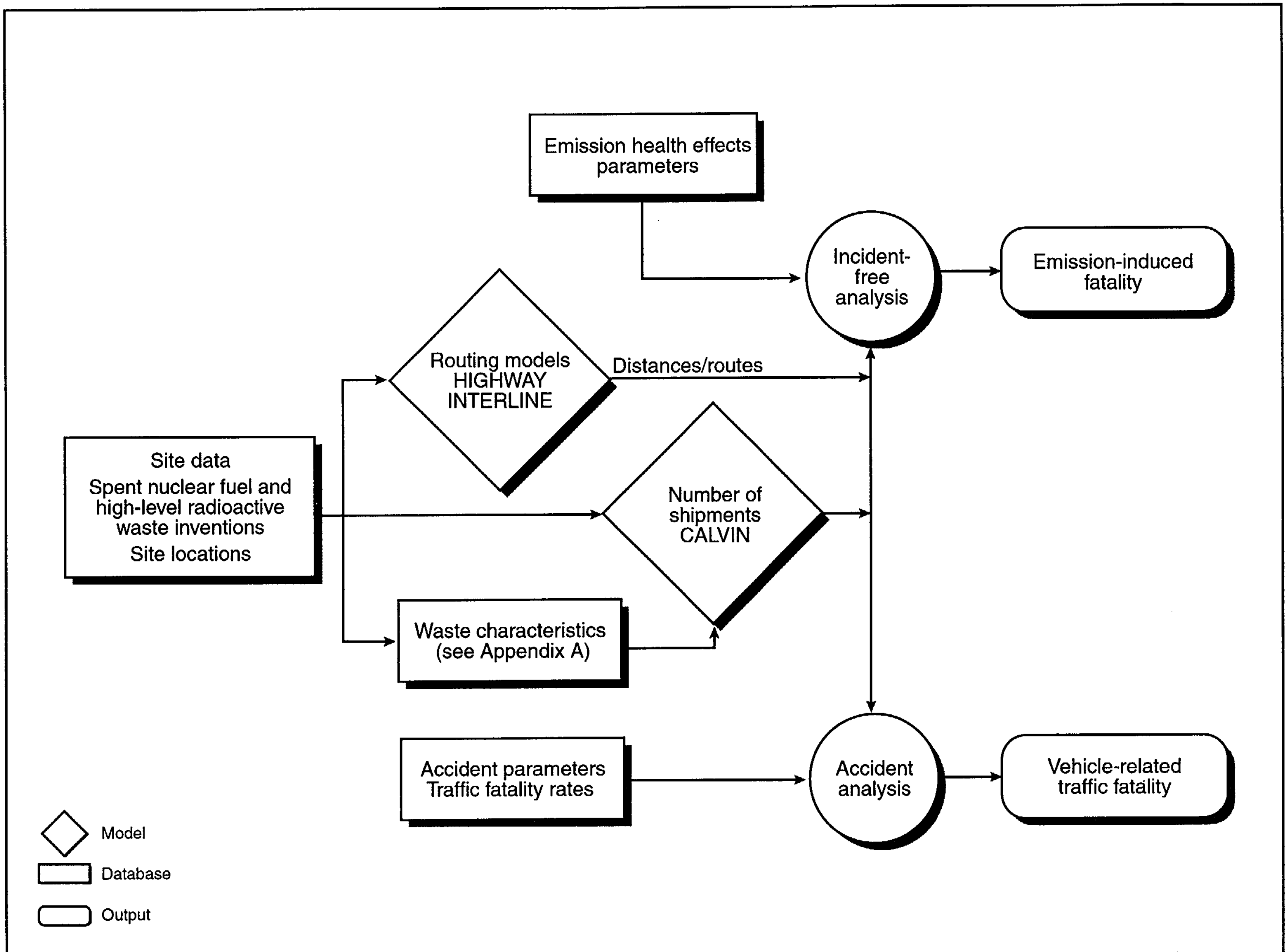


Figure J-1. Methods and approach for analyzing transportation radiological health risk.



F[illegible] J-2. Methods and approach for analyzing transportation no[illegible]logical health risk.

DOSE RISK

Dose risk is a measure of radiological impacts to populations – public or workers – from the potential for exposure to radioactive materials. Thus, a potential of 1 chance in 1,000 of a population receiving a collective dose of 1 rem (1 person-rem) from an accident would result in a dose risk of 0.001 person-rem (0.001 is the product of 1 person-rem and the quotient of 1 over 1,000). Dose risk is often expressed in units of latent cancer fatalities.

The use of dose risk to measure radiological impacts allows a comparison of alternatives with differing characteristics in terms of radiological consequences that could result and the likelihood that the consequences would actually occur.

J.1.1.1 CALVIN

The Civilian Radioactive Waste Management System Analysis and Logistics Visually Interactive (CALVIN) model (TRW 1998, all) was developed to be a planning tool to estimate the logistic and cost impacts of various operational assumptions for accepting radioactive wastes. CALVIN is used in transportation modeling to determine the number of shipments of commercial spent nuclear fuel from each reactor site. The parameters that the CALVIN model used to determine commercial spent nuclear fuel movement include the shipping cask specifications including heat limits, $k_{infinity}$ (measure of criticality) limits for the contents of the casks, capacity (assemblies or canisters/cask), burnup/enrichment curves, and cooling time for the fuel being shipped.

The source data used by CALVIN for commercial spent nuclear fuel projections include the RW-859 historic data collected by the Energy Information Administration, and the corresponding projection produced based on current industry trends for commercial fuel (see Appendix A). This EIS used CALVIN to estimate commercial spent nuclear fuel shipment numbers based on the cask capacity (see Section J.1.2) and the shipping cask handling capabilities at each site. For the mostly rail national transportation scenario, CALVIN assumed that shipments would use the largest cask a site would be capable of handling. In some cases, CALVIN estimated that the characteristics of the spent nuclear fuel that would be picked up at a site would exceed the capabilities of the largest cask if the cask was fully loaded. In such cases, to provide a realistic estimate of the number of shipments that would be made, the program derated (reduced the capacity of) the casks. The reduction in capacity was sufficient to accommodate the characteristics of the spent nuclear fuel the program estimated for pickup at the site.

J.1.1.2 HIGHWAY

The HIGHWAY computer program (Johnson et al. 1993a, all) was used to select highway routes for the analysis of impacts presented in this EIS. HIGHWAY calculates routes by minimizing the total impedance between the origin and the destination. The impedance is determined by distance and driving time along a particular segment of highway. Using Rand McNally route data and rules that apply to carriers of Highway Route-Controlled Quantities of Radioactive Materials (49 CFR 397.101), HIGHWAY selected highway routes for legal-weight truck shipments from each commercial and DOE site to the Yucca Mountain site. In addition, DOE used this program to estimate the populations within 800 meters (0.5 mile) of the routes it selected. These population densities were used in calculating incident-free radiological risks to the public along the routes.

One of the features of the HIGHWAY model is its ability to estimate routes for the transport of Highway Route-Controlled Quantities of Radioactive Materials. The Department of Transportation has established a set of routing regulations for the transport of these materials (49 CFR 397.101). Routes following these regulations are frequently called HM-164 routes. The regulations require the transportation of these shipments on preferred highways, which include:

- Interstate highways
- An Interstate System bypass or beltway around a city
- State-designated preferred routes

State routing agencies can designate preferred routes as an alternative to, or in addition to, one or more Interstate highways. In making this determination, the state must consider the safety of the alternative preferred route in relation to the Interstate route it is replacing, and must register all such designated preferred routes with the Department of Transportation.

Frequently, the origins and destinations of Highway Route-Controlled Quantities of Radioactive Materials are not near Interstate highways. In general, the Department of Transportation routing regulations require the use of the shortest route between the pickup location to the nearest preferred route entry location and the shortest route to the destination from the nearest preferred route exit location. In general, HM-164 routes tend to be somewhat longer than other routes; however, the increased safety associated with Interstate highway travel is the primary purpose of the routing regulations.

Because many factors can influence the time in transit over a preferred route, a carrier of Highway Route-Controlled Quantities of Radioactive Materials must select a route for each shipment. Seasonal weather conditions, highway repair or construction, highways that are closed because of natural events (for example, a landslide in North Carolina closed Interstate 40 near the border with Tennessee from June until November 1997), and other events (for example, the 1996 Olympic Games in Atlanta, Georgia) are all factors that must be considered in selecting preferred route segments to reduce time in transit. For this analysis, the highway routes were selected by the HIGHWAY program using an assumption of normal travel and without consideration for factors such as seasons of the year or road construction delays. Although these shipments could use other routes, DOE considers the impacts determined in the analyses to be representative of other possible routings that would also comply with Department of Transportation regulations. Specific route mileages for truck transportation are presented in Section J.1.2.1.1.

In selecting existing routes for use in the analysis, the HIGHWAY program determined the length of travel in each type of population zone—rural, suburban, and urban. The program characterized rural, suburban, and urban population areas according to the following breakdown: rural population densities range from 0 to 54 persons per square kilometer (0 to 140 persons per square mile); the suburban range is 55 to 1,300 persons per square kilometer (140 to 3,300 persons per square mile); and urban is all population densities greater than 1,300 persons per square kilometer (3,300 persons per square mile). The population densities along a route used by the HIGHWAY program are derived from 1990 data from the Bureau of the Census.

J.1.1.3 INTERLINE

Shipments of radioactive materials by rail are not subject to route restrictions imposed by regulations. For general freight rail service, DOE anticipates that railroads would route shipments of spent nuclear fuel and high-level radioactive waste to provide expeditious travel and the minimum practical number of interchanges between railroads. The selection of a route determines the potentially exposed population along the route as well as the expected frequency of transportation-related accidents. The analysis used the INTERLINE computer program (Johnson et al. 1993b, all) to project the railroad routes that DOE would use to ship spent nuclear fuel and high-level radioactive waste from the sites to the Yucca Mountain site. Specific routes were projected for each originating generator with the exception of 9 that do not have capability to handle or load a rail transportation cask (see Section J.1.2.1.1, Table J-6).

INTERLINE computes rail routes based on rules that simulate historic routing practices of U.S. railroads. The INTERLINE data base consists of 94 separate subnetworks and represents various competing rail companies in the United States. The data base, which was originally based on data from the Federal Railroad Administration and reflected the U.S. railroad system in 1974, has been expanded and modified extensively over the past two decades. The program is updated periodically to reflect current track conditions and has been benchmarked against reported mileages and observations of commercial rail firms. The program also provides an estimate of the population within 800 meters (0.5 mile) of the routes it selected. This population estimate was used to calculate incident-free radiological risk to the public along the routes selected for analysis.

In general, rail routes are calculated by minimizing the value of a factor called *impedance* between the origin and the destination. The impedance is determined by considering trip distance along a route, the mainline classification of the rail lines that would be used, and the number of interchanges that would occur between different railroad companies involved. In general, impedance determined by the INTERLINE program:

- Decreases as the distance traveled decreases
- Is reduced by use of mainline track that has the highest traffic volume (see below)
- Is reduced for shipments that involve the fewest number of railroad companies

Thus, routes that are the most direct, that use high-traffic volume mainline track, and that involve only one railroad company would have the lowest impedance. The most important of these characteristics from a routing standpoint is the *mainline classification*, which is the measure of traffic volume on a particular link. The mainline classifications used in the INTERLINE routing model are as follows:

- A – mainline – more than 20 million gross ton miles per year
- B – mainline – between 5 and 20 million gross ton miles per year
- A – branch line – between 1 and 5 million gross ton miles per year
- B – branch line – less than 1 million gross ton miles per year

The INTERLINE routing algorithm is designed to route a shipment preferentially on the rail lines having the highest traffic volume. Frequently traveled routes are preferred because they are generally well maintained because the railroad depends on these lines for a major portion of its revenue. In addition, routing along the high-traffic lines usually replicates railroad operational practices.

The population densities along a route were derived from 1990 data from the Bureau of the Census, as described above for the HIGHWAY computer program.

DOE anticipates that routing of rail shipments in dedicated (special) train service, if used, would be similar to routing of general freight shipments for the same origin and destination pairs. However, because cask cars would not be switched between trains at classification yards, dedicated train service would be likely to result in less time in transit.

J.1.1.4 RADTRAN4

The RADTRAN4 computer program (Neuhauser and Kanipe 1992, all) was used for the routine and accident cargo-related risk assessment to estimate the radiological impacts to collective populations. RADTRAN4 was developed by Sandia National Laboratories to calculate population risks associated with the transportation of radioactive materials by a variety of modes, including truck, rail, air, ship, and barge. The code has been used extensively for transportation risk assessment since it was issued in the late 1970s and has been reviewed and updated periodically. In 1995, a validation of the RADTRAN4 code demonstrated that it yielded acceptable results (Maheras and Pippen 1995, page iii). In the context of the validation analysis, *acceptable results* means that the difference between the estimates generated by the RADTRAN4 code and hand calculations were small, that is, less than 5 percent (Maheras and Pippen 1995, page 3-1).

The RADTRAN4 calculations for routine (or incident-free) dose are based on expressing the dose rate as a function of distance from a point source. Associated with the calculation of routine doses for each exposed population group are parameters such as the radiation field strength, the source-receptor distance, the duration of the exposure, vehicular speed, stopping time, traffic density, and route characteristics such as population density. In calculating population doses from incident-free transportation, the RADTRAN4 program used population density data provided by the HIGHWAY and INTERLINE computer programs. These data are based on the 1990 Census.

In addition to routine doses, RADTRAN4 was used to estimate dose risk from a spectrum of accident scenarios. The spectrum of accident scenarios encompass the range of possible accidents, including low-probability accident scenarios that have high consequences, and high-probability accident scenarios that have low consequences (fender benders). The RADTRAN4 calculation of collective accident risk for populations along routes employed models that quantified the range of potential accident severities and the responses of the shipping casks to the accident scenarios. The spectrum of accident severity was divided into categories. Each category of severity received a conditional probability of occurrence; that is, the probability that an accident will be of a particular severity if an accident occurs — the more severe the accident, the more remote the chance of such an accident. A release fraction, which is the fraction of the material in a shipping cask that could be released in an accident, is assigned to each accident scenario severity category on the basis of the physical and chemical form of the material being transported. The model also takes into account the mode of transportation, the state-specific accident rates, and population densities for rural suburban, and urban population zones through which shipments would pass to estimate accident risks for this analysis. The RADTRAN4 program used actual population densities within 800 meters (0.5 mile) of transportation routes based on 1990 census data as the basis for estimating populations within 80 kilometers (50 miles).

For accident scenarios involving the release of radioactive material, RADTRAN4 assumes that the material is dispersed in the environment as described by a Gaussian dispersion model. The dispersion analysis assumes that meteorological conditions are national averages for wind speed and atmospheric stability. For the risk assessment, the analysis used these meteorological conditions and assumed an instantaneous ground-level release and a small diameter source cloud (Neuhauser and Kanipe 1993, page 5-6). The calculation of the collective population dose following the release and the dispersal of radioactive material includes the following exposure pathways:

- External exposure to the passing radioactive cloud
- External exposure to contaminated ground
- Internal exposure from inhalation of airborne contaminants
- Internal exposure from ingestion of contaminated food

For the ingestion pathway, the analysis used state-specific food transfer factors (TRW 1999a, page 35), which relate the amount of radioactive material ingested to the amount deposited on the ground, as input to the RADTRAN4 code. Radiation doses from the ingestion or inhalation of radionuclides were calculated by using standard dose conversion factors from Federal Guidance Reports No. 11 and 12 (TRW 1999a, page 36).

J.1.1.5 RISKIND

The RISKIND computer program (Yuan et al. 1995, all) was used as a complement to the RADTRAN4 calculations to estimate scenario-specific doses to maximally exposed individuals for both routine operations and accident conditions and to estimate population impacts for the assessment of accident scenario consequences. The RISKIND code was originally developed for the DOE Office of Civilian Radioactive Waste Management specifically to analyze radiological consequences to individuals and population subgroups from the transportation of spent nuclear fuel and is used now to analyze the transport of other radioactive materials, as well as spent nuclear fuel.

The RISKIND external dose model considers direct external exposure and exposure from radiation scattered from the ground and air. RISKIND was used to calculate the dose as a function of distance from a shipment on the basis of the dimensions of the shipment (millirem per hour for stationary exposures and millirem per event for moving shipments). The code approximates the shipment as a cylindrical volume source, and the calculated dose includes contributions from secondary radiation scatter from buildup (scattering by material contents), cloudshine (scattering by air), and groundshine (scattering by the ground). Credit for potential shielding between the shipment and the receptor was not considered.

The RISKIND code was also used to provide a scenario-specific assessment of radiological consequences of severe transportation-related accidents. Whereas the RADTRAN4 risk assessment considers the entire range of accident severities and their related probabilities, the RISKIND consequence assessment focuses on accident scenarios that result in the largest releases of radioactive material to the environment. The consequence assessment was intended to provide an estimate of the potential impacts posed by a severe, but highly unlikely, transportation-related accident scenario.

The dose to each maximally exposed individual considered was calculated with RISKIND for an exposure scenario defined by a given distance, duration, and frequency of exposure specific to that receptor. The distances and durations were similar to those given in previous transportation risk assessments. The scenarios were not meant to be exhaustive but were selected to provide a range of potential exposure situations.

J.1.2 NUMBER AND ROUTING OF SHIPMENTS

This section discusses the number of shipments and routing information used to analyze potential impacts that would result from preparation for and conduct of transportation operations to ship spent nuclear fuel and high-level radioactive waste to the Yucca Mountain site. Table J-1 summarizes the estimated numbers of shipments for the various inventory and national shipment scenario combinations.

J.1.2.1 Number of Shipments

DOE used two analysis scenarios—mostly legal-weight truck and mostly train (rail)—as bases for estimating the number of shipments of spent nuclear fuel and high-level radioactive waste from 72 commercial and 5 DOE sites. The number of shipments for the scenarios was used in analyzing transportation impacts for the Proposed Action and Inventory Modules 1 and 2. DOE selected the scenarios because, more than 10 years before the projected start of operations at the repository, it cannot accurately predict the actual mix of rail and legal-weight truck transportation that would occur from the 77 sites to the repository. Therefore, the selected scenarios enable the analysis to bound (or bracket) the ranges of legal-weight truck and rail shipments that could occur.

Table J-1. Summary of estimated numbers of shipments for the various inventory and national transportation analysis scenario combinations.

	Mostly truck		Mostly rail	
	Truck	Rail	Truck	Rail
Proposed Action				
Commercial spent nuclear fuel	37,738	0	2,601	8,386
High-level radioactive waste	8,315	0	0	1,663
Spent nuclear fuel	3,470	300	0	766
Greater-Than-Class-C waste	0	0	0	0
Special-Performance-Assessment-Required waste	0	0	0	0
Proposed Action totals	**49,523**	**300**	**2,601**	**10,815**
Module 1[a]				
Commercial spent nuclear fuel	66,850	0	3,701	13,906
High-level radioactive waste	22,280	0	0	4,456
Spent nuclear fuel	3,721	300	0	797
Greater-Than-Class-C waste	0	0	0	0
Special-Performance-Assessment-Required waste	0	0	0	0
Module 1 totals	**92,851**	**300**	**3,701**	**19,159**
Module 2[a]				
Commercial spent nuclear fuel	66,850	0	3,701	13,906
High-level radioactive waste	22,280	0	0	4,456
Spent nuclear fuel	3,721	300	0	797
Greater-Than-Class-C waste	1,096	0	0	282
Special-Performance-Assessment-Required waste	2,010	0	0	404
Module 2 totals	**95,957**	**300**	**3,701**	**19,845**

a. The number of shipments for Module 1 includes all shipments of spent nuclear fuel and high-level radioactive waste included in the Proposed Action and shipments of additional spent nuclear fuel and high-level radioactive waste as described in Appendix A. The number of shipments for Module 2 includes all the shipments in Module 1 and additional shipments of highly radioactive materials described in Appendix A.

The analysis estimated the number of shipments from commercial sites where spent nuclear fuel would be loaded and shipped and from DOE sites where spent nuclear fuel, naval spent nuclear fuel, and high-level radioactive waste would be loaded and shipped.

For the mostly legal-weight truck scenario, with one exception, shipments were assumed to use legal-weight trucks. Overweight, overdimensional trucks weighing between about 36,300 and 52,300 kilograms (80,000 and 115,000 pounds) but otherwise similar to legal-weight trucks could be used for some spent nuclear fuel and high-level radioactive waste (for example, spent nuclear fuel from the South Texas reactors). The exception that gives the scenario its name—mostly legal-weight truck—was for shipments of naval spent nuclear fuel. Under this scenario, naval spent nuclear fuel would have to be shipped by rail because of the size and weight of the shipping container (cask) that would be used.

For the mostly rail scenario, the analysis assumed that all sites would ship by rail, with the exception of those with physical limitations that would make rail shipment impractical. The exception would be for shipments by legal-weight trucks from 9 commercial sites that do not have the capability to load rail casks. The analysis assumed that 19 commercial sites that do not have direct rail service but that could handle large casks would ship by barge or heavy-haul truck to nearby railheads with intermodal capability.

For commercial spent nuclear fuel, the CALVIN code was used to compute the number of shipments. The number of shipments of DOE spent nuclear fuel and high-level radioactive waste was estimated based on the data in Appendix A and information provided by the DOE sites. The numbers of shipments were estimated based on the characteristics of the materials shipped, mode interface capability (for example, the lift capacity of the cask-handling crane) of each shipping facility, and the modal-mix case analyzed. Table J-2 summarizes the basis for the national and Nevada transportation impact analysis.

Table J-2. Analysis basis—national and Nevada transportation scenarios.[a,b]

Material	Mostly legal-weight truck scenario national and Nevada	National mostly rail scenario: Nevada rail scenario	National mostly rail scenario: Nevada heavy-haul truck scenario
Casks			
Commercial SNF	Truck casks – about 1.8 MTHM per cask	Rail casks – 6 to 12 MTHM per cask for shipments from 63 sites	Rail casks – 6 to 12 MTHM per cask for shipments from 63 sites
		Truck casks – about 1.8 MTHM per cask for shipments from 9 sites	Truck casks – about 1.8 MTHM per cask for shipments from 9 sites
DOE HLW and DOE SNF, except naval SNF	Truck casks – 1 SNF or HLW canister per cask	Rail casks – four to nine SNF or HLW canisters per cask	Rail casks – four to nine SNF or HLW canisters per cask
Naval SNF	Disposal canisters in large rail casks for shipment from INEEL	Disposable canisters in large rail casks for shipments from INEEL	Disposable canisters in large rail casks for shipments from INEEL
Transportation modes			
Commercial SNF	Legal-weight trucks	Direct rail from 44 sites served by railroads to repository	Rail from 44 sites served by railroads to intermodal transfer station in Nevada, then heavy-haul trucks to repository
		Heavy-haul trucks from 5 sites to railhead, then rail to repository	Heavy-haul trucks from 5 sites to railheads, then rail to intermodal transfer station in Nevada, then heavy-haul trucks to repository
		Heavy-haul trucks or barges[c] from 14 sites to railhead, then rail to repository	Heavy-haul trucks or barges from 14 sites to railheads, then rail to intermodal transfer station in Nevada, then heavy-haul trucks to repository[e]
		Legal-weight trucks from 9 sites to repository	Legal-weight trucks from 9 sites to repository
DOE HLW and DOE SNF, except naval SNF	Legal-weight trucks	Rail from DOE sites[d] to repository	Rail from DOE sites to intermodal transfer station in Nevada, then heavy-haul trucks to repository
Naval SNF	Rail from INEEL to intermodal transfer station in Nevada, then heavy-haul trucks to repository	Rail from INEEL to repository	Rail from INEEL to intermodal transfer station in Nevada, then heavy-haul trucks to repository

a. Abbreviations: SNF = spent nuclear fuel; MTHM = metric tons of heavy metal; HLW = high-level radioactive waste; INEEL = Idaho National Engineering and Environmental Laboratory.

b. G. E. Morris facility is included with the Dresden reactor facilities in the 72 commercial sites.

c. Fourteen of 19 commercial sites not served by a railroad are on or near a navigable waterway. Some of these 14 sites could ship by barge rather than by heavy-haul truck to a nearby railhead.

d. Hanford Site, Savannah River Site, Idaho National Engineering and Environmental Laboratory, West Valley Demonstration Project, and Ft. St. Vrain.

Detailed descriptions of spent nuclear fuel and high-level radioactive waste that would be shipped to the Yucca Mountain site are presented in Appendix A.

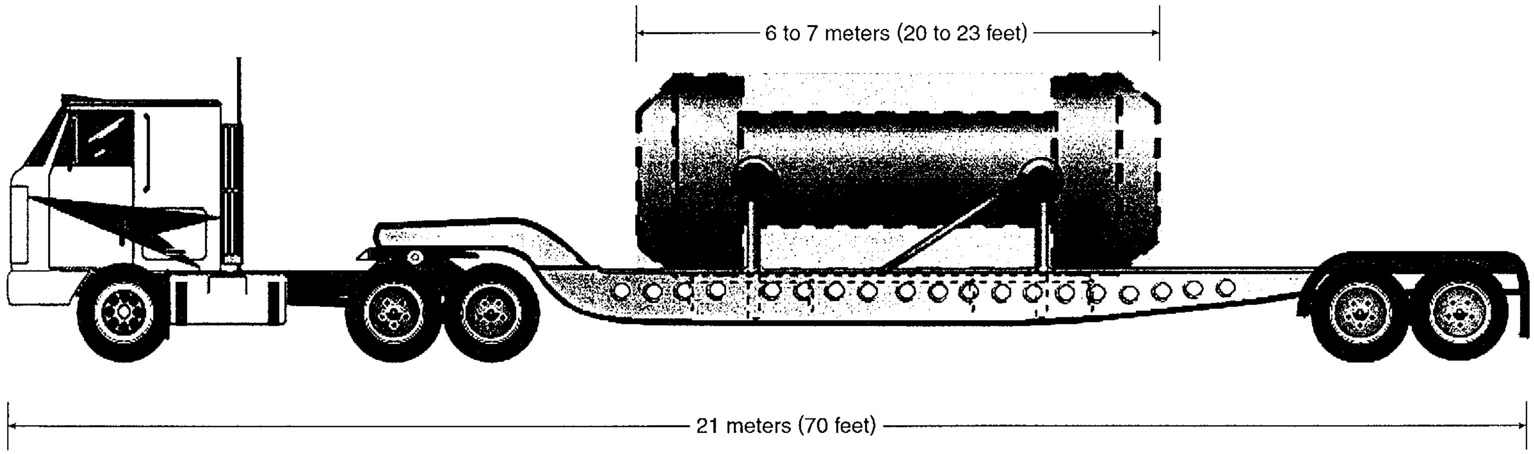
J.1.2.1.1 *Commercial Spent Nuclear Fuel*

For the analysis, the CALVIN model used 32 shipping cask configurations: 15 for legal-weight truck casks (Figure J-3) and 17 for rail casks (Figure J-4). Table J-3 lists the legal-weight truck and rail cask configurations used in the analysis and their capacities. The analysis assumed that all shipments would use one of the 32 configurations. If the characteristics of the spent nuclear fuel projected for shipment exceeded the capabilities of one of the casks, the model reduced the cask's capacity for the affected shipments. The reduction, which is sometimes referred to as cask derating, was needed to satisfy nuclear criticality, shielding, and thermal constraints. For shipments that DOE would make using specific casks, derating would be accomplished by partially filling the assigned casks in compliance with provisions of applicable Nuclear Regulatory Commission certificates of compliance. An example of derating is discussed in Section 5 of the GA-4 legal-weight truck shipping cask design report (General Atomics 1993, page 5.5-1). The analysis addresses transport of two high-burnup or short cooling time pressurized-water reactor assemblies rather than four design basis assemblies.

RAIL SHIPMENTS

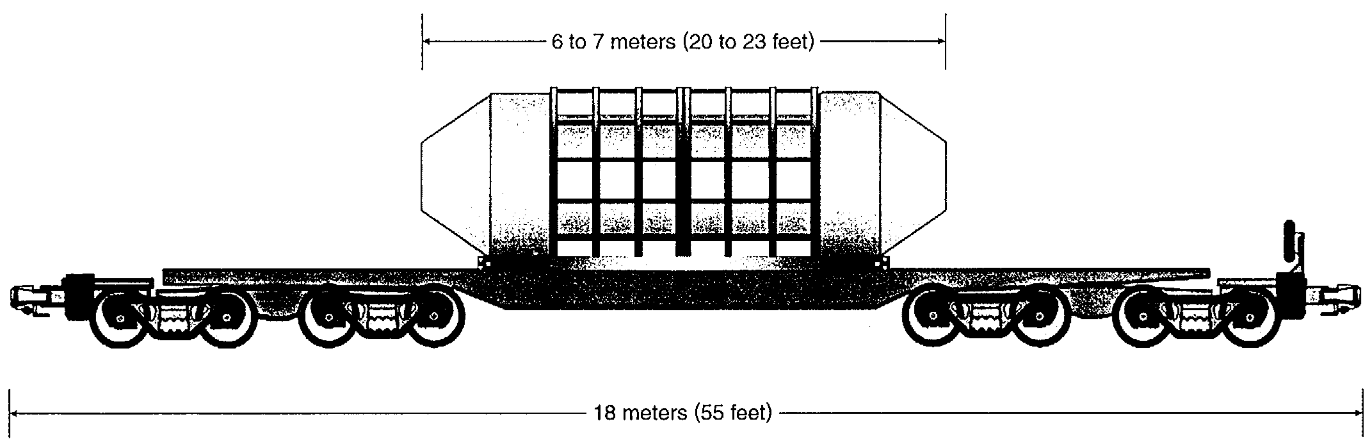
This appendix assumes that rail shipments of spent nuclear fuel would use large rail shipping casks, one per railcar. DOE anticipates that as many as five railcars with casks containing spent nuclear fuel or high-level radioactive waste would move together in individual trains with buffer cars and escort cars. For general freight service, a train would include other railcars with other materials. In dedicated (or special) service, trains would move only railcars containing spent nuclear fuel or high-level radioactive waste and the buffer and escort cars.

For the mostly rail scenario, 9 sites without sufficient crane capacity to lift a rail cask or without other factors such as sufficient floor loading capacity or ceiling height were assumed to ship by legal-weight truck. The 19 sites with sufficient crane capacity but without direct rail access were assumed to ship by heavy-haul truck to the nearest railhead. Of these 19 sites, 14 with access to navigable waterways were analyzed for shipping by barge to a railhead (see Section J.2.1). The number of rail shipments (direct or indirect) was estimated based on each site using the largest cask size feasible based on the load capacity of its cask handling crane. In calculating the number of shipments from the sites, the model used the DOE allocation of delivery rights (10 CFR Part 961) to the sites and the anticipated receipt rate at the repository listed in Table J-4. Using CALVIN, the number of shipments of legal-weight truck casks (Figure J-3) of commercial spent nuclear fuel estimated for the Proposed Action (63,000 MTU of commercial spent nuclear fuel) for the mostly legal-weight truck scenario, would be about 14,000 containing boiling-water reactor assemblies and 24,000 containing pressurized-water reactor assemblies. Under Inventory Modules 1 and 2, for which approximately 105,000 MTU of commercial spent nuclear fuel would be shipped to the repository (see Appendix A), the estimated number of shipments for the mostly legal-weight truck scenario would be 24,000 for boiling-water reactor spent nuclear fuel and 43,000 for pressurized-water reactor spent nuclear fuel. Table J-5 lists the number of shipments of commercial spent nuclear fuel for the mostly legal-weight truck scenario. Specifically, it lists the site, plant, and state where shipments would originate, the total number of shipments from each site, and the type of spent nuclear fuel that would be shipped. A total of 72 commercial sites with 104 plants (or facilities) are listed in the table.



Source: Kelderhouse (1999, page 7).

Figure J-3. Artist's conception of a truck cask on a legal-weight tractor-trailer truck.



Source: Kelderhouse (1999, page 8).

Figure J-4. Artist's conception of a large rail cask on a railcar.

Table J-3. Shipping cask configurations.

Shipping casks	Capacity (number of spent nuclear fuel assemblies)	Description[a,b]
Rail		
B-RAIL-LGSP	61	Large BWR single-purpose shipping container
B-RAIL-SMSP	24	Small BWR single-purpose shipping container
BP-TRAN-OVLG74	74	Big Rock Point dual-purpose shipping container
B-TRAN-OVLG	61	Large BWR dual-purpose shipping container
B-TRAN-OVMED	44	Medium BWR dual-purpose shipping container
B-TRAN OVSM	24	Small BWR dual-purpose shipping container
B-High Heat Rail	17	BWR high heat shipping container
P-RAIL-LGSP	26	Large PWR single-purpose shipping container
P-RAIL-SMSP	12	Small PWR single-purpose shipping container
P-RAIL-MOX	9	Mixed-oxide SNF shipping container
P-RL-LGSP-ST	12	South Texas single-purpose shipping container
P-TRAN-OVLG-YR	36	Yankee Rowe dual-purpose shipping container
P-TRAN-OVLG	24	Large PWR dual-purpose shipping container
P-TRAN-OVMED	21	Medium PWR dual-purpose shipping container
P-TRAN-OVSM	12	Small PWR dual-purpose shipping container
P-TRNST-OVLG	12	South Texas dual-purpose shipping container
P-High Heat-Rail	7	PWR high heat shipping container
Truck		
B-LWT-GA9I	9	Primary BWR shipping container
B-LWT-GA9II	7	Derated BWR shipping container
B-LWT-GA9III	5	Derated BWR shipping container
B-LWT-GA9IV	4	Derated BWR shipping container
B-LWT-GAV	2	Derated BWR shipping container
BP-LWT-GA4I	4	Big Rock Point shipping container
B-NLI-1/2	2	Secondary BWR shipping container
P-LWT-GA4I	4	Primary PWR shipping container
P-LWT-GA4II	3	Derated PWR shipping container
P-LWT-GA4III	2	Derated PWR shipping container
P-LWT-GA4I-ST	4	South Texas shipping container
P-LWT-GA4II-ST	3	Derated South Texas shipping container
P-LWT-GA4III-ST	2	Derated South Texas shipping container
P-NLI-1/2	1	Secondary PWR shipping container
P-LWT-MOX	4	Mixed-oxide SNF shipping container

a. Source: TRW (1999a, page 3).
b. BWR = boiling-water reactor; PWR = pressurized-water reactor; SNF = spent nuclear fuel.

The number of shipments of truck and rail casks (Figure J-4) of commercial spent nuclear fuel estimated for the Proposed Action for the mostly rail scenario would be 4,200 for boiling-water reactor spent nuclear fuel and 6,800 for pressurized-water reactor spent nuclear fuel. Under Modules 1 and 2, the estimated number of shipments for the mostly rail scenario would be 6,500 containing boiling-water reactor spent nuclear fuel and 11,100 containing pressurized-water reactor spent nuclear fuel. Table J-6 lists the number of shipments for the mostly rail scenario. It also lists the site and state where shipments would originate, the total number of shipments from each site, the size of rail cask assumed for each site, and the type of spent nuclear fuel that would be shipped. In addition, it lists the 19 sites not served by a railroad that would ship rail casks by barge or heavy-haul trucks to a nearby railhead and the 9 commercial sites without capability to load a rail cask.

Table J-4. Anticipated receipt rate for spent nuclear fuel and high-level radioactive waste at the Yucca Mountain Repository[a].

	Commercial spent nuclear fuel annual receipt[b]			High-level radioactive waste and DOE spent nuclear fuel[c] annual receipts		
		Shipments			Shipments	
Year	MTHM[d]	Mostly LWT[e]	Mostly rail	MTHM	Mostly LWT	Mostly rail
2010	300	267	100	0	0	0
2011	600	413	184	0	0	0
2012	1,200	757	294	0	0	0
2013	2,000	1,246	478	0	0	0
2014	3,000	1,805	663	0	0	0
2015	3,000	1,792	638	400	650	140
2016	3,000	1,797	600	400	650	140
2017	3,000	1,803	555	400	650	140
2018	3,000	1,787	497	400	650	140
2019	3,000	1,782	508	400	650	140
2020	3,000	1,773	501	400	650	140
2021	3,000	1,780	514	400	650	140
2022	3,000	1,771	513	400	650	140
2023	3,000	1,772	484	400	650	140
2024	3,000	1,796	496	400	650	140
2025	3,000	1,779	472	400	650	140
2026	3,000	1,777	437	400	650	140
2027	3,000	1,793	488	400	650	140
2028	3,000	1,772	469	400	650	140
2029	3,000	1,794	460	400	650	140
2030	3,000	1,768	419	400	675	140
2031	3,000	1,808	451	400	685	140
2032	3,000	1,781	458	200	675	49
2033	1,900	1,125	308	0	0	0
Totals	**63,000**	**37,738**	**10,987**	**7,000**	**12,085**	**2,429**

a. Receipt rates based on assumptions presented in the *Analysis of the Total System Life-Cycle Cost of the Civilian Radioactive Waste Management Program* (DOE 1998a, all) and the results of the CALVIN analysis.
b. Projected spent nuclear fuel acceptance rates (until agreements are reached with purchasers/producers/custodians).
c. DOE spent nuclear fuel at the Idaho National Engineering and Environmental Laboratory to be removed by 2035. Three hundred rail shipments of Navy fuel will be among the early shipments to a DOE receiving facility.
d. MTHM = metric tons of heavy metal.
e. LWT = legal-weight truck.

J.1.2.1.2 *DOE Spent Nuclear Fuel and High-Level Radioactive Waste*

To estimate the number of DOE spent nuclear fuel and high-level radioactive waste shipments, the analysis used the number of handling units or number of canisters and the number of canisters per shipment reported by the DOE sites in 1998 (see Appendix A, page A-34; Jensen 1998, all). To determine the number of shipments of DOE spent nuclear fuel and high-level radioactive waste, the analysis assumed one canister would be shipped in a legal-weight truck cask. For rail shipments, the analysis assumed that five 61-centimeter (24-inch)-diameter high-level radioactive waste canisters would be shipped in a rail cask. For rail shipments of DOE spent nuclear fuel, the analysis assumed that rail casks would contain nine approximately 46-centimeter (18-inch) canisters or four approximately 61-centimeter canisters. The number of DOE spent nuclear fuel canisters of each size is presented in Appendix A.

Table J-5. Shipments of commercial spent nuclear fuel, mostly legal-weight truck scenario[a] (page 1 of 2).

Site	Reactor	State	Fuel type	Proposed Action (2010-2033)	Modules 1 and 2 (2010-2048)
Browns Ferry	Browns Ferry 1	AL	B[b]	856	1,465
	Browns Ferry 3	AL	B	319	602
Joseph M. Farley	Joseph M. Farley 1	AL	P[c]	336	544
	Joseph M. Farley 2	AL	P	297	582
Arkansas Nuclear One	Arkansas Nuclear One, Unit 1	AR	P	302	438
	Arkansas Nuclear One, Unit 2	AR	P	332	525
Palo Verde	Palo Verde 1	AZ	P	345	797
	Palo Verde 2	AZ	P	364	840
	Palo Verde 3	AZ	P	309	861
Diablo Canyon	Diablo Canyon 1	CA	P	327	617
	Diablo Canyon 2	CA	P	305	691
Humboldt Bay	Humboldt Bay	CA	B	44	44
Rancho Seco	Rancho Seco 1	CA	P	124	124
San Onofre	San Onofre 1	CA	P	52	52
	San Onofre 2	CA	P	402	600
	San Onofre 3	CA	P	413	632
Haddam Neck	Haddam Neck	CT	P	255	255
Millstone	Millstone 1	CT	B	463	543
	Millstone 2	CT	P	358	551
	Millstone 3	CT	P	245	575
Crystal River	Crystal River 3	FL	P	283	442
St. Lucie	St. Lucie 1	FL	P	389	571
	St. Lucie 2	FL	P	292	515
Turkey Point	Turkey Point 3	FL	P	295	413
	Turkey Point 4	FL	P	287	458
Edwin I. Hatch	Edwin I. Hatch 1	GA	B	871	1,334
Vogtle	Vogtle 1	GA	P	593	1,462
Duane Arnold	Duane Arnold	IA	B	279	420
Braidwood	Braidwood 1	IL	P	615	1,494
Byron	Byron 1	IL	P	617	1,444
Clinton	Clinton 1	IL	B	296	690
Dresden/Morris	Dresden 1	IL	B	76	76
	Dresden 2	IL	B	430	521
	Dresden 3	IL	B	473	565
	Morris[d]	IL	B	319	319
	Morris[d]	IL	P	88	88
LaSalle	LaSalle 1	IL	B	596	1,261
Quad Cities	Quad Cities 1	IL	B	798	1,123
Zion	Zion 1	IL	P	771	1,028
Wolf Creek	Wolf Creek 1	KS	P	349	708
River Bend	River Bend 1	LA	B	324	823
Waterford	Waterford 3	LA	P	313	675
Pilgrim	Pilgrim 1	MA	B	316	476
Yankee-Rowe	Yankee-Rowe 1	MA	P	134	134
Calvert Cliffs	Calvert Cliffs 1	MD	P	757	1,140
Maine Yankee	Maine Yankee	ME	P	356	356
Big Rock Point	Big Rock Point	MI	B	131	131
D. C. Cook	D. C. Cook 1	MI	P	824	1,235
Fermi	Fermi 2	MI	B	312	764
Palisades	Palisades	MI	P	367	454
Monticello	Monticello	MN	B	267	342
Prairie Island	Prairie Island 1	MN	P	572	805
Callaway	Callaway 1	MO	P	392	735
Grand Gulf	Grand Gulf 1	MS	B	516	1,016
Brunswick	Brunswick 1	NC	P	40	40
	Brunswick 2	NC	P	36	36

Table J-5. Shipments of commercial spent nuclear fuel, mostly legal-weight truck scenario[a] (page 2 of 2).

Site	Reactor	State	Fuel type	Proposed Action (2010-2033)	Modules 1 and 2 (2010-2048)
Brunswick (continued)					
	Brunswick 1	NC	B[b]	232	426
	Brunswick 2	NC	B	232	401
Shearon Harris	Shearon Harris 1	NC	P[c]	298	769
	Shearon Harris	NC	B	152	152
McGuire	McGuire 1	NC	P	387	690
	McGuire 2	NC	P	436	774
Cooper Station	Cooper Station	NE	B	274	454
Fort Calhoun	Fort Calhoun	NE	P	258	362
Seabrook	Seabrook 1	NH	P	235	630
Oyster Creek	Oyster Creek 1	NJ	B	424	519
Salem/Hope Creek	Salem 1	NJ	P	330	545
	Salem 2	NJ	P	298	571
	Hope Creek	NJ	B	399	876
James A. FitzPatrick/ Nine Mile Point	James A. FitzPatrick	NY	B	364	554
	Nine Mile Point 1	NY	B	401	499
	Nine Mile Point 2	NY	B	329	918
Ginna	Ginna	NY	P	309	379
Indian Point	Indian Point 1	NY	P	40	40
	Indian Point 2	NY	P	364	590
	Indian Point 3	NY	P	297	525
Davis-Besse	Davis-Besse 1	OH	P	286	535
Perry	Perry 1	OH	B	288	631
Trojan	Trojan	OR	P	195	195
Beaver Valley	Beaver Valley 1	PA	P	330	534
	Beaver Valley 2	PA	P	221	622
Limerick	Limerick 1	PA	B	693	1,722
Peach Bottom	Peach Bottom 2	PA	B	480	696
	Peach Bottom 3	PA	B	444	712
Susquehanna	Susquehanna 1	PA	B	808	1,582
Three Mile Island	Three Mile Island 1	PA	P	287	435
Catawba	Catawba 1	SC	P	325	663
	Catawba 2	SC	P	318	667
Oconee	Oconee 1	SC	P	727	1,043
	Oconee 3	SC	P	280	457
H. B. Robinson	H. B. Robinson 2	SC	P	231	306
Summer	Summer 1	SC	P	291	538
Sequoyah	Sequoyah	TN	P	560	1,179
Watts Bar	Watts Bar 1	TN	P	146	840
Comanche Peak	Comanche Peak 1	TX	P	559	1,558
South Texas	South Texas 1	TX	P	256	738
	South Texas 2	TX	P	229	710
North Anna	North Anna 1	VA	P	634	1,079
Surry	Surry 1	VA	P	647	902
Vermont Yankee	Vermont Yankee 1	VT	B	369	484
WPPSS[e] 2	WPPSS 2	WA	B	353	736
Kewaunee	Kewaunee	WI	P	288	401
LaCrosse	LaCrosse	WI	B	37	37
Point Beach	Point Beach	WI	P	575	742
Total BWR[b]				**13,965**	**23,914**
Total PWR[c]				**23,773**	**42,936**

a. Source: TRW (1999a, Section 2).
b. B = boiling-water reactor (BWR).
c. P = pressurized-water reactor (PWR).
d. Morris is a storage facility located close to the three Dresden reactors.
e. WPPSS = Washington Public Power Supply System.

Table J-6. Shipments of commercial spent nuclear fuel, mostly rail scenario[a] (page 1 of 2).

Site	Reactor	State	Fuel type	Cask	Proposed Action 2010 - 2033	Modules 1 and 2 2010 - 2048
Browns Ferry	Browns Ferry 1	AL	B[b]	Medium	239	422
	Browns Ferry 3	AL	B	Medium	88	168
Joseph M. Farley	Joseph M. Farley 1	AL	P[c]	Large	54	78
	Joseph M. Farley 2	AL	P	Large	49	79
Arkansas Nuclear One	Arkansas Nuclear One, Unit 1	AR	P	Medium	81	115
	Arkansas Nuclear One, Unit 2	AR	P	Medium	89	137
Palo Verde	Palo Verde 1	AZ	P	Large	53	120
	Palo Verde 2	AZ	P	Large	56	124
	Palo Verde 3	AZ	P	Large	47	106
Diablo Canyon	Diablo Canyon 1	CA	P	Medium	103	169
	Diablo Canyon 2	CA	P	Medium	97	174
Humboldt Bay	Humboldt Bay	CA	B	Truck	44	44
Rancho Seco	Rancho Seco 1	CA	P	Large	21	21
San Onofre	San Onofre 1	CA	P	Large	9	8
	San Onofre 2	CA	P	Large	66	97
	San Onofre 3	CA	P	Large	68	102
Haddam Neck	Haddam Neck	CT	P	Truck	255	255
Millstone	Millstone 1	CT	B	Small	174	204
	Millstone 2	CT	P	Small	120	183
	Millstone 3	CT	P	Medium	73	137
Crystal River	Crystal River 3	FL	P	Truck	283	442
St. Lucie	St. Lucie 1	FL	P	Truck	389	571
	St. Lucie 2	FL	P	Medium	88	140
Turkey Point	Turkey Point 3	FL	P	Medium	73	111
	Turkey Point 4	FL	P	Medium	72	117
Edwin I. Hatch	Edwin I. Hatch 1	GA	B	Large	128	197
Vogtle	Vogtle 1	GA	P	Small	195	431
Duane Arnold	Duane Arnold	IA	B	Small	105	158
Braidwood	Braidwood 1	IL	P	Large	95	215
Byron	Byron 1	IL	P	Large	136	244
Clinton	Clinton 1	IL	B	Medium	103	200
Dresden/Morris	Dresden 1	IL	B	Small	29	29
	Dresden 2	IL	B	Small	162	193
	Dresden 3	IL	B	Small	177	208
	Morris[d]	IL	B	Large	47	47
	Morris[d]	IL	P	Large	14	14
LaSalle	LaSalle 1	IL	B	Large	89	172
Quad Cities	Quad Cities 1	IL	B	Small	299	419
Zion	Zion 1	IL	P	Medium	147	250
Wolf Creek	Wolf Creek 1	KS	P	Large	52	106
River Bend	River Bend 1	LA	B	Large	48	101
Waterford	Waterford 3	LA	P	Large	49	91
Pilgrim	Pilgrim 1	MA	B	Truck	316	476
Yankee-Rowe	Yankee-Rowe 1	MA	P	Large	15	15
Calvert Cliffs	Calvert Cliffs 1	MD	P	Medium	198	303
Maine Yankee	Maine Yankee	ME	P	Large	60	60
Big Rock Point	Big Rock Point	MI	B	Large	8	8
D. C. Cook	D. C. Cook 1	MI	P	Medium	214	346
Fermi	Fermi 2	MI	B	Medium	100	199
Palisades	Palisades	MI	P	Medium	78	117
Monticello	Monticello	MN	B	Truck	267	342
Prairie Island	Prairie Island 1	MN	P	Medium	151	221
Callaway	Callaway 1	MO	P	Large	62	114
Grand Gulf	Grand Gulf 1	MS	B	Large	76	143

Table J-6. Shipments of commercial spent nuclear fuel, mostly rail scenario[a] (page 2 of 2).

Site	Reactor	State	Fuel type	Cask	Proposed Action 2010 - 2033	Modules 1 and 2 2010 - 2048
Brunswick	Brunswick 1	NC	P[c]	Small	14	14
	Brunswick 2	NC	P	Small	12	12
	Brunswick 1	NC	B[b]	Small	88	150
	Brunswick 2	NC	B	Small	87	145
Shearon Harris	Shearon Harris 1	NC	P	Small	93	201
	Shearon Harris	NC	B	Small	57	57
McGuire	McGuire 1	NC	P	Medium	115	199
	McGuire 2	NC	P	Medium	138	228
Cooper Station	Cooper Station	NE	B	Small	103	166
Fort Calhoun	Fort Calhoun	NE	P	Small	87	121
Seabrook	Seabrook 1	NH	P	Large	37	83
Oyster Creek	Oyster Creek 1	NJ	B	Medium	108	151
Salem/Hope Creek	Salem 1	NJ	P	Medium	97	153
	Salem 2	NJ	P	Medium	83	143
	Hope Creek	NJ	B	Large	59	125
James A. FitzPatrick/ Nine Mile Point	FitzPatrick	NY	B	Large	54	79
	Nine Mile Point 1	NY	B	Medium	135	167
	Nine Mile Point 2	NY	B	Medium	101	206
Ginna	Ginna	NY	P	Truck	309	379
Indian Point	Indian Point 1	NY	P	Truck	40	40
	Indian Point 2	NY	P	Truck	364	590
	Indian Point 3	NY	P	Truck	297	525
Davis-Besse	Davis-Besse 1	OH	P	Large	44	71
Perry	Perry 1	OH	B	Large	42	82
Trojan	Trojan	OR	P	Large	33	33
Beaver Valley	Beaver Valley 1	PA	P	Large	52	81
	Beaver Valley 2	PA	P	Large	34	79
Limerick	Limerick 1	PA	B	Medium	262	497
Peach Bottom	Peach Bottom 2	PA	B	Medium	138	206
	Peach Bottom 3	PA	B	Medium	127	197
Susquehanna	Susquehanna 1	PA	B	Large	119	219
Three Mile Island	Three Mile Island 1	PA	P	Medium	71	113
Catawba	Catawba 1	SC	P	Large	72	123
	Catawba 2	SC	P	Large	76	130
Oconee	Oconee 1	SC	P	Medium	187	266
	Oconee 3	SC	P	Medium	67	107
H. B. Robinson	H. B. Robinson 2	SC	P	Small	75	97
Summer	Summer 1	SC	P	Large	46	82
Sequoyah	Sequoyah	TN	P	Large	90	161
Watts Bar	Watts Bar 1	TN	P	Large	21	121
Comanche Peak	Comanche Peak 1	TX	P	Large	90	246
South Texas	South Texas 1	TX	P	Large	79	180
	South Texas 2	TX	P	Large	72	178
North Anna	North Anna 1	VA	P	Large	101	167
Surry	Surry 1	VA	P	Large	105	144
Vermont Yankee	Vermont Yankee 1	VT	B	Small	139	182
WPPSS[e] 2	WPPSS 2	WA	B	Large	53	107
Kewaunee	Kewaunee	WI	P	Medium	73	106
La Crosse	La Crosse	WI	B	Truck	37	37
Point Beach	Point Beach	WI	P	Large	93	118
Total BWR[b]					**4,208**	**6,503**
Total PWR[c]					**6,779**	**11,104**

a. Source: TRW (1999a, Section 2).
b. B = boiling-water reactor (BWR).
c. P = pressurized-water reactor (PWR).
d. Morris is a storage facility located close to the three Dresden reactors.
e. WPPSS = Washington Public Power Supply System.

Under the mostly legal-weight truck scenario for the Proposed Action, a total of about 11,800 truck shipments of DOE spent nuclear fuel and high-level radioactive waste would be shipped to the repository. In addition, due to the size and weight of the shipping casks for canisters that would contain naval spent fuel, DOE would transport 300 shipments of naval spent fuel by rail from the Idaho National Engineering and Environmental Laboratory to the repository. For Modules 1 and 2, under the mostly legal-weight truck scenario, the analysis estimated 3,740 DOE spent nuclear fuel and 22,300 high-level radioactive waste truck shipments and 300 naval spent nuclear fuel shipments by rail.

Under the mostly rail scenario for the Proposed Action, the analysis estimated that 770 railcar shipments of DOE spent nuclear fuel, including 300 railcar shipments of naval spent nuclear fuel (one naval spent nuclear fuel canister per rail cask), and 1,660 railcar shipments of high-level waste would travel to the repository. For Modules 1 and 2, under this scenario 800 railcar shipments of DOE spent nuclear fuel, including 300 railcar shipments of naval spent nuclear fuel, and 4,460 railcar shipments of high-level radioactive waste would be shipped. Table J-7 lists the estimated number of shipments of DOE spent nuclear fuel from each of the four sites for both the Proposed Action and Modules 1 and 2. Table J-8 lists the number of shipments of high-level radioactive waste for the Proposed Action and for Modules 1 and 2.

Table J-7. DOE spent nuclear fuel shipments by site.

	Proposed Action		Module 1 or 2	
Site	Mostly truck	Mostly rail	Mostly truck	Mostly rail
INEEL[a,b]	1,388	434	1,467	443
Savannah River Site	1,316	149	1,411	159
Hanford	754	147	809	157
Fort St. Vrain	312	36	334	38
Totals	**3,770**	**766**	**4,021**	**797**

a. INEEL = Idaho National Engineering and Environmental Laboratory.
b. Includes 300 railcar shipments of naval spent nuclear fuel.

Table J-8. Number of canisters of high-level radioactive waste and shipments from DOE sites.

	Proposed Action			Module 1 or 2	
Site	Canisters	Mostly truck	Mostly rail	Mostly truck	Mostly rail
INEEL[a]	1,300	0	0	1,300	260
Hanford	14,500	1,960	400	14,500	2,900
Savannah River Site	6,200	6,055	1,200	6,200	1,240
West Valley[b]	300	300	60	300	60
Totals	**22,300**	**8,315**	**1,660**	**22,300**	**4,460**

a. INEEL = Idaho National Engineering and Environmental Laboratory.
b. High-level radioactive waste at West Valley is commercial rather than DOE waste.

J.1.2.1.3 *Greater-Than-Class-C and Special-Performance-Assessment-Required Waste Shipments*

Reasonably foreseeable future actions could include shipment of Greater-Than-Class-C and Special-Performance-Assessment-Required waste to the Yucca Mountain Repository (Appendix A describes Greater-Than-Class-C and Special-Performance-Assessment-Required wastes). Commercial nuclear powerplants, research reactors, radioisotope manufacturers, and other manufacturing and research institutions generate low-level radioactive waste that exceeds the Nuclear Regulatory Commission Class

C shallow-land-burial disposal limits. In addition to DOE-held material, there are three other sources or categories of Greater-Than-Class-C low-level radioactive waste:

- Nuclear utilities
- Sealed sources
- Other generators

The activities of nuclear electric utilities and other radioactive waste generators to date have produced relatively small quantities of Greater-Than-Class-C low-level radioactive waste. As the utilities take their reactors out of service and decommission them, they could generate more waste of this type.

DOE Special-Performance-Assessment-Required low-level radioactive waste could include the following materials:

- Production reactor operating wastes
- Production and research reactor decommissioning wastes
- Non-fuel-bearing components of naval reactors
- Sealed radioisotope sources that exceed Class C limits for waste classification
- DOE isotope production-related wastes
- Research reactor fuel assembly hardware

The analysis estimated the number of shipments of Greater-Than-Class-C and Special-Performance-Assessment-Required waste by assuming that 10 cubic meters (about 350 cubic feet) would be shipped in a rail cask and 2 cubic meters (about 71 cubic feet) would be shipped in a truck cask. Table J-9 lists the resulting number of commercial Greater-Than-Class-C shipments in Inventory Module 2 for both truck and rail shipments. The shipments of Greater-Than-Class-C waste from commercial utilities would originate among the commercial reactor sites. Typically, boiling-water reactors would ship a total of about 9 cubic meters (about 318 cubic feet) of Greater-Than-Class-C waste per site, while pressurized-water reactors would ship about 20 cubic meters (about 710 cubic feet) per site (see Appendix A). The impacts of transporting this waste were examined for each reactor site. The analysis assumed that sealed sources and Greater-Than-Class-C waste identified as "other" would be shipped from the DOE Savannah River Site (see Table J-10).

Table J-9. Commercial Greater-Than-Class-C waste shipments.

Category	Volume (cubic meters)[a,b]	Truck	Rail
Commercial utilities	1,350	740	210
Sealed sources	240	120	25
Other	470	230	50
Total	**2,060**	**1,090**	**285**

a. Source: Appendix A.
b. To convert cubic meters to cubic feet, multiply by 35.314.

The analysis assumed DOE Special-Performance-Assessment-Required waste would be shipped from 4 DOE sites listed in Table J-10. Naval reactor and Argonne East Special-Performance-Assessment-Required waste is assumed to be shipped from the Idaho National Engineering and Environmental Laboratory.

J.1.2.1.4 *Sensitivity of Transportation Impacts to Number of Shipments*

As discussed in Section J.1.2.1, the number of shipments from commercial and DOE sites to the repository would depend on the mix of legal-weight truck and rail shipments. Because DOE has decided

Table J-10. DOE Special-Performance-Assessment-Required waste shipments.

Site[a]	Volume (cubic meters)[b,c]	Rail	Truck
Hanford	20	2	10
INEEL	520	57[d]	260
SRS (ORNL)	2,900	290	1,470
West Valley	550	56	280
Total	**3,990**	**405**	**2,020**

a. Abbreviations: INEEL = Idaho National Engineering and Environmental Laboratory; SRS = Savannah River Site; ORNL = Oak Ridge National Laboratory.
b. Source: Appendix A.
c. To convert cubic meters to cubic feet, multiply by 35.314.
d. Includes 55 shipments from naval reactors.

not to determine this mix at this time (10 years before the projected start of shipping operations), the analysis used two scenarios to provide results that bound the range of anticipated impacts. Thus, for a mix of legal-weight truck and rail shipments within the range of the mostly legal-weight truck and mostly rail scenarios, the impacts would be likely to lie within the bounds of the impacts predicted by the analysis. For example, a mix that is different from the scenarios analyzed could consist of 5,000 legal-weight truck shipments and 9,000 rail shipments over 24 years (compared to 2,600 and 10,800, respectively, for the mostly rail scenario). In this example, the number of traffic fatalities would be between 3.6 (estimated for the Proposed Action under the mostly rail scenario) and 3.9 (estimated for the mostly legal-weight truck scenario). Other examples that have different mixes within the ranges bounded by the scenarios would lead to results that would be within the range of the evaluated impacts.

In addition to mixes within the brackets, the number of shipments could fall outside the ranges used for the mostly legal-weight truck and rail transportation scenarios. If, for example, the mostly rail scenario used smaller rail casks than the analysis assumed, the number of shipments would be greater. If spent nuclear fuel was placed in the canisters before they were shipped, the added weight and size of the canisters would reduce the number of fuel assemblies that a given cask could accommodate; this would increase the number of shipments. However, for the mostly rail scenario, even if the capacity of the casks was half that used in the analysis, the impacts would remain below those forecast for the mostly legal-weight truck scenario. Although impacts would be related to the number of shipments, because the number of rail shipments would be very small in comparison to the total railcar traffic on the Nation's railroads, increases or decreases would be small for impacts to biological resources, air quality, hydrology, noise, and other environmental resource areas. Thus, the impacts of using smaller rail casks would be covered by the values estimated in this EIS.

For legal-weight truck shipments, the use of casks carrying smaller payloads than those used in the analysis (assuming the shipment of the same spent nuclear fuel) would lead to larger impacts for incident-free transportation and traffic fatalities and about the same level of radiological accident risk. The relationship is approximately linear; if the payloads of truck shipping casks in the mostly legal-weight truck scenario were less by one-half, the incident-free impacts would increase by approximately a factor of 2. Conversely, because the amount of radioactive material in a cask would be less (assuming shipment of the same spent nuclear fuel), the radiological consequences of maximum reasonably foreseeable accident scenarios would be less with the use of smaller casks. If smaller casks were used to accommodate shipments of spent nuclear fuel with shorter cooling time and higher burnup, the radiological consequences of maximum reasonably foreseeable accident scenarios would be about the same.

J.1.2.2 Transportation Routes

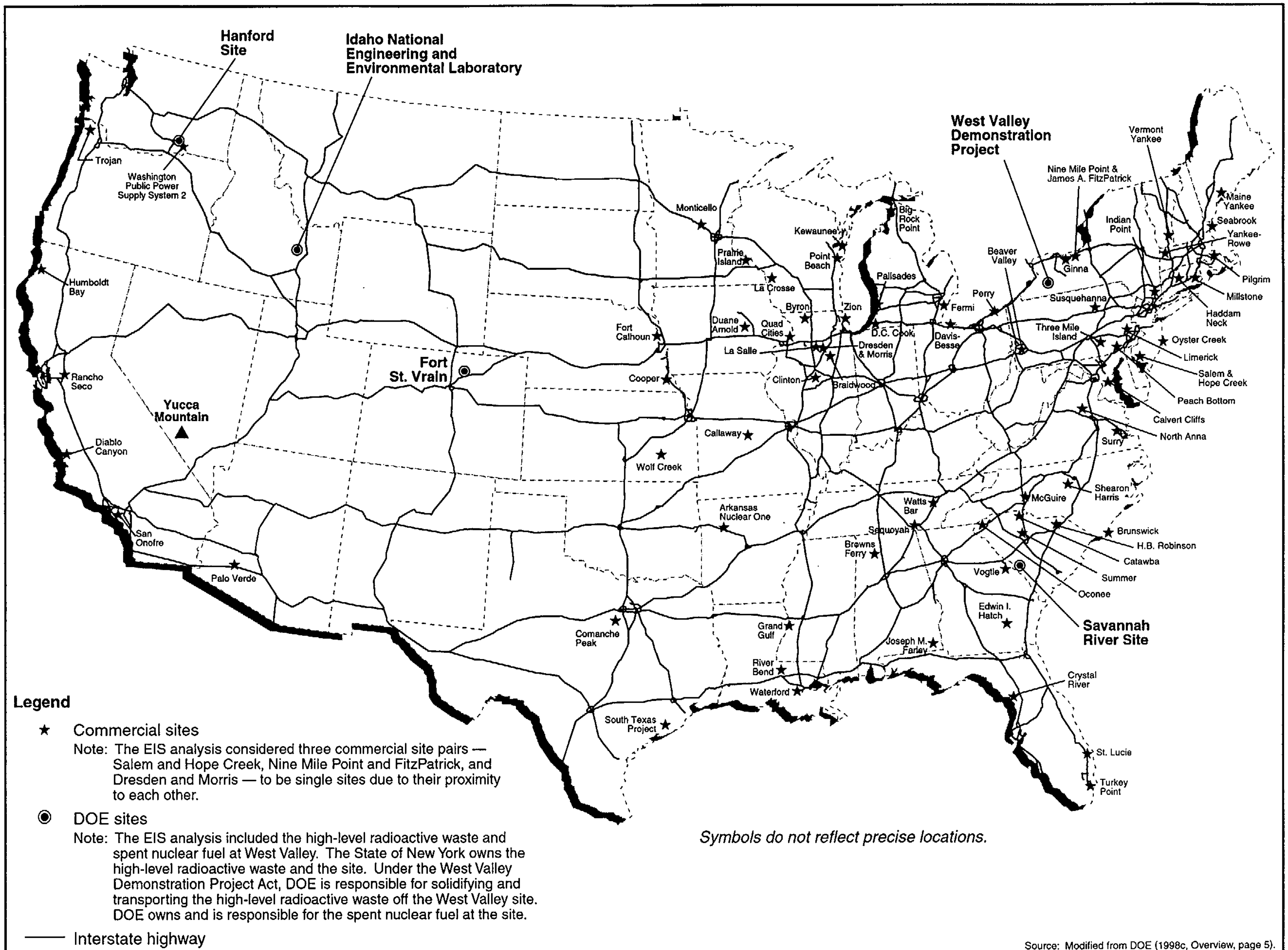
At this time, about 10 years before shipments could begin, DOE has not determined the specific routes it would use to ship spent nuclear fuel and high-level radioactive waste to the proposed repository. Nonetheless, this analysis used current regulations governing highway shipments and historic rail industry practices to select existing highway and rail routes to estimate potential environmental impacts of national transportation. Routing for shipments of spent nuclear fuel and high-level radioactive waste to the proposed repository would comply with applicable regulations of the Department of Transportation and the Nuclear Regulatory Commission in effect at the time the shipments occurred, as stated in the proposed DOE revised policy and procedures for implementing Section 180(c) of the Nuclear Waste Policy Act (DOE 1998b, all).

Approximately 4 years before shipments to the proposed repository began, the Office of Civilian Radioactive Waste Management plans to identify the preliminary routes that DOE anticipates using in state and tribal jurisdictions so it can notify governors and tribal leaders of their eligibility for assistance under the provisions of Section 180(c) of the Nuclear Waste Policy Act. DOE has published a revised proposed policy statement that sets forth its revised plan for implementing a program of technical and financial assistance to states and Native American tribes for training public safety officials of appropriate units of local government and tribes through whose jurisdictions the Department plans to transport spent nuclear fuel or high-level radioactive waste (63 *FR* 83, January 2, 1998).

The analysis of impacts of the Proposed Action and Modules 1 and 2 used characteristics of routes that shipments of spent nuclear fuel and high-level radioactive waste could travel from the originating sites listed in Tables J-5 through J-8. Existing routes that could be used were identified for the mostly legal-weight truck and mostly rail transportation scenarios and included the 10 rail and heavy-haul truck implementing alternatives evaluated in the EIS for transportation in Nevada. The route characteristics used were the transportation mode (highway, railroad, or navigable waterway) and, for each of the modes, the total distance between an originating site and the repository. In addition, the analysis estimated the fraction of travel that would occur in rural, suburban, and urban areas for each route. The fraction of travel in each population zone was determined using 1990 census data (see Section J.1.1.2 and J.1.1.3) to identify population-zone impacts for route segments. The highway routes were selected for the analysis using the HIGHWAY computer program and routing requirements of the Department of Transportation for shipments of Highway Route-Controlled Quantities of Radioactive Materials (49 CFR 397.101). Shipments of spent nuclear fuel and high-level radioactive waste would contain Highway Route-Controlled Quantities of Radioactive Materials.

J.1.2.2.1 *Routes Used in the Analysis*

Routes used in the analysis of transportation impacts of the Proposed Action and Inventory Modules 1 and 2 are highways and rail lines that DOE anticipates it could use for legal-weight truck or rail shipments from each origin to Nevada. For rail shipments that would originate at sites not served by railroads, routes used for analysis include highway routes for heavy-haul trucks or barge routes from the sites to railheads. Figures J-5 and J-6 show the Interstate System highways and mainline railroads, respectively, and their relationship to the commercial and DOE sites and Yucca Mountain. Tables J-11 and J-12 list the lengths of trips and the distances of the highway and rail routes, respectively, in rural, suburban, and urban population zones. Sites that would be capable of loading rail casks, but that do not have direct rail access, are listed in Table J-12. The analysis used four ending rail nodes in Nevada (Beowawe, Caliente, Jean, and Apex) to select rail routes from the 77 sites. These rail nodes would be starting points for the rail and heavy-haul truck implementing alternatives analyzed for transportation in Nevada.



F[illegible] **J-5.** Commercial and DOE sites and Yucca Mountain in rel[illegible] the U.S. Interstate Highway System.

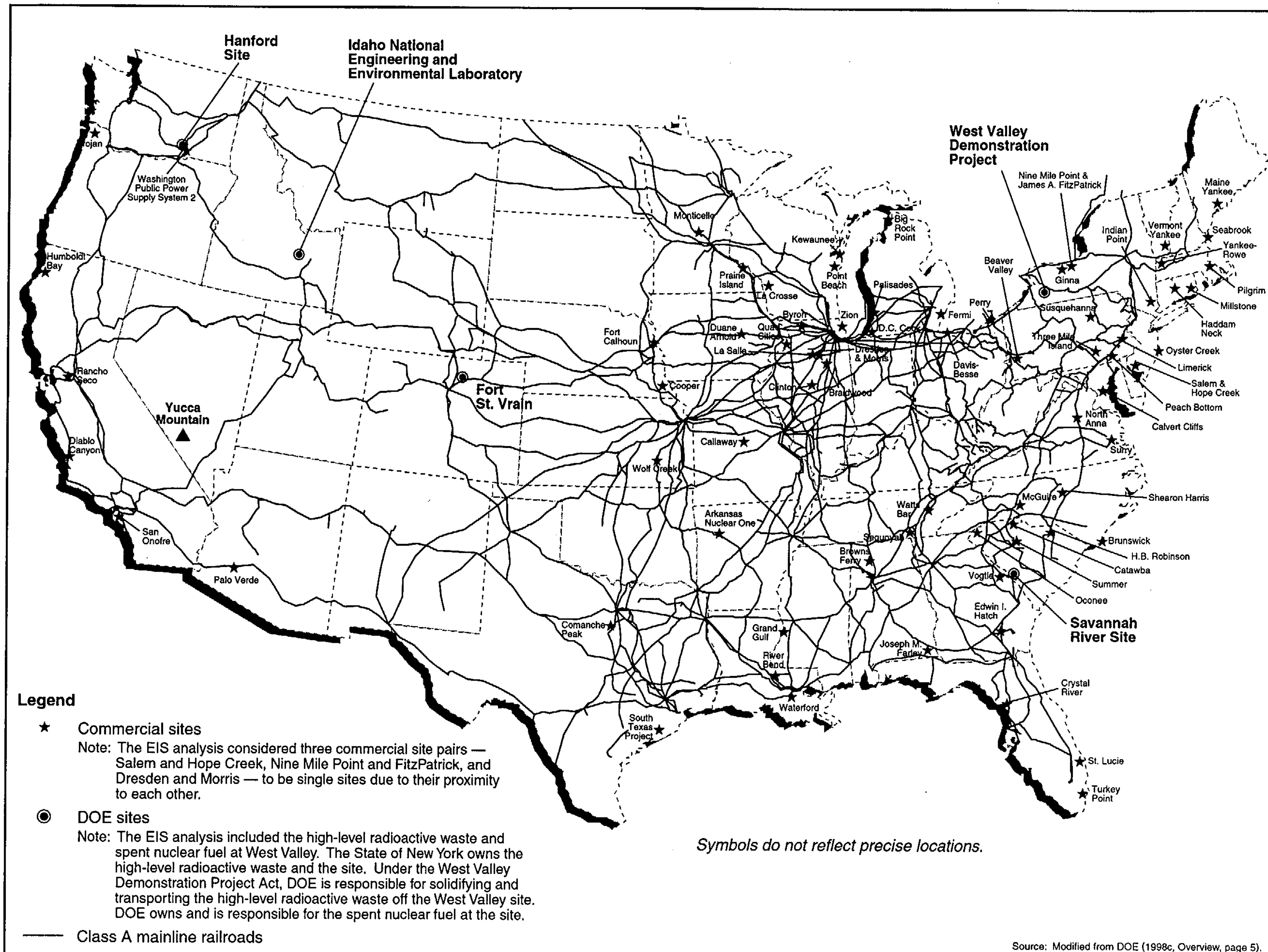


Figure J-6. Commercial and DOE sites and Yucca Mountain in relation to the U.S. railroad system.

Table J-11. Highway distances for legal-weight truck shipments from commercial and DOE sites to Yucca Mountain, mostly legal-weight truck transportation (kilometers)[a,b] (page 1 of 2).

Origin	State	Total[c]	Rural	Suburban	Urban
Browns Ferry	AL	3,442	3,022	374	45
Joseph M. Farley	AL	4,229	3,647	520	62
Arkansas Nuclear One	AR	2,810	2,588	192	30
Palo Verde	AZ	1,007	886	100	21
Diablo Canyon	CA	1,016	828	119	68
Humboldt Bay	CA	1,749	1,465	192	92
Rancho Seco	CA	1,228	1,028	124	76
San Onofre	CA	694	517	89	88
Haddam Neck	CT	4,519	3,708	736	75
Millstone	CT	4,527	3,673	746	109
Crystal River	FL	4,319	3,606	653	59
St. Lucie	FL	4,588	3,793	729	64
Turkey Point	FL	4,842	3,888	821	132
Edwin I. Hatch	GA	3,986	3,373	553	58
Vogtle	GA	3,938	3,301	573	63
Duane Arnold	IA	2,773	2,544	189	40
Braidwood	IL	3,063	2,796	231	36
Byron	IL	3,032	2,773	223	36
Clinton	IL	3,104	2,814	252	38
Dresden/Morris	IL	3,059	2,798	225	36
La Salle	IL	3,017	2,766	215	36
Quad Cities	IL	2,877	2,631	211	36
Zion	IL	3,167	2,834	284	50
Wolf Creek	KS	2,374	2,226	131	16
River Bend	LA	3,446	2,941	420	85
Waterford	LA	3,531	3,003	444	84
Pilgrim	MA	4,722	3,697	930	94
Yankee-Rowe	MA	4,616	3,692	831	92
Calvert Cliffs	MD	4,278	3,511	684	82
Maine Yankee	ME	4,894	3,733	1,052	108
Big Rock Point	MI	3,866	3,266	547	52
D. C. Cook	MI	3,196	2,827	319	51
Fermi	MI	3,524	3,014	449	61
Palisades	MI	3,244	2,855	338	51
Monticello	MN	3,003	2,702	261	41
Prairie Island	MN	2,993	2,720	233	41
Callaway	MO	2,633	2,399	206	27
Grand Gulf	MS	3,354	2,989	311	54
Brunswick	NC	4,418	3,672	680	66
Shearon Harris	NC	4,187	3,493	630	63
McGuire	NC	3,991	3,415	516	58
Cooper Station	NE	2,523	2,328	160	36
Fort Calhoun	NE	2,348	2,165	148	35
Seabrook	NH	4,725	3,676	942	107
Oyster Creek	NJ	4,424	3,530	825	69
Salem/Hope Creek	NJ	4,350	3,531	739	79
Ginna	NY	4,089	3,357	642	91
Indian Point	NY	4,382	3,695	620	67
James FitzPatrick/Nine Mile Point	NY	4,234	3,461	688	85

Table J-11. Highway distances for legal-weight truck shipments from commercial and DOE sites to Yucca Mountain, mostly legal-weight truck transportation (kilometers)[a,b] (page 2 of 2).

Origin	State	Total[c]	Rural	Suburban	Urban
Davis-Besse	OH	3,520	3,106	358	56
Perry	OH	3,693	3,157	464	73
Trojan	OR	2,137	1,865	237	36
Beaver Valley	PA	3,779	3,215	500	64
Limerick	PA	4,287	3,484	741	62
Peach Bottom	PA	4,205	3,479	662	64
Susquehanna	PA	4,126	3,539	528	59
Three Mile Island	PA	4,147	3,443	643	60
Catawba	SC	3,994	3,364	575	54
Oconee	SC	3,853	3,264	532	55
H. B. Robinson	SC	4,112	3,417	628	65
Summer	SC	3,996	3,383	557	55
Sequoyah	TN	3,500	3,039	414	45
Watts Bar	TN	3,578	3,138	394	45
Comanche Peak	TX	2,794	2,547	213	34
South Texas	TX	3,011	2,652	295	64
North Anna	VA	4,081	3,503	515	63
Surry	VA	4,255	3,577	610	67
Vermont Yankee	VT	4,616	3,675	847	94
WPPSS[d] 2	WA	1,880	1,669	178	32
Kewaunee	WI	3,347	2,979	314	55
La Crosse	WI	3,014	2,773	198	43
Point Beach	WI	3,341	2,972	314	55
Ft. St. Vrain[e]	CO	1,415	1,311	93	10
INEEL[f]	ID	1,201	1,044	130	27
West Valley[g]	NY	3,959	3,322	562	75
Savannah River[f]	SC	3,961	3,321	574	64
Hanford[g]	WA	1,881	1,671	178	32

a. To convert kilometers to miles, multiply by 0.62137.
b. Distances determined for purposes of analysis using HIGHWAY computer program.
c. Totals might differ from sums due to method of calculation and rounding.
d. DOE spent nuclear fuel site.
e. DOE spent nuclear fuel and high-level waste site.
f. DOE high-level waste site.
g. WPPSS = Washington Public Power Supply System.

STATE-DESIGNATED PREFERRED ROUTES

Department of Transportation regulations specify that states and tribes can designate preferred routes that are alternatives, or in addition to, Interstate System highways including bypasses or beltways for the transportation of Highway Route-Controlled Quantities of Radioactive Materials. Highway Route-Controlled Quantities of Radioactive Materials include spent nuclear fuel and high-level radioactive waste in quantities that would be shipped on a truck or railcar to the repository. If a state or tribe designated such a route, shipments of spent nuclear fuel and high-level radioactive waste would use the preferred route if (1) it was an alternative preferred route, (2) it would result in reduced time in transit, or (3) it would replace pickup or delivery routes. Ten states—Alabama, Arkansas, California, Colorado, Iowa, Kentucky, Nebraska, New Mexico, Tennessee, and Virginia—have designated alternative or additional preferred routes (Rodgers 1998, all). Although Nevada has designated a State routing agency to the Department of Transportation (Nevada Revised Statutes, Chapter 408.141), the State has not designated alternative preferred routes for Highway Route-Controlled Quantities of Radioactive Materials.

Table J-12. Rail transportation distances from commercial and DOE sites to Nevada ending rail nodes[a] (kilometers)[b,c] (page 1 of 5)

Site	State	Destination	Total[d]	Rural	Suburban	Urban
Commercial sites with direct rail access						
Joseph M. Farley	AL	Apex	4,495	3,872	562	60
		Caliente	4,322	3,698	562	60
		Beowawe	4,177	3,593	535	48
		Jean	4,577	3,937	574	65
Arkansas Nuclear One	AR	Apex	3,170	2,960	181	29
		Caliente	2,996	2,786	181	29
		Beowawe	2,852	2,681	154	17
		Jean	3,251	3,024	193	34
Palo Verde	AZ	Apex	976	864	89	23
		Caliente	1,149	1,038	89	23
		Beowawe	1,908	1,524	274	109
		Jean	894	800	77	18
Rancho Seco	CA	Apex	985	781	151	53
		Caliente	1,159	955	151	53
		Beowawe	706	589	83	32
		Jean	904	717	139	48
San Onofre	CA	Apex	576	409	105	63
		Caliente	750	582	105	63
		Beowawe	1,576	1,167	286	121
		Jean	495	344	93	58
Millstone	CT	Apex	4,728	3,526	994	208
		Caliente	4,555	3,353	994	208
		Beowawe	4,411	3,247	966	197
		Jean	4,810	3,591	1,005	213
Edwin I. Hatch	GA	Apex	4,403	3,830	514	58
		Caliente	4,229	3,656	514	58
		Beowawe	4,085	3,551	486	47
		Jean	4,484	3,894	525	64
Vogtle	GA	Apex	4,459	3,877	523	58
		Caliente	4,286	3,703	523	58
		Beowawe	4,141	3,598	495	47
		Jean	4,541	3,942	534	64
Duane Arnold	IA	Apex	2,745	2,547	167	31
		Caliente	2,572	2,374	167	31
		Beowawe	2,428	2,268	140	20
		Jean	2,827	2,612	178	36
Braidwood	IL	Apex	3,166	2,798	284	85
		Caliente	2,993	2,624	285	85
		Beowawe	2,849	2,518	257	73
		Jean	3,248	2,862	296	90
Byron	IL	Apex	2,979	2,740	205	35
		Caliente	2,806	2,566	205	35
		Beowawe	2,662	2,461	177	24
		Jean	3,061	2,805	216	41
Clinton	IL	Apex	3,172	2,891	228	53
		Caliente	2,998	2,718	228	53
		Beowawe	2,854	2,612	201	42
		Jean	3,253	2,956	239	58
Dresden/Morris	IL	Apex	3,087	2,786	255	46
		Caliente	2,914	2,613	255	46
		Beowawe	2,769	2,507	227	35
		Jean	3,169	2,851	266	51
La Salle	IL	Apex	3,060	2,831	196	33
		Caliente	2,887	2,657	196	33
		Beowawe	2,953	2,691	225	37
		Jean	3,403	3,201	181	20
Quad Cities	IL	Apex	3,003	2,759	210	33
		Caliente	2,829	2,586	210	33
		Beowawe	2,895	2,619	238	38
		Jean	3,345	3,130	195	21

Table J-12. Rail transportation distances from commercial and DOE sites to Nevada ending rail nodes[a] (kilometers)[b,c] (page 2 of 5).

Site	State	Destination	Total[d]	Rural	Suburban	Urban
Commercial sites with direct rail access (continued)						
Zion	IL	Apex	3,119	2,765	279	75
		Caliente	2,946	2,591	279	75
		Beowawe	2,801	2,486	252	64
		Jean	3,201	2,829	291	81
Wolf Creek	KS	Apex	2,685	2,528	131	27
		Caliente	2,512	2,354	131	27
		Beowawe	2,368	2,249	103	16
		Jean	2,767	2,593	142	32
River Bend	LA	Apex	3,509	3,114	322	73
		Caliente	3,380	2,944	377	59
		Beowawe	3,445	2,975	406	65
		Jean	3,428	3,049	311	68
Waterford	LA	Apex	3,551	3,173	304	74
		Caliente	3,423	3,003	359	61
		Beowawe	3,487	3,033	388	66
		Jean	3,470	3,108	293	69
Yankee-Rowe	MA	Apex	4,471	3,466	823	183
		Caliente	4,298	3,292	823	183
		Beowawe	4,153	3,187	796	171
		Jean	4,553	3,530	835	188
Maine Yankee	ME	Apex	4,908	3,629	1,075	204
		Caliente	4,734	3,455	1,075	204
		Beowawe	4,590	3,350	1,048	193
		Jean	4,989	3,693	1,087	209
Big Rock Point	MI	Apex	3,835	3,299	431	105
		Caliente	3,662	3,126	431	105
		Beowawe	3,517	3,020	404	93
		Jean	3,917	3,364	443	110
D. C. Cook	MI	Apex	3,209	2,799	324	86
		Caliente	3,035	2,625	324	86
		Beowawe	2,891	2,520	297	75
		Jean	3,290	2,863	336	91
Fermi	MI	Apex	3,649	3,046	469	135
		Caliente	3,476	2,872	469	135
		Beowawe	3,332	2,767	442	123
		Jean	3,731	3,110	481	140
Prairie Island	MN	Apex	2,980	2,715	238	28
		Caliente	2,807	2,541	238	28
		Beowawe	2,663	2,436	210	16
		Jean	3,062	2,780	249	33
Brunswick	NC	Apex	4,768	3,972	724	71
		Caliente	4,594	3,799	724	71
		Beowawe	4,450	3,693	697	59
		Jean	4,849	4,037	736	76
Shearon Harris	NC	Apex	4,669	3,910	689	69
		Caliente	4,495	3,737	689	69
		Beowawe	4,351	3,631	662	58
		Jean	4,751	3,975	701	75
McGuire	NC	Apex	4,539	3,779	683	77
		Caliente	4,366	3,605	683	77
		Beowawe	4,221	3,500	656	65
		Jean	4,621	3,844	694	82
Seabrook	NH	Apex	4,755	3,567	987	201
		Caliente	4,582	3,393	987	201
		Beowawe	4,437	3,288	960	190
		Jean	4,837	3,632	999	206
FitzPatrick/Nine Mile Point	NY	Apex	4,213	3,296	728	188
		Caliente	4,039	3,123	728	188
		Beowawe	3,895	3,017	701	177
		Jean	4,294	3,361	740	193

Table J-12. Rail transportation distances from commercial and DOE sites to Nevada ending rail nodes[a] (kilometers)[b,c] (page 3 of 5).

Site	State	Destination	Total[d]	Rural	Suburban	Urban
Commercial sites with direct rail access (continued)						
Davis Besse	OH	Apex	3,590	3,133	342	114
		Caliente	3,416	2,960	342	114
		Beowawe	3,272	2,854	315	103
		Jean	3,671	3,198	354	120
Perry	OH	Apex	3,692	3,131	416	145
		Caliente	3,519	2,958	416	145
		Beowawe	3,374	2,852	389	133
		Jean	3,774	3,196	428	150
Trojan	OR	Apex	2,202	1,897	244	61
		Caliente	2,031	1,871	136	23
		Beowawe	1,539	1,445	85	9
		Jean	2,121	1,833	233	56
Beaver Valley	PA	Apex	3,819	3,212	499	108
		Caliente	3,645	3,039	499	108
		Beowawe	3,501	2,933	472	96
		Jean	3,901	3,277	510	113
Limerick	PA	Apex	4,389	3,349	843	197
		Caliente	4,216	3,175	843	197
		Beowawe	4,072	3,070	816	186
		Jean	4,471	3,414	855	203
Susquehanna	PA	Apex	4,406	3,412	819	175
		Caliente	4,232	3,238	819	175
		Beowawe	4,088	3,133	791	164
		Jean	4,487	3,477	830	180
Three Mile Island	PA	Apex	4,283	3,330	767	186
		Caliente	4,110	3,157	767	186
		Beowawe	3,966	3,051	739	175
		Jean	4,365	3,395	778	191
Catawba	SC	Apex	4,537	3,756	702	77
		Caliente	4,363	3,583	702	77
		Beowawe	4,219	3,477	675	66
		Jean	4,618	3,821	714	82
H. B. Robinson	SC	Apex	4,513	3,745	688	78
		Caliente	4,339	3,572	688	78
		Beowawe	4,195	3,466	661	67
		Jean	4,594	3,810	700	83
Summer	SC	Apex	4,472	3,782	621	68
		Caliente	4,299	3,609	621	68
		Beowawe	4,154	3,503	594	57
		Jean	4,554	3,847	633	74
Sequoyah	TN	Apex	3,890	3,480	361	48
		Caliente	3,716	3,307	361	48
		Beowawe	3,572	3,201	333	37
		Jean	3,971	3,545	372	53
Watts Bar	TN	Apex	3,887	3,544	286	57
		Caliente	3,714	3,370	286	57
		Beowawe	3,569	3,265	259	46
		Jean	3,969	3,608	298	62
Comanche Peak	TX	Apex	2,890	2,639	213	38
		Caliente	2,716	2,465	213	38
		Beowawe	2,791	2,512	236	43
		Jean	2,445	2,338	101	5
South Texas	TX	Apex	3,055	2,800	206	49
		Caliente	3,228	2,973	206	49
		Beowawe	3,320	2,948	330	43
		Jean	2,973	2,735	194	44
North Anna	VA	Apex	4,521	3,669	686	165
		Caliente	4,347	3,496	686	165
		Beowawe	4,203	3,390	659	153
		Jean	4,602	3,734	698	170

Table J-12. Rail transportation distances from commercial and DOE sites to Nevada ending rail nodes[a] (kilometers)[b,c] (page 4 of 5).

Site	State	Destination	Total[d]	Rural	Suburban	Urban
Commercial sites with direct rail access (continued)						
Vermont Yankee	VT	Apex	4,551	3,519	846	186
		Caliente	4,378	3,345	846	186
		Beowawe	4,233	3,240	818	175
		Jean	4,633	3,584	857	192
WPPSS[j] 2	WA	Apex	1,946	1,807	116	22
		Caliente	1,772	1,634	116	22
		Beowawe	1,565	1,490	66	9
		Jean	2,027	1,872	128	28
Commercial sites with indirect rail access						
Browns Ferry HH – 55.4 kilometers	AL	Apex	3,741	3,332	357	52
		Caliente	3,567	3,158	357	52
		Beowawe	3,423	3,053	329	41
		Jean	3,822	3,397	368	57
Diablo Canyon HH – 43.5 kilometers	CA	Apex	893	609	174	110
		Caliente	1,067	783	174	110
		Beowawe	1,157	872	203	82
		Jean	812	544	162	105
St. Lucie HH – 23.3 kilometers	FL	Apex	4,938	4,073	780	85
		Caliente	4,765	3,899	780	85
		Beowawe	4,621	3,794	753	73
		Jean	4,863	4,006	732	125
Turkey Point HH – 17.4 kilometers	FL	Apex	5,285	4,305	841	138
		Caliente	5,111	4,132	841	138
		Beowawe	4,967	4,026	814	126
		Jean	5,366	4,370	853	143
Calvert Cliffs HH – 41.9 kilometers	MD	Apex	4,543	3,448	881	213
		Caliente	4,369	3,275	881	213
		Beowawe	4,225	3,169	854	201
		Jean	4,625	3,513	893	218
Palisades HH – 41.9 kilometers	MI	Apex	3,257	2,816	353	88
		Caliente	3,083	2,642	353	88
		Beowawe	2,939	2,537	326	77
		Jean	3,339	2,881	365	93
Callaway HH – 18.5 kilometers	MO	Apex	2,807	2,636	140	32
		Caliente	2,634	2,462	140	32
		Beowawe	2,490	2,357	113	20
		Jean	2,889	2,701	151	37
Grand Gulf HH – 47.8 kilometers	MS	Apex	3,686	3,355	291	39
		Caliente	3,512	3,181	291	39
		Beowawe	3,368	3,076	264	28
		Jean	3,767	3,419	303	44
Cooper Station HH – 53.8 kilometers	NE	Apex	2,429	2,252	141	36
		Caliente	2,256	2,078	141	36
		Beowawe	2,111	1,973	114	25
		Jean	2,511	2,317	153	42
Fort Calhoun HH – 6.0 kilometers	NE	Apex	2,313	2,189	102	21
		Caliente	2,139	2,015	102	21
		Beowawe	1,995	1,910	75	10
		Jean	2,394	2,254	114	27
Salem/Hope Creek HH – 51.0 kilometers	NJ	Apex	4,551	3,375	946	229
		Caliente	4,378	3,202	946	229
		Beowawe	4,234	3,097	919	218
		Jean	4,633	3,440	958	235
Oyster Creek HH – 28.5 kilometers	NJ	Apex	4,568	3,395	952	221
		Caliente	4,395	3,222	952	221
		Beowawe	4,251	3,116	925	209
		Jean	4,650	3,460	964	226

Table J-12. Rail transportation distances from commercial and DOE sites to Nevada ending rail nodes[a] (kilometers)[b,c] (page 5 of 5).

Site	State	Destination	Total[d]	Rural	Suburban	Urban
Commercial sites with indirect rail access (continued)						
Peach Bottom	PA	Apex	4,304	3,335	778	190
HH – 58.9 kilometers		Caliente	4,131	3,161	778	190
		Beowawe	3,986	3,056	751	179
		Jean	4,386	3,400	790	196
Oconee	SC	Apex	4,257	3,662	534	61
HH – 17.5 kilometers		Caliente	4,084	3,488	534	61
		Beowawe	3,940	3,383	507	50
		Jean	4,339	3,726	545	66
Surry	VA	Apex	4,505	3,927	512	66
HH – 75.2 kilometers		Caliente	4,332	3,753	512	66
		Beowawe	4,188	3,648	484	55
		Jean	4,587	3,992	523	72
Kewaunee	WI	Apex	3,444	2,954	395	95
HH – 9.7 kilometers		Caliente	3,270	2,780	395	95
		Beowawe	3,126	2,675	368	84
		Jean	3,526	3,019	406	100
Point Beach	WI	Apex	3,397	2,938	370	89
HH – 36.4 kilometers		Caliente	3,224	2,765	370	89
		Beowawe	3,080	2,659	343	78
		Jean	3,479	3,003	381	94
DOE spent nuclear fuel and high-level waste (direct rail access)						
Ft. St. Vrain[g]	CO	Apex	1,561	1,453	93	14
		Caliente	1,387	1,280	93	14
		Beowawe	1,298	1,266	29	3
		Jean	1,643	1,518	105	20
INEEL[h]	ID	Apex	1,059	978	66	15
		Caliente	885	804	66	15
		Beowawe	741	699	39	4
		Jean	1,140	1,042	78	21
West Valley[i]	NY	Apex	3,972	3,169	638	165
		Caliente	3,798	2,995	638	165
		Beowawe	3,654	2,890	611	153
		Jean	4,053	3,234	650	170
Savannah River Site[h]	SC	Apex	4,374	3,690	609	75
		Caliente	4,201	3,517	609	75
		Beowawe	4,057	3,411	581	64
		Jean	4,456	3,755	620	80
Hanford Site[h]	WA	Apex	1,933	1,795	116	22
		Caliente	1,760	1,622	116	22
		Beowawe	1,553	1,477	66	9
		Jean	2,015	1,860	128	28

a. The ending rail nodes (INTERLINE computer program designations) are Apex-14763; Caliente-14770; Beowawe-14791; and Jean-16328.
b. To convert kilometers to miles, multiply by 0.62137.
c. This analysis used the INTERLINE computer program to estimate distances.
d. Totals might differ from sums due to method of calculation and rounding.
e. NP = nuclear plant.
f. DOE spent nuclear fuel.
g. DOE spent nuclear fuel and high-level radioactive waste.
h. DOE high-level radioactive waste.
i. WPPSS = Washington Public Power Supply System.

Selection of Highway Routes. The analysis of national transportation impacts used route characteristics of existing highways, such as distances, population densities, and state-level accident statistics. The analysis of highway shipments of spent nuclear fuel and high-level radioactive waste used the HIGHWAY computer model (Johnson et al. 1993a, all) to determine highway routes using regulations of the Department of Transportation (49 CFR 397.101) that specify how routes are selected. The selection of "preferred routes" is required for shipment of these materials. DOE has determined that the HIGHWAY program is appropriate for calculating highway routes and related information (Maheras and

Pippen 1995, pages 2 to 5). HIGHWAY is a routing tool that DOE has used in previous EISs [for example, the programmatic EIS on spent nuclear fuel (DOE 1995, page I-6) and the Waste Isolation Pilot Plant Supplement II EIS (DOE 1997a, pages 5 to 13)] to determine highway routes for impact analysis.

Because the regulations require that the preferred routes result in reduced time in transit, changing conditions, weather, and other factors could result in the use of more than one route at different times for shipments between the same origin and destination. However, for this analysis the program selected only one route for travel from each site to the Yucca Mountain site.

Although shipments could use more than one preferred route in national highway transportation to comply with Department of Transportation regulations (49 CFR 397.101), under current Department of Transportation regulations all preferred routes would ultimately enter Nevada on Interstate 15 and travel to the repository on U.S. Highway 95. States can designate alternative or additional preferred routes for highway shipments (49 CFR 397.103). At this time the State of Nevada has not identified any alternative or additional preferred routes that DOE could use for shipments to the repository.

Selection of Rail Routes. Rail transportation routing of spent nuclear fuel and high-level radioactive waste shipments is not regulated by the Department of Transportation. As a consequence, the routing rules used by the INTERLINE computer program (Johnson et al. 1993b, all) assumed that railroads would select routes using historic practices. DOE has determined that the INTERLINE program is appropriate for calculating routes and related information for use in transportation analyses (Maheras and Pippen 1995, pages 2 to 5). Because the routing of rail shipments would be subject to future, possibly different practices of the involved railroads, DOE could use other rail routes.

For the 19 commercial sites that have the capability to handle and load rail casks but do not have direct rail service, DOE used the HIGHWAY computer program to identify routes for heavy-haul transportation to nearby railheads. For such routes, routing agencies in affected states would need to approve the transport and routing of overweight and overdimensional shipments.

J.1.2.2.2 *Routes for Shipping Rail Casks from Sites Not Served by a Railroad*

In addition to routes for legal-weight trucks and rail shipments, 19 commercial sites that are not served by a railroad, but that have the capability to load rail casks, could ship spent nuclear fuel to nearby railheads using heavy-haul trucks (see Table J-12). Fourteen of these sites are on navigable waterways; some of these could ship by barge to railheads. Distances to the nearest railheads for barge shipments were estimated for each of the 14 reactor sites. These distances are listed in Table J-13.

J.1.2.2.3 *Sensitivity of Analysis Results to Routing Assumptions*

Routing for shipments of spent nuclear fuel and high-level radioactive waste to the proposed repository would comply with regulations of the Department of Transportation and the Nuclear Regulatory Commission in effect at the time shipments would occur. Unless the State of Nevada designates alternative or additional preferred routes, to comply with Department of Transportation regulations all preferred routes would ultimately enter Nevada on Interstate 15 and travel to the repository on U.S. Highway 95. States can designate alternative or additional preferred routes for highway shipments. At this time the State of Nevada has not identified any alternative or additional preferred routes DOE could use for shipments to the repository. Section J.3.1.3 examines the sensitivity of transportation impacts both nationally and regionally (within Nevada) to changes in routing assumption within Nevada.

Table J-13. Barge transportation distances from sites to intermodal rail nodes (kilometers).[a,b]

Site	State	Total[d]	Rural	Suburban	Urban
Browns Ferry	AL	57	52	5	0
Diablo Canyon	CA	143	143	0	0
St. Lucie	FL	140	50	52	39
Turkey Point	FL	54	53	0	1
Calvert Cliffs	MD	99	98	2	0
Palisades	MI	256	256	0	0
Grand Gulf	MS	51	51	0	0
Cooper	NE	117	100	16	1
Salem/Hope Creek	NJ	30	30	0	0
Oyster Creek	NJ	130	77	36	17
Surry	VA	71	60	8	3
Kewaunee	WI	293	285	2	7
Point Beach	WI	301	293	2	7

a. To convert kilometers to miles, multiply by 0.62137.
b. Distances estimated with INTERLINE (Johnson et al. 1993b, all).
c. Intermodal rail nodes selected for purpose of analysis. Source: TRW (1999a, Section 4).
d. Totals might differ from sums due to methods of calculation and rounding.

J.1.3 ANALYSIS OF IMPACTS FROM INCIDENT-FREE TRANSPORTATION

DOE analyzed the impacts of incident-free transportation for shipments of commercial and DOE spent nuclear fuel and DOE high-level radioactive waste that would be shipped under the Proposed Action and Inventory Modules 1 and 2 from 77 sites to the repository. The analysis estimated impacts to the public and workers and included impacts of loading shipping casks at commercial and DOE sites and other preparations for shipment as well as intermodal transfers of casks from heavy-haul trucks or barges to rail cars.

J.1.3.1 Methods and Approach for Analysis of Impacts for Loading Operations

The analysis used methods and assessments developed for spent nuclear fuel loading operations at commercial sites to estimate radiological impacts to involved workers at commercial and DOE sites. Previously developed conceptual radiation shield designs for shipping casks (Schneider et al. 1987, Sections 4 and 5), rail and truck shipping cask dimensions, and estimated radiation dose rates at locations where workers would load and prepare casks (Smith, Daling, and Faletti 1992, page 4.2) for shipment were the analysis bases for loading operations. In addition, tasks and time-motion evaluations from these studies were used to describe spent nuclear fuel handling and loading. These earlier evaluations were based on normal, incident-free operations that would be conducted according to Nuclear Regulatory Commission regulations that establish radiation protection criteria for workers.

The analysis assumed that noninvolved workers would not have tasks that would result in radiation exposure. In a similar manner, the analysis projected that the dose to the public from loading operations would be extremely small, resulting in no or small impacts. A separate evaluation of the potential radiation dose to members of the public from loading operations at commercial nuclear reactor facilities showed that the dose would be very low, less than 0.001 person-rem per metric ton uranium of spent nuclear fuel loaded (DOE 1986, page 2.42, Figure 2.9). Public doses from activities at commercial and DOE sites generally come from exposure to airborne emissions and, in some cases, waterborne effluents containing low levels of radionuclides. However, direct radiation at publicly accessible locations near these sites typically is not measurable and contributes negligibly to public dose and radiological impacts. Though DOE expects no releases from loading operations, this analysis estimated that the dose to the public would be 0.001 person-rem per metric ton uranium, and metric ton equivalents, for DOE spent nuclear fuel and high-level radioactive waste. Noninvolved workers could also be exposed to low levels of radioactive materials and radioactivity from loadout operations. However, because these workers would not work in radiation areas they would receive a very small fraction of the dose received by involved workers. DOE anticipates that noninvolved workers would receive individual doses similar to those received by members of the public. Because the population of noninvolved workers would be small compared to the population of the general public near the 77 sites, the dose to these workers would be a small fraction of the public dose.

The analysis used several basic assumptions to evaluate impacts from loading operations at DOE sites:

- Operations to load spent nuclear fuel and high-level radioactive waste at DOE facilities would be similar to loading operations at commercial facilities.

- Commercial spent nuclear fuel would be in storage pools or in dry storage at the reactors and DOE spent nuclear fuel would be in dry storage, ready to be loaded directly in Nuclear Regulatory Commission-certified shipping casks and then on transportation vehicles. In addition, DOE high-level radioactive waste could be loaded directly in casks. All preparatory activities, including packaging, repackaging, and validating the acceptability of spent nuclear fuel for acceptance at the repository would be complete prior to loading operations.

- Commercial spent nuclear fuel to be placed in the shipping casks would be uncanistered or canistered fuel assemblies, with at least one assembly in a canister. DOE spent nuclear fuel and high-level radioactive waste would be in disposable canisters. Typically, uncanistered assemblies would be loaded into shipping casks under water in storage pools (wet storage). Canistered spent nuclear fuel could be loaded in casks directly from dry storage facilities or storage pools.

In addition, because handling and loading operations for DOE spent nuclear fuel and high-level radioactive waste and commercial spent nuclear fuel would be similar, the analysis assumed that impacts to workers during the loading of commercial spent nuclear fuel could represent those for the DOE materials, even though the radionuclide inventory of commercial fuel and the resultant external dose rate would be higher than those of the DOE materials. This conservative assumption of selecting impacts from commercial handling and loading operations overestimated the impacts of DOE loading operations, but it enabled the use of detailed real information developed for commercial loading operations to assess impacts for DOE operations. Equivalent information was not available for operations at DOE facilities. To gauge the conservatism of the assumption DOE compared the radioactivity of contents of shipments of commercial and DOE spent nuclear fuel and high-level radioactive waste. Table J-14 compares typical inventories of important contributors to the assessment of worker and public health impacts. These are cesium-137 and actinide isotopes (including plutonium) for rail shipments of commercial spent nuclear fuel, DOE spent nuclear fuel, and DOE high-level radioactive waste. Although other factors are also important (for example, material form and composition), these indicators provide an index of the relative hazard potential of the materials. Appendix A contains additional information on the radionuclide inventory and characteristics of spent nuclear fuel and high-level radioactive waste.

J.1.3.1.1 *Radiological Impacts of Loading Operations at Commercial Sites*

In 1987, DOE published a study of the estimated radiation doses to the public and workers resulting from the transport of spent nuclear fuel from commercial nuclear power reactors to a hypothetical deep geologic repository (Schneider et al. 1987, all). This study was based on a single set of spent nuclear fuel characteristics and a single split [30 percent/70 percent by weight; 900 metric tons uranium/2,100 metric tons uranium per year] between truck and rail conveyances. DOE published its findings on additional radiological impacts on monitored retrievable storage workers in an addendum to the 1987 report (Smith, Daling, and Faletti 1992, all). The technical approaches and impacts summarized in these DOE reports

Table J-14. Typical cesium-137, actinide isotope, and total radioactive material content (curies) in a rail shipping cask.[a]

Material	Cesium-137	Actinides (excluding uranium)[b]	Total
Commercial spent nuclear fuel	810,000	650,000	2,000,000
High-level radioactive waste	120,000	40,000[c]	280,000
DOE spent nuclear fuel (except naval spent nuclear fuel)	260,000	160,000	620,000
Naval spent nuclear fuel	550,000	30,000	1,200,000

a. Source: Appendix A. Source estimated based on 36 typical pressurized-water reactor fuel assemblies for commercial spent nuclear fuel; one dual-purpose shipping canister for naval spent fuel; five canisters of DOE spent nuclear fuel; and five canisters of high-level radioactive waste.
b. Uranium would not be an important contributor to health and safety risk.
c. Includes plutonium can-in-canister with high-level radioactive waste.

were used to project involved worker impacts that would result from commercial at-reactor spent nuclear fuel loading operations. DOE did not provide a separate analysis of noninvolved worker impacts in these reports. For the analysis in this EIS, DOE assumed that noninvolved workers would not receive radiation exposures from loading operations. This assumption is appropriate because noninvolved workers would be personnel with managerial or administrative support functions directly related to the loading tasks but at locations, typically in offices, away from areas where loading activities took place.

In the DOE study, worker impacts from loading operations were estimated for a light-water reactor with pool storage of spent nuclear fuel. The radiological characteristics of the spent nuclear fuel in the analysis was 10-year-old, pressurized-water reactor fuel with an exposure history (burnup) of 35,000 megawatt-days per metric ton. In addition, the reference pressurized-water reactor and boiling-water reactor fuel assemblies were assumed to contain 0.46 and 0.19 MTU, respectively, prior to reactor irradiation. These parameters for spent nuclear fuel are similar to those presented in Appendix A of this EIS. The use of the parameters for spent nuclear fuel presented in Appendix A would be likely to lead to similar results.

In the 1987 study, radiation shielding analyses were done to provide information on (1) the conceptual configuration of postulated reference rail and truck transportation casks, and (2) the direct radiation levels at accessible locations near loaded transportation casks. The study also presented the results of a detailed time-motion analysis of work tasks that used a loading concept of operations. This task analysis was coupled with cask and at-reactor direct radiation exposure rates to estimate radiation doses to involved workers (that is, those who would participate directly in the handling and loading of the transportation casks and conveyances). Impacts to members of the public from loading operations had been shown to be small [fraction of a person-millirem population dose; (Schneider et al. 1987, page 2.9)] and were eliminated from further analysis in the 1987 report. The at-reactor-loading concept of operations included the following activities:

1. Receiving the empty transportation cask at the site fence

2. Preparing and moving the cask into the facility loading area

3. Removing the cask from the site prime mover trailer

4. Preparing the cask for loading and placing it in the water-filled loading pit

5. Transferring spent nuclear fuel from its pool storage location to the cask

6. Removing the cask from the pool and preparing it for shipment

7. Placing the cask on the site prime mover trailer

8. Moving the loaded cask to the site fence where the trailer is connected to the transportation carrier's prime mover for offsite shipment

The results for loading operations are listed in Table J-15.

Table J-15. Principal logistics bases and results for the reference at-reactor loading operations.[a]

Parameter	Conveyance		
	Rail[b]	Truck[c]	Total
Annual loading rate (MTU/year)[d]	2,100	900	3,000
Transportation cask capacity, PWR - BWR (MTU/cask)	6.5/6.70	0.92/0.93	NA[e]
Annual shipment rate (shipments/year)	320	970	1,290
Average loading duration,[f] PWR - BWR (days)	2.3/2.5	1.3/1.4	NA
Involved worker specific CD,[g] PWR - BWR (person-rem/MTU)	0.06/0.077	0.29/0.31	NA

a. Source: Schneider et al. (1987, pages 2.5 and 2.7).
b. 14 pressurized-waste reactor and boiling-water reactor spent nuclear fuel assemblies per rail transportation cask.
c. 2 pressurized-waste reactor and boiling-water reactor spent nuclear fuel assemblies per truck transportation cask.
d. MTU = metric tons of uranium.
e. NA = not applicable.
f. Based on single shift operations; carrier drop-off and pick-up delays were not included.
g. Collective dose expressed as the sum of the doses accumulated by all loading (involved) workers, regardless of the total number of workers assigned to loading tasks.

The loading activities that the study determined would produce the highest collective unit impacts are listed in Table J-16. As listed in this table, the involved worker collective radiation doses would be dominated by tasks in which the workers would be near the transportation cask when it contained spent nuclear fuel, particularly when they were working around the cask lid area. These activities would deliver at least 40 percent of the total collective worker doses. Worker impacts from the next largest dose-producing tasks (working to secure the transportation cask on the trailer) would account for 12 to 19 percent of the total impact. The impacts are based on using crews of 13 workers [the number of workers assumed in the Schneider et al. (1987, Section 2) study] dedicated solely to performing cask-handling work. The involved worker collective dose was calculated using the following formula:

$$\text{Collective dose (person-rem)} = A \times B \times C \times D \times E$$

where: A = number of pressurized-water or boiling-water reactor spent nuclear fuel shipments being analyzed under each transportation scenario (from Tables J-5 and J-6)

B = number of transportation casks included in a shipment (set at 1 for both transportation scenarios)

C = number of pressurized-water or boiling-water reactor spent nuclear fuel assemblies in a transportation cask (from Table J-3)

D = amount of uranium in the spent nuclear fuel assembly prior to reactor irradiation, expressed as metric tons uranium per assembly (from Table J-15)

E = involved worker-specific collective dose in person-rem/metric ton uranium for each fuel type (from Table J-15)

Table J-16. At-reactor reference loading operations—collective impacts to involved workers.[a]

Task description	Rail CD/MTU[b] (PWR - BWR)[c]	Rail Percent of total impact	Truck CD/MTU (PWR - BWR)	Truck Percent of total impact
Install cask lids; flush cask interior; drain, dry and seal cask	0.025/0.024	40/31	0.126/0.126	43/40
Install cask binders, impact limiters, personnel barriers	0.010/0.009	15/12	0.056/0.055	19/18
Load SNF into cask	0.011/0.027	17/35	0.011/0.027	4/9
On-vehicle cask radiological decontamination and survey	0.003/0.003	5/4	0.018/0.018	6/6
Final inspection and radiation surveys	0.002/0.002	4/3	0.016/0.015	5/5
All other (19) activities	0.011/0.012	19/16	0.066/0.073	23/23
Task totals	**0.062/0.077**	**100/100**	**0.29/0.31**	**100/100**

a. Source: Schneider et al. (1987, page 2.9).
b. CD/MTU = Collective dose (person-rem effective dose equivalent) per metric ton uranium. The at-reactor loading
c. crew size is 13 involved workers.
d. PWR = pressurized-water reactor; BWR = boiling-water reactor.

Because worker doses are linked directly to the number of loading operations performed, the highest average individual doses under each transportation scenario would occur at the reactor sites having the most number of shipments. Accordingly, the average individual dose impacts were calculated for the limiting site using the equation:

$$\text{Average individual dose (rem per involved worker)} = (A \times B \times C \times D \times E) \div F$$

where: A = largest value for the number of shipments from a site under each transportation scenario (from Tables J-5 and J-6)

B = number of transportation casks included in a shipment (set at 1 for both transportation options)

C = number of spent nuclear fuel assemblies in a transportation cask (from Table J-3)

D = amount of uranium in the spent nuclear fuel assembly prior to reactor irradiation in metric tons uranium per assembly (from Table J-15)

E = involved worker-specific collective dose in person-rem per metric ton uranium for each fuel type (from Table J-15)

F = involved worker crew size (set at 13 persons for both transportation options; from Table J-16)

J.1.3.1.2 *Radiological Impacts of DOE Spent Nuclear Fuel and High-Level Radioactive Waste Loading Operations*

The methodology used to estimate impacts to workers during loading operations for commercial spent nuclear fuel was also used to estimate impacts of loading operations for DOE spent nuclear fuel and high-level radioactive waste. The exposure factor for loading boiling-water reactor spent nuclear fuel in truck casks at commercial facilities (person-rem per MTU) was used (see Table J-16). The exposure factor for truck shipments of boiling-water reactor spent nuclear fuel was based on a cask capacity of five boiling-water reactor spent nuclear fuel assemblies (about 0.9 MTHM). The analysis used this factor because it would result in the largest estimates for dose per operation.

J.1.3.2 Methods and Approach for Analysis of Impacts from Incident-Free Transportation

The potential exists for human health impacts to workers and members of the public from incident-free transportation of spent nuclear fuel and high level radioactive waste. *Incident-free* transportation means normal accident-free shipment operations during which traffic accidents and accidents in which radioactive materials could be released do not occur; these are addressed separately in Section J.1.4. Incident-free impacts could occur from exposure to (1) external radiation in the vicinity of the transportation casks, or (2) transportation vehicle emissions, both during normal transportation.

J.1.3.2.1 *Incident-Free Radiation Dose to Populations*

The analysis used the RADTRAN4 computer program (Neuhauser and Kanipe 1992, all) to evaluate incident-free impacts for populations. The RADTRAN4 input parameters used to estimate incident-free impacts are listed in Table J-17. Through extensive review (Maheras and Pippen 1995, Section 3 and 4), DOE has determined that this program provides valid estimates of population doses for use in the evaluation of risks of transporting radioactive materials, including spent nuclear fuel and high-level radioactive waste. DOE has used the RADTRAN4 code to analyze transportation impacts for other environmental impact statements (for example, DOE 1995, Appendix E; DOE 1997b, Appendixes F and G). The program used population densities from 1990 census data to calculate the collective dose to populations that live along transportation routes [within 800 meters (0.5 mile) of either side of the route]. Table J-18 lists the estimated number of people who live within 800 meters of national routes.

The analysis used five kinds of information to estimate collective doses to populations:

- External radiation dose rate around shipping casks
- Number of people who would live within 800 meters (0.5 mile) along the routes of travel
- Distances individuals would live from the routes
- Amount of time each individual would be exposed as a shipment passed by
- Number of shipments that would be transported over each route

The first four were developed using the data listed in Table J-19. The fifth kind of information (the number of shipments that would use a transportation route) was developed with the use of the CALVIN computer program discussed in Section J.1.1.1, the DOE Throughput Study (TRW 1997, Section 6.1.1), data on DOE spent nuclear fuel and high-level radioactive waste inventories in Appendix A, and data from DOE sites (Jensen 1998, all). The analysis used CALVIN to estimate the number of shipments from each commercial site. The Throughput Study provided the estimated number of shipments of high-level radioactive waste from the four DOE sites. Information provided by the DOE National Spent Nuclear Fuel Program (Jensen 1998, all) and in Appendix A was used to estimate shipments of DOE spent nuclear fuel.

The analysis used a value of 10 millirem per hour at a distance of 2 meters (6.6 feet) from the side of a transport vehicle for the external dose rate around shipping casks. This value is the maximum allowed by regulations of the Department of Transportation for shipments of radioactive materials [49 CFR 173.441(b)]. Dose rates at distances greater than 2 meters from the side of a vehicle would be less. The dose rate at 30 meters (100 feet) from the vehicle would be less than 0.2 millirem per hour; at a distance of 800 meters (2,625 feet) the dose rate would be less than 0.0002 millirem per hour.

Table J-17. Input parameters and parameter values used for the incident-free national truck and rail transportation analysis.

Parameter	Legal-weight truck transportation	Rail transportation	Legal-weight truck and rail
Package type			Type B shipping cask
Package dimension			4.77 meters[a] long
Dose rate			10 millirem per hour, 2 meters from side of vehicle
Number of crewmen	2	5	
Distance from source to crew	3 meters	152 meters	
Speed			
Rural	88 km[b] per hour	64 km per hour	
Suburban			40 km per hour
Urban			24 km per hour
Stop time per km	0.011 hours per km	0.033 hours per km[c]	
Number of people exposed while stopped	50	Based on suburban population density	
Number of people per vehicle sharing route	2	3	
Population densities (persons per km²)[d]			
Rural			(e)
Suburban			(e)
Urban			(e)
One-way traffic count (vehicles per hour)			
Rural	470	1	
Suburban	780	5	
Urban	2,800	5	

a. To convert meters to feet, multiply by 3.2808.
b. To convert kilometers (km) to miles, multiply by 0.62137.
c. Assumes general freight rather than dedicated service.
d. To convert square kilometers to square miles, multiply by 0.3861.
e. Population densities along transportation routes were estimated using the HIGHWAY and INTERLINE computer programs. These programs used 1990 Census data.

Table J-18. Population within 800 meters (0.5 mile) of routes for incident-free transportation using 1990 census data.

Transportation scenario	1990 Census data
Mostly legal-weight truck	7,200,000
Mostly rail	11,100,000

a. Source: TRW (1999a, pages 18 and 19).

The second kind of information used in the analysis was the number of people who potentially would be close enough to shipments to be exposed to radiation from the casks. The analysis determined the estimated offlink number of people [those within the 1.6-kilometer (1-mile) region of influence] by multiplying the population densities (persons per square kilometer) in population zones through which a route would pass by the 1.6-kilometer width of the region of influence and by the length of the route through the population zones. Onlink populations (those sharing the route and people at stops along the route) were estimated using assumptions from other EISs that have evaluated transportation impacts (DOE 1995, Appendix I; DOE 1996a, Appendix E; DOE 1997b, Appendixes F and G). The travel distance in each population zone was determined for legal-weight truck shipments by using the HIGHWAY computer program (Johnson et al. 1993a, all) and for rail shipments by using the

Table J-19. Information used for analysis of incident-free transportation impacts.

Population zones	Population within 800 meters[a] (per kilometer of route)	Travel speed (kilometers per hour) Legal-weight truck	Heavy-haul truck	Rail	Dose rate 2 meters[b] from vehicle (millirem per hour)
Urban	(c)	24	24	24[d]	10
Suburban	(c)	40[d]	40	40	10
Rural	(c)	88	40	64	10

a. 800 meters = about 2,600 feet.
b. 2 meters = about 6.6 feet.
c. Estimates of population within 800 meters of a route are based on analysis of census block data using HIGHWAY (Johnson et al. 1993a, all) and INTERLINE (Johnson et al. 1993b, all) computer programs. The analysis used actual populations along routes based on the 1990 Census.
d. Analysis of impacts for shipments of naval spent nuclear fuel used 40 kilometers (25 miles) per hour for heavy-haul truck speed and 24 kilometers (15 miles) per hour for train speed in urban, suburban, and rural zones.

INTERLINE program (Johnson et al. 1993b, all). These programs used 1990 census block group data to identify where highways and railroads enter and exit each type of population zone, which the analysis used to determine the total lengths of the highways and railroads in each population zone.

The third kind of information—the distances individuals live from the route used in the analysis—is the estimated the number of people who live within 800 meters (about 2,600 feet) of the route. The analysis assumed that population density is uniform in population zones.

The determination of the fourth kind of information used in the analysis—the time that people could be exposed as shipments passed—was based on the assumed travel speed of shipments in each population zone along the route. For example, travel at 24 kilometers (15 miles) an hour in urban areas would lead to a longer exposure time than travel at 88 kilometers (55 miles) an hour in rural areas. Persons in vehicles traveling along a route with a shipment of spent nuclear fuel or high-level radioactive waste or persons who lived near railyards where shipments would be switched between trains could be exposed for longer periods.

With the five kinds of information, the analysis used RADTRAN4 to calculate exposures for the following groups:

- *Public along the route (Offlink Exposure):* Collective doses for persons living or working within 0.8 kilometer (0.5 mile) on each side of the transportation route.

- *Public sharing the route (Onlink Exposure):* Collective doses for persons in vehicles sharing the transportation route; this includes persons traveling in the same or opposite direction and those in vehicles passing the shipment.

- *Public during stops (Stops):* Collective doses for people who could be exposed while a shipment was stopped en route. For truck transportation, these would include stops for refueling, food, and rest. For rail transportation, stops would occur in railyards along the route to switch railcars from inbound trains to outbound trains traveling toward the Yucca Mountain site, and to change train crews and equipment (locomotives).

- *Worker exposure (Occupational Exposure):* Collective doses for truck and rail transportation crew members.

- *Security escort exposure (Occupational Exposure):* Collective doses for security escorts. In calculating doses to workers the analysis conservatively assumed that the maximum number of escorts required by regulations (10 CFR 73.37) would be present for urban, suburban, and rural population zones.

The sum of the doses for the first three categories is the total nonoccupational (public) dose.

Unit dose factors were used to calculate collective dose. These factors, which are listed in Table J-20, represent the dose that would be received by a population of 1 person per square kilometer for one shipment of radioactive material moving a distance of 1 kilometer (0.62 mile) in the indicated population density zone. The unit dose factors for incident-free transportation reflect the assumption that the dose rate external to shipments of spent nuclear fuel and high-level radioactive waste would be the maximum value allowed by Department of Transportation regulations—10 millirem per hour at 2 meters (6 feet) from the side of the transport vehicle (49 CFR 173.441). The incident-free dose from transporting a single shipment was determined by multiplying the appropriate unit dose factors by corresponding distances in each of the population zones the shipment route passes through and the population density of the zone. The collective dose from all shipments from a site were determined by multiplying the dose from a single shipment by the number of shipments that would be required to transport the site's spent nuclear fuel or high-level radioactive waste to the repository. Collective dose was converted to the estimated number of latent cancer fatalities using conversion factors recommended by the International Commission on Radiological Protection (ICRP 1991, page 22). These values are 0.0004 for radiation workers and 0.0005 for the general population.

Table J-20. Unit dose factors for incident-free national truck and rail transportation of spent nuclear fuel and high-level radioactive waste.

		Unit dose factors (person-rem per kilometer)[a]		
Mode	Exposure group	Rural	Suburban	Urban
Truck	*Involved worker*	4.56×10^{-5}	1×10^{-4}	1.67×10^{-4}
	Public			
	Offlink[b]	3.2×10^{-8}	3.52×10^{-8}	4.33×10^{-8}
	Onlink[c]	7.81×10^{-6}	2.25×10^{-5}	2.32×10^{-4}
	Stops	1.87×10^{-4}	1.87×10^{-4}	1.87×10^{-4}
Rail	*Involved worker*[d]	1.22×10^{-5}	1.22×10^{-5}	1.22×10^{-5}
	Public			
	Offlink	4.38×10^{-8}	7.02×10^{-8}	1.17×10^{-7}
	Onlink	1.03×10^{-7}	1.32×10^{-6}	3.65×10^{-6}
	Stops[e]	7.42×10^{-6}	7.42×10^{-6}	7.42×10^{-6}

a. The methodology, equations, and data used to develop the unit dose factors are discussed in Madsen et al. (1986, all) and Neuhauser and Kanipe (1992, page 4-15). Cashwell et al. (1986, page 44) contains a detailed explanation of the use of unit factors.

b. Offlink general population included persons within 800 meters (2,625 feet) of the road or railway.

c. Onlink general population included persons sharing the road or railway.

d. The nonlinear component of incident-free rail dose for crew workers because of railcar inspections and classifications is 0.014 person-rem per shipment. Ostmeyer (1986, all) contains a detailed explanation of the rail exposure model.

e. The nonlinear component of incident-free rail dose for the general population because of railcar inspections and classifications is 0.0014 person-rem per shipment. Ostmeyer (1986, all) contains a detailed explanation of the rail exposure model.

J.1.3.2.2 *Methods Used To Evaluate Incident-Free Impacts to Maximally Exposed Individuals.*

To estimate impacts to maximally exposed individuals, the same kinds of information as those used for population doses (except for population size) was needed. The analysis of doses to maximally exposed individuals used projected exposure times, the distance a hypothetical individual would be from a shipment, the number of times an exposure event could occur, and the assumed external radiation dose rate 2 meters (6.6 feet) from a shipment (10 millirem per hour). These analyses used the RISKIND computer program (Yuan et al. 1995, all). DOE has used RISKIND for analyses of transportation impacts in other environmental impact statements (DOE 1995, Appendix J; DOE 1996a, Appendix E; DOE 1997b, Appendix E). RISKIND provides appropriate results for analyses of incident-free transportation and transportation accidents involving radioactive materials (Maheras and Pippen 1995, Sections 5.2 and 6.2; Biwer et al. 1997, all).

The maximally exposed individual is a hypothetical person who would receive the highest dose. Because different maximally exposed individuals can be postulated for different exposure scenarios, the analysis evaluated the following exposure scenarios.

- *Crew Members.* In general, truck crew members, including security escorts and rail security escorts, would receive the highest doses during incident-free transportation (see discussion in J.1.3.2.2.1 below). The analysis assumed that the crews would be limited to a total job-related exposure of 2 rem per year (DOE 1994, Article 211).
- *Inspectors (Truck and Rail).* Inspectors would be Federal or state vehicle inspectors. On the basis of information provided by the Commercial Vehicle Safety Alliance (Battelle 1998, all; CVSA 1999, all), the analysis assumed an average exposure distance of 1 meter (3 feet) and an exposure duration of 1 hour (see discussion in J.1.3.2.2).
- *Railyard Crew Member.* For a railyard crew member working in a rail classification yard assembling trains, the analysis assumed an average exposure distance of 10 meters (33 feet) and an exposure duration of 2 hours (DOE 1997b, page E-50).
- *Resident.* The analysis assumed this maximally exposed individual is a resident who lives 30 meters (100 feet) from a point where shipments would pass. The resident would be exposed to all shipments along a particular route (DOE 1995, page I-52).
- *Individual Stuck in Traffic (Truck or Rail).* The analysis assumed that a member of the public could be 1.2 meter (4 feet) from the transport vehicle carrying a shipping cask for 1 hour. Because these circumstances would be random and unlikely to occur more than once for the same individual, the analysis assumed the individual to be exposed only once.
- *Resident near a Rail Stop.* The analysis assumed a resident who lives within 200 meters (660 feet) of a switchyard and an exposure time of 20 hours for each occurrence. The analysis of exposure for this maximally exposed individual assumes that the same resident would be exposed to all rail shipments to the repository (DOE 1995, page I-52).
- *Person at a Truck Service Station.* The analysis assumed that a member of the public (a service station attendant) would be exposed to shipments for 1 hour for each occurrence at a distance of 20 meters (70 feet). The analysis also assumed this individual would work at a location where all truck shipments would stop.

As discussed above for exposed populations, the analysis converted radiation doses to estimates of radiological impacts using dose-to-risk conversion factors of the International Commission on Radiological Protection.

J.1.3.2.2.1 Incident-Free Radiation Doses to Inspectors. DOE estimated radiation doses to the state inspectors who would inspect shipments of spent nuclear fuel and high-level radioactive waste originating in, passing through, or entering a state. For legal-weight truck and railcar shipments, the analysis assumed that:

- Each inspection would involve one individual working for 1 hour at a distance of 2 meters (6.6 feet) from a shipping cask.
- The radiation field surrounding the cask would be the maximum permitted by regulations of the Department of Transportation (49 CFR 173.441).
- There would be no shielding between an inspector and a cask.

For rail shipments, the analysis assumed that:

- There would be a minimum of two inspections per trip—one at origin and one at destination—with additional inspections in route occurring about once every 500 kilometers (300 miles) of railcar travel.
- Rail crews would conduct the remaining along-the-route inspections.

For legal-weight truck shipments, the analysis assumed that:

- On average, state officials would conduct two inspections during each trip – one at the origin and one at the destination.
- The inspectors would use the Enhanced North American Uniform Inspection Procedures and Out-of-Service Criteria for Commercial Highway Vehicles Transporting Transuranics, Spent Nuclear Fuel, and High-Level Radioactive Waste (CVSA 1999, all).
- The shipments would receive a Commercial Vehicle Safety Alliance inspection sticker on passing inspection and before departing from the 77sites.
- Display of such a sticker would provide sufficient evidence to state authorities along a route that a shipment complied with Department of Transportation regulations (unless there was contradictory evidence), and there would be no need for additional inspections.

The analysis determined doses to state inspectors in two ways. For rail shipments, inspector doses were based on the equations and assumptions used in the RADTRAN4 computer program. The program uses an empirically derived equation that is based on observations of rail classification yard operations, as follows:

$$\text{Dose} = K_0 \times \text{dose rate} \times \text{casks per shipment} \times \text{number of shipments} \times 0.16 \times 0.001$$

where:

dose = rem of exposure to an inspector

K_0	=	a shape factor for the cask assumed for purposes of analysis (meters); 6 meters for rail cask that would ship spent nuclear fuel
dose rate	=	the dose rate in millirem per hour 1 meter from the surface of the cask; set to 14 millirem per hour for the analysis
casks per shipment	=	the average number of casks (one cask per railcar) in a train; set to 1 for the analysis
number of shipments	=	number of shipments inspected (set to 1 for the analysis)
0.16	=	exposure factor that translates the product of cask dose rate and shape factor into inspector dose (meters per hour)
0.001	=	conversion factor to convert millirem per hour to rem per hour.

The equation shows that the calculated value for whole-body dose to an individual inspector for one inspection would be 13.4 millirem. An inspector in Nevada who inspected all rail shipments under the mostly rail scenario would receive a whole body dose of $470 \times 13.4 = 6.3$ rem in a year. If the same inspector inspected all shipments over the 24 years of the Proposed Action, he or she would be exposed to 150 rem. Using the dose to risk conversion factors published by the International Commission on Radiation Protection, this exposure would increase the likelihood of the inspector incurring a fatal cancer. This would add 6 percent to the likelihood for fatal cancers from all other causes, increasing the likelihood from approximately 23 percent (ACS 1998, page 10) to 29 percent.

For shipments by legal-weight truck, the analysis used the RISKIND computer program to estimate doses to inspectors (Yuan et al. 1995, all). The data used by the code to calculate dose includes the estimated value for dose rate at 1 meter (3.3 feet) from a cask surface, the length and diameter of the cask, the distance between the location of the individual and the cask surface, and the estimated time of exposure. For this calculation, the analysis assumed that an inspector following Commercial Vehicle Safety Alliance procedures (CVSA 1999, all) would work for 1 hour at an average distance of 2 meters (6.6 feet) from the cask. The analysis assumed that a typical legal-weight truck cask would be about 1 meter in diameter and about 5 meters (16 feet) long and that the dose rate 1 meter from the cask surface would be 14 millirem per hour. A dose rate of 14 millirem per hour 1 meter from the surface of a truck cask is approximately equivalent to the maximum dose rate allowed by Department of Transportation regulations for exclusive-use shipments of radioactive materials (49 CFR 173.441).

Using this data, the RISKIND computer code calculated an expected dose of 18 millirem for an individual inspector. Under the mostly legal-weight truck scenario in which approximately 2,100 legal-weight truck shipments would arrive in Nevada annually, a Nevada inspector working 1,800 hours per year could inspect as many as 470 shipments in a year. This inspector would receive a whole-body dose of 8.5 rem. If this same inspector inspected all shipments over the 24 years of the Proposed Action, he or she would be exposed to 204 rem. Using the dose to risk conversion factors published by the International Commission on Radiation Protection, this exposure would increase the likelihood of this individual contracting a fatal cancer. This would add about 8 percent to the likelihood for fatal cancers from all other causes, increasing the likelihood from approximately 22 percent (ACS 1998, page 10) to 32 percent.

Under the mostly legal-weight truck scenario, the annual committed dose to inspectors in a state that inspected all incoming legal-weight truck shipments containing spent nuclear fuel or high-level radioactive waste would be about 38 person-rem. Over 24 years, the population dose for these inspectors would be about 910 person-rem. This would result in about 0.34 latent cancer fatality (this is equivalent to a 36-percent likelihood that there would be 1 additional latent cancer fatality among the exposed group).

DOE implements radiation protection programs at its facilities where there is the potential for worker exposure to cumulative doses from ionizing radiation. The Department anticipates that the potential for individual whole-body doses such as those reported above would lead an involved state to implement such a radiation protection program. If similar to those for DOE facilities, the administrative control limit on individual dose would not exceed 2 rem per year (DOE 1994, Article 211) and the expected maximum exposure for inspectors would be less than 500 millirem per year.

J.1.3.2.2.2 Incident-Free Radiation Doses to Escorts. Transporting spent nuclear fuel to the Yucca Mountain site would require the use of physical security and other escorts for the shipments. Regulations (10 CFR 73.37) require escorts for highway and rail shipments. These regulations require two escorts (individuals) for truck shipments traveling in highly populated (urban) areas. One of the escorts must be in a vehicle that is separate from the shipment vehicle. For rail shipments in urban areas, at least two escorts must maintain visual surveillance of a shipment from a railcar that accompanies a cask car.

In areas that are not highly populated (suburban and rural), one escort must accompany truck shipments. The escort can ride in the cab of the shipment vehicle. At least one escort is required for rail shipments in suburban and rural areas. However, for rail shipments, the escort must occupy a railcar that is separate from the cask car and must maintain visual surveillance of the shipment at all times.

For legal-weight truck shipments, the analysis assumed that a second driver, who would be a member of the vehicle crew, would serve as an escort in all areas. The analysis assigned a second escort for travel in urban areas and assumed that this escort would occupy a vehicle that followed or led the transport vehicle by at least 60 meters (about 200 feet). The analysis assumed that the dose rate at a location 2 meters (6.6 feet) behind the vehicle would be 10 millirem per hour, which is the limit allowed by Department of Transportation regulations (49 CFR 173.441). Using this information, the analysis used the RISKIND computer program to calculate a value of approximately 0.11 millirem per hour for the dose rate 60 meters behind the transport vehicle; this is the estimated value for the dose rate in a following escort vehicle. The value for the dose rate in an escort vehicle that preceded a shipment would be lower. Because the dose rate in the occupied crew area of the transport vehicle would be less than 2 millirem per hour, the dose rate 2 meters in front of the vehicle would be much less than 10 millirem per hour, the value assumed for a location 2 meters behind the vehicle. The value of 2 millirem per hour in normally occupied areas of transport vehicles is the maximum allowed by Department of Transportation regulations (49 CFR 173.441).

To calculate the dose to escorts, the analysis assumed that escorts in separate vehicles would be required in urban areas as shipments traveled to the Yucca Mountain site. The calculations used the RISKIND computer program (Yuan et al. 1995, all); the distance of travel in urban areas provided by the HIGHWAY and INTERLINE computer codes; and the estimated speed of travel in urban areas based on data in Table J-19 to estimate the total dose to escorts. For example, truck shipments could be escorted through an average of five urban areas on average for 30 minutes in each. Using these assumptions and the estimated dose rate in an escort vehicle, the estimated dose for escorts in separate vehicles is 0.28 millirem per shipment (0.28 millirem = 5 areas per shipment × 0.5 hour per area × 0.11 millirem per hour). For the 24 years of the Proposed Action, the total dose to escorts in separate vehicles would, therefore, be about 14 rem (0.28 millirem per shipment × 50,000 shipments). This dose would lead to 0.02 latent cancer fatality in the population of escorts who would be affected.

For rail shipments, the analysis assumed that escorts would be 30 meters (98 feet) away from the end of the shipping cask on the nearest railcar. This separation distance is the sum of the:

- Length of a buffer car [about 15 meters (49 feet)] between a cask car and an escort car required by Department of Transportation regulations (49 CFR 174.89),
- Normal separation between cars [a total of about 2 meters (6.6 feet) for two separations],
- Distance from the end of a cask to the end of its rail car [about 5 meters (16 feet)], and
- Assumed average distance from the escort car's near-end to its occupants [5 to 10 meters (16 to 32 feet)].

This analysis assumed that the dose rate at 2 meters (6.6 feet) from the end of the cask car would be 10 millirem per hour, the maximum allowed by Department of Transportation regulations (49 CFR 173.441). The analysis used these assumptions and the RISKIND computer program to estimate 0.46 millirem per hour as the dose rate in the occupied areas of the escort railcar. For example, an individual escort who occupied the escort car continuously for a 5-day cross-country trip would receive a maximum dose of about 55 millirem. Escorting 26 shipments in a year, this individual would receive a maximum dose of 1.4 rem. Over the 24 years of the Proposed Action, if the same individual escorted 26 shipments every year, he or she would receive a dose of about 34 rem. Using the dose-to-risk conversion factors recommended by the International Commission on Radiation Protection (ICRP 1991, page 22), this dose would increase the potential for the individual to contract a fatal cancer from about 22 percent (ACS 1998, page 10) to 24 percent.

J.1.3.2.3 *Vehicle Emission Impacts*

Human health impacts from exposures to vehicle exhaust depend principally on the distance traveled in an urban population zone and on the impact factors for particulates and sulfur dioxide from truck (including escort vehicles) or rail emissions, fugitive dust generation, and tire abrasion (DOE 1995, page I-52).

The analysis estimated incident-free impacts from nonradiological causes using unit risk factors that account for both fatalities associated with the emissions of pollution in urban, suburban, and rural areas by transportation vehicles, including escort vehicles. Because the impacts would occur equally for trucks transporting loaded or unloaded shipping casks, the analysis used round-trip distances. Escort vehicle impacts were included only for loaded shipment miles.

The analysis used impact factors for effects on urban areas of 0.00000016 fatality per urban mile traveled (0.0000001 fatality per kilometer) by trucks and 0.00000021 fatality per urban mile traveled (0.00000013 fatality per kilometer) by trains (Rao, Wilmot, and Luna 1982, all). The region of influence used in the analysis for exposure to vehicle emissions was a band between 30 and 805 meters (98 and 2,640 feet) wide on both sides of the transportation route.

In addition to unit risk factors used to estimate impacts from vehicle emissions in urban areas, an additional factor was used to estimate health effects from vehicle exhaust emissions in rural areas. Based on data in a study by the Environmental Protection Agency that addressed latent cancer consequences of vehicle exhausts, a factor of 0.000000000072 fatality per kilometer traveled was calculated for use in rural and suburban population zones (DOE 1995, page I-52).

Although the analysis estimated human health and safety impacts of transporting spent nuclear fuel and high-level radioactive waste, exhaust and other pollutants emitted by transport vehicles into the air would not measurably affect national air quality. National transportation of spent nuclear fuel and high-level radioactive waste, which would use existing highways and railroads would average 14.2 million truck kilometers per year for the mostly truck case and 3.5 million railcar kilometers per year from the mostly rail case. The national yearly average for total highway and railroad traffic is 186 billion truck kilometers and 49 billion railcar kilometers (BTS 1999, Table 3-22). Spent nuclear fuel and high-level radioactive waste transportation would represent a very small fraction of the total national highway and railroad traffic (0.008 percent of truck kilometers and 0.007 percent of rail car kilometers). In addition, the contributions to vehicle emissions in the Las Vegas air basin, where all truck shipments (an average of five per day) would travel under the mostly legal-weight truck scenario, would be small in comparison to those from other vehicle traffic in the area. The annual average daily traffic on I-15 0.3 kilometer (0.2 mile) north of the Sahara Avenue interchange is almost 200,000 vehicles (NDOT 1997, page 7), about 20 percent of which are trucks (Cerocke 1998, all). For these reasons, national transportation of spent nuclear fuel and high-level radioactive waste by truck and rail would not constitute a meaningful source of air pollution along the nation's highways and railroads.

J.1.3.2.4 *Sensitivity of Dose Rate to Characteristics of Spent Nuclear Fuel*

For this analysis, DOE assumed that the dose rate external to all shipments of spent nuclear fuel and high-level radioactive waste would be the maximum value allowed by regulations (49 CFR 173.441). However, the dose rate for actual shipments would not be the maximum value of 10 millirem per hour at 2 meters (6.6 feet) from the sides of vehicles. Administrative margins of safety that are established to compensate for limits of accuracy in instruments and methods used to measure dose rates at the time shipments are made would result in lower dose rates. In addition, the characteristics of spent nuclear fuel and high-level radioactive waste that would be loaded into casks would always be within the limit values allowed by the cask's design and its Nuclear Regulatory Commission certificate of compliance.

For example, DOE used data provided in the *GA-4 Legal-Weight Truck Cask Design Report* (General Atomics 1993, pages 5.5-18 and 5.5-19) to estimate dose rates 2 meters (6.6 feet) from transport vehicles for various characteristics of spent nuclear fuel payloads. Figure J-7 shows ranges of burnup and cooling times for spent nuclear fuel payloads for the GA-4 cask. The figure indicates the characteristics of a typical pressurized-water reactor spent nuclear fuel assembly (see Appendix A). Based on the design data for the GA-4 cask, a shipment of typical pressurized-water reactor spent nuclear fuel would result in a dose rate of about 6 millirem per hour at 2 meters from the side of the transport vehicle, or about 60 percent of the limit established by Department of Transportation regulations (49 CFR 173.441).

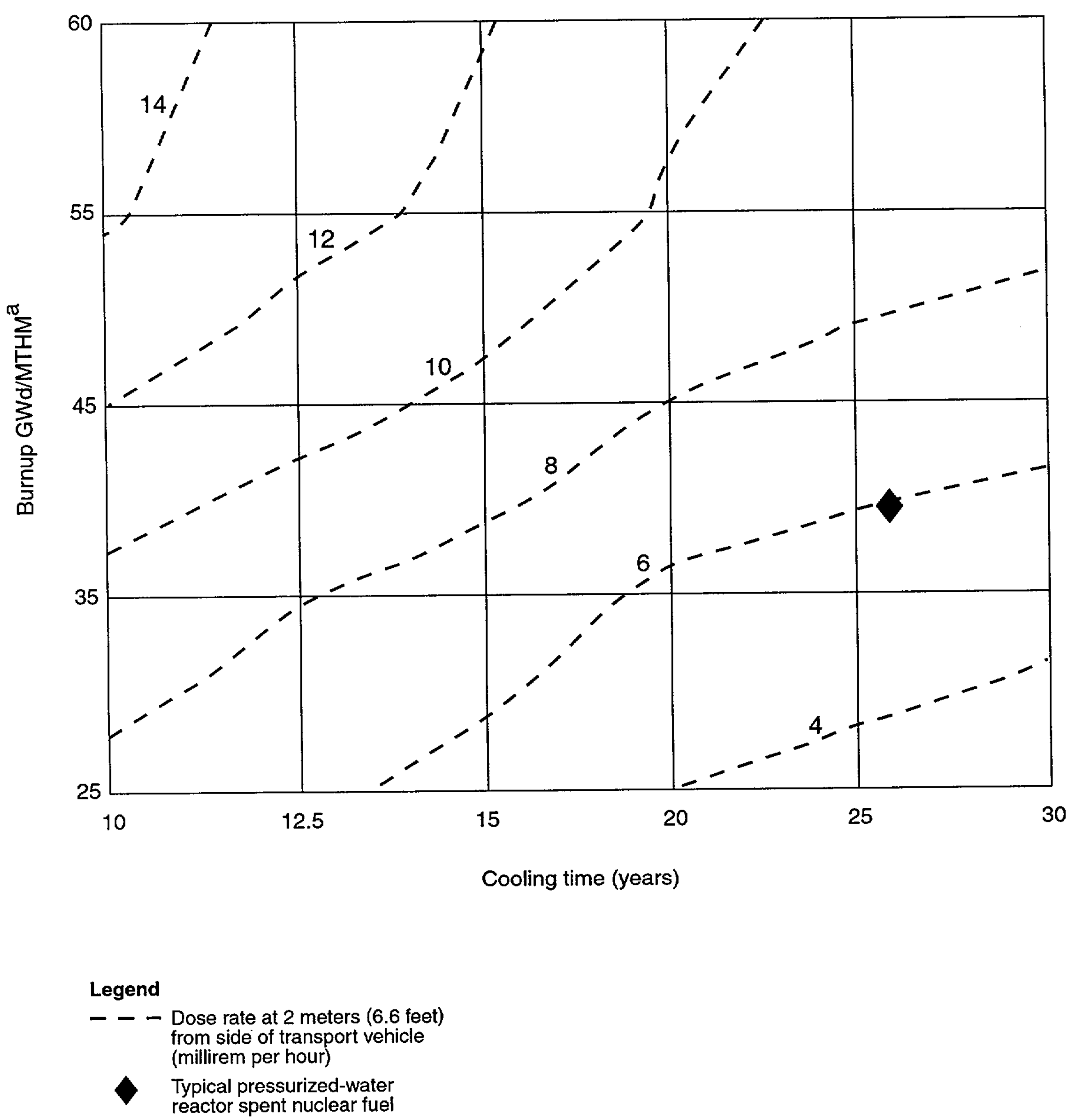
Therefore, DOE estimates that, on average, dose rates at locations 2 meters (6.6 feet) from the sides of transport vehicles would be about 50 to 70 percent of the regulatory limits. As a result, DOE expects radiological risks to workers and the public from incident-free transportation to be no more than 50 to 70 percent of the values presented in this EIS.

J.1.4 METHODS AND APPROACH TO ANALYSIS OF ACCIDENT SCENARIOS

J.1.4.1 Accidents in Loading Operations

J.1.4.1.1 *Radiological Impacts of Loading Accidents*

The analysis used information in existing reports to consider the potential for radiological impacts from accidents during spent nuclear fuel loading operations at the commercial and DOE sites. These included a report that evaluated health and safety impacts of multipurpose canister systems (TRW 1994, all) and two safety analysis reports for onsite dry storage of commercial spent nuclear fuel at independent spent fuel storage installations (PGE 1996, all; CP&L 1989, all). The latter reports address the handling and loading of spent nuclear fuel assemblies in large casks similar to large transportation casks. In addition,



a. Gigawatt days per metric ton of heavy metal.

Figure J-7. Comparison of GA-4 cask dose rate and spent nuclear fuel burnup and cooling time.

DOE environmental impact statements on the management of spent nuclear fuel and high-level radioactive waste (DOE 1995, all; DOE 1997b, all) provided information on radiological impacts from loading accidents.

TRW (1994, Sections 3.2 and 4.2) discusses potential accident scenario impacts of four cask management systems at electric utility and other spent nuclear fuel storage sites. This report concentrated on unplanned contact (bumping) during lift-handling of casks, canisters, or fuel assemblies. The two safety analysis reports for independent spent fuel storage installations for commercial spent nuclear fuel (PGE 1996, all; CP&L 1989, all) evaluated a comprehensive spectrum of accident-initiating events. These events included fires, chemical explosions, seismic events, nuclear criticality, tornado strikes and tornado-generated missile impacts, lightning strikes, volcanism, canister and basket drop, loaded shipping cask drop, and interference (bumping, binding) between the transfer cask and storage module. The DOE environmental impact statements for the interim management of spent nuclear fuel and high-level radioactive waste (DOE 1995, Appendix E; DOE 1997b, Appendixes F and G) included radiological impacts from potential accident scenarios associated with preparing, storing, and shipping these materials. These EISs do not discuss quantitative radiological impacts for accident scenarios associated with material loading, but do contain estimates of radiological impacts from accident scenarios for the spent nuclear fuel and high-level radioactive waste management activities considered. As discussed for routine loading operations, this analysis converted radiation doses to estimates of radiological impacts using dose-to-risk conversion factors of the International Commission on Radiological Protection.

J.1.4.1.2 *Industrial Safety Impacts of Loading Operations at Commercial Facilities*

The principal industrial safety impact parameters of importance to commercial industry and the Federal Government are (1) total recordable (injury and illness) cases, (2) lost workday cases associated with workplace injuries and illnesses, and (3) workplace fatalities. The frequency of these impacts under the Proposed Action and the inventory modules (Modules 1 and 2) was projected using the involved worker level of effort, expressed as the number of full-time equivalent worker multiples, that would be needed to conduct shipment tasks. The workplace loss incidence rate for each impact parameter [as shown in the DOE Computerized Accident/Incident Reporting and Recordkeeping System (CAIRS) data base (DOE 1999, all)] was used as a multiplier to convert the level of effort to expected industrial safety losses.

DOE did not explicitly analyze impacts to noninvolved workers in its earlier reports (Schneider et al. 1987, all; Smith, Daling, and Faletti 1992, all). However, for purposes of analysis in this EIS, DOE estimated that impacts to noninvolved workers would be 25 percent of the impacts to the involved workforce. This assumption is based on (1) the DOE estimate that about one of five workers assigned to a specific task would perform administrative or managerial duties, and (2) the fact that noninvolved worker loss incidence rates are generally less than those for involved workers (see Appendix F, Table F-2).

The estimated involved worker full-time equivalent multiples for each shipment scenario were estimated using the following formula:

$$\text{Involved worker full-time equivalent multiples} = (A \times B \times C \times D) \div E$$

where: A = number of shipments (from Tables J-5 and J-6)

B = average loading duration for each shipment by fuel type and conveyance mode (workdays; from Table J-15)

C = workday conversion factor = 8 hours per workday

D = involved worker crew size (13 workers; from Table J-16)

E = full-time equivalent conversion factor = 2,000 worker hours per full-time equivalent

The representative CAIRS data base loss incidence rate for each total recordable case, lost workday case, and fatality trauma category (for example, the number of total recordable cases per full-time equivalent) was then multiplied by the involved worker full-time equivalent multiples to project the associated incidence. The involved worker total recordable case incidence rate used was that reported in the DOE CAIRS data base (DOE 1999, all) for the 1992 to 1997 period of record because neither the Nuclear Regulatory Commission nor the Bureau of Labor Statistics maintains data on commercial power reactor industrial safety losses. The total recordable case incidence rate, 410 cases in a workforce of 15,000 workers (0.03 total recordable case per full-time equivalent), is the averaged loss experience at the three principal DOE sites: the Savannah River Site, Hanford Site, and Idaho National Environmental and Engineering Laboratory. The DOE sites were chosen because the operations and hazards would be representative of those encountered at commercial power reactor sites. Because lost workday cases are linked to the total recordable case experience (that is, each lost workday case would have to be included in the total recordable case category), the same DOE CAIRS data base period of record and facilities were used in the selection of the involved worker lost workday case incidence rate [200 lost workday cases in a workforce of 15,000 workers (0.013 lost workday case per full-time equivalent)].

The TRW (1994, all) study concluded that radiological impacts from handling incidents would be small. The total person-rem exposure for accidents in handling the four cask systems considered in the study would vary from 0.1 rem to 0.04 rem. This exposure would be the total for all persons who would be exposed, onsite workers as well as the public. The highest estimated exposure (0.1 person-rem) would result in 0.00005 latent cancer fatality in the exposed population.

The involved worker fatality incidence rate used was that also reported in the DOE CAIRS data base, but for the 1996 to 1997 (through the third quarter) period of record. The average DOE and contractor fatality rates used (2.9 fatalities among 100,000 workers) represent losses among workers operating equipment and handling waste materials at the principal DOE sites. This fatality incidence rate represents government and contractor experience in the DOE complex and operations that are governed by safety and administrative controls that would be similar to those used at commercial power reactor sites.

For comparison, the noninvolved worker total recordable case, lost workday case, and fatality incidence rates using the same data base sources are 0.033, 0.016, and 0.000029, respectively. However, because the CAIRS data base did not include fatality rates for noninvolved workers, the involved worker rate was used.

J.1.4.1.3 *Industrial Safety Impacts of DOE Loading Operations*

The technical approach and loss multipliers discussed in Section J.1.4.1.2 for commercial power reactor sites analysis were used for the analysis of spent nuclear fuel and high-level radioactive waste loading impacts at DOE sites. Because no information existed on the high-level radioactive waste loading duration for the truck and rail transportation modes, DOE assumed that the number of full-time equivalent involved workers for the two transportation modes would be the same as that for the DOE sites shipping spent nuclear fuel. For those sites, the average number of full-time equivalent workers would be about 0.07 and 0.12 per shipment for the truck and rail transportation modes, respectively.

J.1.4.2 Transportation Accident Scenarios

J.1.4.2.1 *Radiological Impacts of Transportation Accidents*

A potential consequence and risk of transportation would be accidents that released and dispersed radioactive material from safe containment in transportation packages. Such releases and dispersals, if they occurred, would lead to impacts to human health and the environment. The following sections describe the methods for analyzing the risks and consequences of accidents that could occur in the course of transporting spent nuclear fuel and high-level radioactive waste to a nuclear waste repository at the Yucca Mountain site. They discuss the bases for, and methods for, determining rates at which accidents are assumed to occur, the severity of these accidents, and the amounts of materials that could be released. Accident rates, severities, and the corresponding quantities of radioactive materials that could be released are essential data used in the analyses. Appendix A presents the quantities of radioactive materials in a typical pressurized-water reactor spent nuclear fuel assembly used in the analysis of accident consequences and risks. Legal-weight truck casks would contain as many as four pressurized-water reactor spent nuclear fuel assemblies, and rail casks would contain as many as 36 (see Table J-3).

In addition to accident rates and severities, an important variable in assessing impacts from transportation accident scenarios is the type of material that would be shipped. Accordingly, this appendix presents information used in the analyses of impacts of accidents that could occur in the course of transporting commercial pressurized- and boiling-water reactor fuels, DOE spent nuclear fuels, and DOE high-level radioactive waste.

POTENTIAL EFFECTS OF HUMAN ERROR ON ACCIDENT IMPACTS

The accident scenarios described in this chapter would be mostly a direct consequence of error on the part of transport vehicle operators, operators of other vehicles, or persons who maintain vehicles and rights-of-way. The number and severity of the accidents would be minimized through the use of trained and qualified personnel.

Others have argued that other kinds of human error could also contribute to accident consequences: (1) undetected error in the design and certification of transportation packaging (cask) used to ship radioactive material, (2) hidden or undetected defects in the manufacture of these packages, and (3) error in preparing the packages for shipment. DOE has concluded that regulations and regulatory practices of the Nuclear Regulatory Commission and the Department of Transportation address the design, manufacture, and use of transportation packaging and are effective in preventing these kinds of human error by requiring:

- Independent Nuclear Regulatory Commission review of designs to ensure compliance with requirements (10 CFR Part 71)

- Nuclear Regulatory Commission-approved and audited quality assurance programs for design, manufacturing, and use of transportation packages

In addition, Federal provisions (10 CFR Part 21) provide additional assurance of timely and effective actions to identify and initiate corrective actions for undetected design or manufacturing defects. Furthermore, conservatism in the approach to safety incorporated in the regulatory requirements and practices provides confidence that design or manufacturing defects that might remain undetected or operational deficiencies would not lead to a meaningful reduction in the performance of a package under normal or accident conditions of transportation.

For exposures to ionizing radiation following accidents, risks were analyzed in terms of dose and latent cancer fatalities to the public and workers. The analyses of risk also addressed the potential for fatalities that would be the direct result of mechanical forces and other nonradiological effects that occur in everyday vehicle and industrial accidents.

The transportation of spent nuclear fuel and high-level radioactive waste from the 77 sites to the Yucca Mountain site would be conducted in a manner that complied fully with regulations of the U.S. Department of Transportation and Nuclear Regulatory Commission. These regulations specify requirements that promote safety and security in transportation. The requirements apply to carrier operations; in-transit security; vehicles; shipment preparations; documentation; emergency response; quality assurance; and the design, certification, manufacture, inspection, use, and maintenance of packages (casks) that would contain the spent nuclear fuel and high-level radioactive waste.

Because of the high level of performance required by regulations for transportation casks (49 CFR Part 173 and 10 CFR Part 71), the Nuclear Regulatory Commission estimates that in 99.4 percent of rail and truck accidents no cask contents would be released (Fischer et al. 1987, page 9-10). The 0.6 percent of accidents that could cause a release of radioactive materials from casks can be described by a spectrum of accident severity. As the severity of an accident increases, the fraction of radioactive material contents that would be released from transportation casks also increases. However, as the severity of an accident increases it is less likely to occur. In its Modal Study (Fischer et al. 1987, all), the Nuclear Regulatory Commission developed an accident analysis methodology that uses this concept of a spectrum of severe accidents to calculate the probabilities and consequences of unlikely accidents that could occur in transporting highly radioactive materials.

Although the Nuclear Regulatory Commission approach, which was used in this EIS, provides a method for determining the frequency with which severe accidents can be expected to occur, their severity, and their consequences, a method does not exist for predicting where along routes accidents would occur. Therefore, for the analyses of impacts presented here the method used in the RADTRAN4 computer code (Neuhauser and Kanipe 1992, all) is used. This method assumes that accidents could occur at any location along routes, with their frequency of occurrence being determined by the accident rate characteristic of the states through which the route passes and the number of shipments that travel the route.

The transportation accident scenario analysis evaluated radiological impacts to populations and to hypothetical maximally exposed individuals and estimated fatalities that could occur from traffic accidents. It included both rail and legal-weight truck transportation. The analysis used the RADTRAN4 (Neuhauser and Kanipe 1992, all) and RISKIND (Yuan et al. 1995, all) computer programs to determine accident consequences and risks. DOE has used both codes in recent DOE environmental impact statements (DOE 1995, Appendix J; DOE 1996a, Appendix E; DOE 1997b, Appendixes F and G) that address impacts of transporting radioactive materials. The analyses used seven kinds of information to determine the consequences and risks of accidents for populations:

- Routes from the 77 sites to the repository and their lengths in each state and population zone
- The number of shipments that would be transported over each route
- State-specific accident rates
- The kind and amount of radioactive material that would be transported in shipments
- Probabilities of release and fractions of cask contents that could be released in accidents

ESTIMATING ACCIDENT RISK

Assessing the radiological impact of accidents involves estimating the probability that an accident might occur and estimating the accident consequences. The probability, or chance, that an accident will occur is multiplied by the consequences of the accident to determine accident risk.

One method for estimating accident probabilities uses historic information on the rate at which accidents of a similar type or severity occur (accidents per vehicle-mile traveled). Information of this type is maintained as transportation accident data by the Department of Transportation and by transportation safety organizations in state governments. Accident rates are multiplied by the total number of miles that vehicles would travel to estimate the number of accidents.

Determining radiological accident consequences requires estimating the quantity of radionuclides likely to be released and the environmental transport mechanisms that would bring the radionuclides into contact with people and then calculating the resultant radiation dose. Because of the large amounts of data these calculations require, conservative or bounding assumptions are commonly used to simplify the calculation task. As a result, calculated risks tend to be overestimates.

- The number of people who could be exposed to accidents and how far they lived from the routes
- Exposure scenarios that include multiple exposure pathways, state-specific agricultural factors, and atmospheric dispersion factors for neutral and stable conditions applicable to the entire country for calculating radiological impacts

The analysis used the same routes and lengths of travel as the analysis of incident-free transportation impacts discussed above.

DOE used the CALVIN computer code discussed earlier, the DOE Throughput Study (TRW 1997, all), and information provided by the DOE National Spent Nuclear Fuel Program (Jensen 1998, all) to calculate the number of shipments from each site and, thus, the number of shipments that would use a particular route.

The state-specific accident rates (accidents and fatalities per kilometer of vehicle travel) used in the analysis included accident statistics for commercial motor carrier operations for the Interstate Highway System, other U.S. highways, and state highways for each of the 48 contiguous states (Saricks and Tompkins 1999, all). The analysis also used average accident and fatality rates for railroads in each state. The data specifically reflect accident and fatality rates that apply to commercial motor carriers and railroads.

Appendix A contains information on the radioactive material contents of shipments. Appendix A, Section A.2.1.5 describes the characteristics of the spent nuclear fuel and high-level radioactive waste that would be shipped. The analysis assumed that the average inventory of radioactive materials in shipments would be typical pressurized-water reactor spent nuclear fuel that had been removed from reactors for 25.8 years. Appendix A describes this inventory. The estimated impacts would be less if the analysis used the characteristics of a typical boiling-water reactor spent nuclear fuel, DOE spent nuclear fuel (including naval spent nuclear fuel, which the analysis assumed would be removed from reactors 5 years before its shipment to the repository), or high-level radioactive waste.

The analysis also used the number of people who potentially would be close enough to transportation routes at the time of an accident to be exposed to radiation or radioactive material released from casks, and the distances these people would be from the accidents. It used the HIGHWAY and INTERLINE computer programs to determine this estimated number of people and their distances from accidents. HIGHWAY and INTERLINE used 1990 Census data for this analysis. The analysis assumed that the region of influence extended 80 kilometers (50 miles) from an accident.

Accident Severity Categories and Conditional Probabilities

The classification scheme used in the Modal Study for both truck and rail transportation accidents is shown in Figure J-8. As shown, accident severity is a function of two variables. The first variable is the mechanical force that occurs in impacts. In the figure, mechanical force is represented by the deformation (strain) in a cask's containment (inner shell) that the force would cause. The second variable is thermal energy, or the heat input to a cask engulfed by fire. In the figure, thermal energy is represented by the midpoint temperature of a cask's lead shield wall following heating, as in a fire.

Because all accident scenarios that would involve casks can be described in these terms, the severity of accidents can be analyzed independently of specific accident sequences. In other words, any sequence of events that results in an accident in which a cask is subjected to mechanical forces, within a certain range of values, and possibly fire is assigned to the accident severity category associated with the applicable ranges for the two parameters. This accident severity scheme enables analysis of a manageable number of accident situations while accounting for all reasonably foreseeable transportation accidents, including accidents with low probabilities but high consequences and those with high probabilities but low consequences.

For the analysis of impacts, a conditional probability was assigned to each accident severity category. Figure J-8 also shows the conditional probabilities developed in the Modal Study for the accident severity matrix. These conditional probabilities are used in the analysis of impacts presented in this chapter. The conditional probabilities are the chances that accidents will involve the mechanical forces and the heat energy in the ranges that apply to the categories. For example, accidents that would fall into the category labeled R(1,1), which represents the least severe accident in the matrix, would be likely to make up 99.4 percent of all accidents that would involve truck and railcar shipments of casks carrying spent nuclear fuel or high-level radioactive waste. The mechanical forces and heat in accidents in this category would not exceed the regulatory design standards for casks. Using the information in the figure, an accident in this category could cause a maximum of 0.2 percent strain (deformation) in a cask's containment and could heat the lead shielding to 260°C (500°F) degrees. These damage conditions are within the range of damage that would occur to casks subjected to the hypothetical accident conditions tests that Nuclear Regulatory Commission regulations require a cask to survive (10 CFR Part 71). Category R(4,5)-accidents, which would cause extensive damage to a cask, are very severe but very infrequent. The Category R(4,5) accidents would occur an estimated 3.4 times in each 100 trillion rail accidents and less than one time in each 10 quadrillion truck accidents.

The analysis of accident risks presented in this appendix used the frequency that would be likely for accidents in each of the severity categories. This frequency was determined by multiplying the category's conditional probability by the accident rates for each state's urban, suburban, and rural population zones and by the shipment distances in each of these zones, and then adding the results. The accident rates in the population density zones in each state are distinct and correspond to traffic conditions, including average vehicle speed, traffic density, and other factors, including rural, suburban, or urban location.

In terms of potential to release radioactivity to the environment, the most severe of reasonably foreseeable accidents are those that would fall into one of the eight categories of very severe accidents. For these eight categories, the fractions and characteristics of radioactive materials that would be released in an

P_t P_r	R(4,1) 1.532×10^{-7} 1.786×10^{-9}	R(4,2) 3.926×10^{-14} 3.290×10^{-13}	R(4,3) 1.495×10^{-14} 2.137×10^{-13}	R(4,4) 7.681×10^{-16} 1.644×10^{-13}	R(4,5) $<1 \times 10^{-16}$ 3.459×10^{-14}
P_t P_r	R(3,1) 1.7984×10^{-3} 5.545×10^{-4}	R(3,2) 1.574×10^{-7} 1.021×10^{-7}	R(3,3) 2.034×10^{-7} 6.634×10^{-8}	R(3,4) 1.076×10^{-7} 5.162×10^{-8}	R(3,5) 4.873×10^{-8} 5.296×10^{-8}
P_t P_r	R(2,1) 3.8192×10^{-3} 2.7204×10^{-3}	R(2,2) 2.330×10^{-7} 5.011×10^{-7}	R(2,3) 3.008×10^{-7} 3.255×10^{-7}	R(2,4) 1.592×10^{-7} 2.531×10^{-7}	R(2,5) 7.201×10^{-8} 1.075×10^{-8}
P_t P_r	R(1,1) 0.994316 0.993962	R(1,2) 1.687×10^{-5} 1.2275×10^{-3}	R(1,3) 2.362×10^{-5} 7.9511×10^{-4}	R(1,4) 1.525×10^{-5} 6.140×10^{-4}	R(1,5) 9.570×10^{-6} 1.249×10^{-4}

Structural response (maximum strain on inner shell, percent)

S_3 (30)

S_2 (2)

S_1 (0.2)

T_1 (500) T_2 (600) T_3 (650) T_4 (1,050)

Thermal response (lead mid-thickness temperature, °F)

Legend

R(x,y) = The label used to identify the cell in the accident response matrix located at the x row from the bottom of the matrix and y column from the left of the matrix. Thus, (R1,1) is the identifier for the cell in the lower left corner of the matrix.

P_t = Probability of occurrence assuming a truck accident occurs.

P_r = Probability of occurrence assuming a rail accident occurs.

Note:
- Maximum strain between 0 and 0.2 percent (S_1) for the inner shell of a cask would be within the design conditions for a Nuclear Regulatory Commission-certified shipping cask. There would be permanent deformation after the load is removed. S_1 strains could occur in impacts against medium hardness structures (for example, bridge abutments) at speeds up to 100 kilometers (60 miles per hour).
- Strains between 0.2 and 2 percent (S_2) would result in small permanent deformations. S_2 strains could occur in impacts against medium hardness structures at speeds up to 130 kilometers (80 miles per hour).
- Strains between 2 and 30 percent (S_3) would result in large permanent deformations. S_3 strains could occur in impacts against medium hardness structures at speeds greater than 130 kilometers (80 miles per hour).

Source: Fischer et al. (1987, pages 4-8, 7-25, and 7-26).

Figure J-8. Probability matrix for mechanical forces and heat in transportation accidents.

accident were estimated to be the same. That is, for a shipment of spent nuclear fuel that is involved in an accident classified as Category R(4,1), the amount and characteristics of radioactive material assumed to be released would be the same as those for an accident that would fall into Category R(4,2), R(4,3), R(4,4), R(4,5), R(1,5), R(2,5), or R(3,5). Because the releases of radioactive materials that could occur are assumed to be the same for each of these eight categories, the probabilities of occurrence can be summed. This sum is used to calculate a collective probability for the most severe of the accidents addressed in this analysis. Thus, the conditional probability of a truck accident of the greatest severity that is analyzed would be 0.0000098 per accident event (about 1 chance in 100,000 per accident).

By combining categories for which the releases of radioactive materials are assumed to be equivalent, the 20 accident categories in Figure J-8 are reduced to six collective categories. The first is the same as severity category R(1,1); the second collects severity categories R(1,2) and R(1,3); the third R(2,1), R(2,2) and R(2,3); the fourth R(3,1), R(3,2) and R(3,3); the fifth, R(1,4), R(2,4), and R(3,4); and, as discussed above, the sixth collects R(4,1) through R(4,5) and R(1,5) through R(3,5).

Accident Releases

Radiological consequences were calculated by assigning cask release fractions to each accident severity category for each chemically and physically distinct radioisotope. The *release fraction* is defined as the fraction of the radioactivity in the cask that could be released from the cask in a given severity of accident. Release fractions vary according to spent nuclear fuel type and the physical/chemical properties of the radioisotopes. Most radionuclides in spent nuclear fuel are in chemically and physically stable, solid, nondispersible forms. Gaseous radionuclides, such as krypton-85, would be released if both the fuel cladding and cask containment boundary were compromised.

The Modal Study developed release fractions for commercial spent nuclear fuel from pressurized-water reactors. These release fractions, listed in Table J-21, are based on best engineering judgment and are believed to be conservative. The analysis estimated the amount of radioactive material released from a cask in an accident by multiplying the approximate release fraction by the number of fuel assemblies in a cask (see Table J-3) and the radionuclide activity of a spent nuclear fuel assembly (see Appendix A). To provide perspective, the release fraction for a category 6 accident involving a large rail cask results in an estimated release of about 1,600 curies of cesium isotopes. For this analysis, the release fractions developed by the Modal Study were used only for commercial pressurized-water reactor fuel and spent nuclear fuel from training, research and isotope reactors built by General Atomics (commonly called *TRIGA* spent nuclear fuel), both of which are rod-type fuels. The availability of fuel-specific data for other types of spent nuclear fuel that would be shipped to the repository allowed the use of release fractions that more closely approximate expected release characteristics.

Table J-21. Fractions of selected radionuclides in commercial spent nuclear fuel projected to be released from casks in transportation accidents for cask response regions.

		Release fraction[a]				
Cask response region	Severity category	Inert gas	Iodine-129	Cesium-134, -135, -137	Ruthenium-106	Particulates
R(1,1)	1	0.0	0.0	0.0	0.0	0.0
R(1,2),R(1,3)	2	9.9×10^{-3}	7.5×10^{-5}	6.0×10^{-6}	8.1×10^{-7}	6.0×10^{-8}
R(2,1),R(2,2),R(2,3)	3	3.3×10^{-2}	2.5×10^{-4}	2.0×10^{-5}	2.7×10^{-6}	2.0×10^{-7}
R(3,1),R(3,2),R(3,3)	4	3.3×10^{-1}	2.5×10^{-3}	2.0×10^{-4}	2.7×10^{-5}	2.0×10^{-6}
R(1,4),R(2,4),R(3,4)	5	3.9×10^{-1}	4.3×10^{-3}	2.0×10^{-4}	4.8×10^{-5}	2.0×10^{-6}
R(1,5),R(2,5),R(3,5),R(4,5), R(4,1),R(4,2),R(4,3),R(4,4)	6	6.3×10^{-1}	4.3×10^{-2}	2.0×10^{-3}	4.8×10^{-4}	2.0×10^{-5}

a. Source: (DOE 1995, page I-86).

Release fractions for aluminum fuels (aluminum alloy fuel, aluminum cladding) were based on laboratory measurements and the U.S. Nuclear Regulatory Commission Modal Study (Fischer et al. 1987, all). Because of the lower melting point of aluminum compared to metals used in other metallic fuels, the aluminum fuel release fractions are considered bounding for metallic fuels (that is, Savannah River Production Reactor, Hanford N-Reactor, and Experimental Breeder Reactor-II Mark V spent nuclear fuel). Release fractions for the aluminum and other metallic fuel types are listed in Table J-22. The estimates of fractions for cask contents released in severe accidents were assumed to be independent of the type of cask.

Table J-22. Fractions of selected radionuclides in aluminum and metallic spent nuclear fuel projected to be released from casks in transportation accidents for cask response regions.[a]

		Release fraction[b]				
Cask response region	Severity category	Inert gas	Iodine-129	Cesium-134, -135, -137	Ruthenium-106	Particulates
R(1,1)	1	0.0	0.0	0.0	0.0	0.0
R(1,2),R(1,3)	2	9.9×10^{-3}	1.1×10^{-7}	3.0×10^{-8}	4.1×10^{-9}	3.0×10^{-10}
R(2,1),R(2,2),R(2,3)	3	3.3×10^{-2}	3.5×10^{-7}	1.0×10^{-7}	1.4×10^{-8}	1.0×10^{-9}
R(3,1),R(3,2),R(3,3)	4	3.3×10^{-1}	3.5×10^{-6}	1.0×10^{-6}	1.4×10^{-7}	1.0×10^{-8}
R(1,4),R(2,4),R(3,4)	5	3.9×10^{-1}	6.0×10^{-6}	1.0×10^{-6}	2.4×10^{-7}	1.0×10^{-8}
R(1,5),R(2,5),R(3,5),R(4,5), R(4,1),R(4,2), R(4,3),R(4,4)	6	6.3×10^{-1}	6.0×10^{-5}	1.0×10^{-5}	2.4×10^{-6}	1.0×10^{-7}

a. Source: DOE (1995, page I-87).
b. These release fractions are applicable to N-Reactor, Savannah River Site production reactor, and DOE research/test reactor spent nuclear fuel types.

Atmospheric Conditions

For the analyses of accident risk and consequences, releases of radioactive materials from casks during and following severe accidents were assumed to be into the atmosphere where these materials would be carried by wind. Because it is not possible to predict specific locations where transportation accidents would occur, atmospheric conditions that generally apply throughout the continental United States were used.

Table J-23 lists the frequency at which atmospheric stability and wind speed conditions occur in the contiguous United States. The data, which are averages for 177 meteorological data collection locations, were used in conjunction with the RISKIND computer program (Yuan et al. 1995, all) to develop estimates of the consequences of maximum reasonably foreseeable accidents and acts of sabotage.

In calculating estimated values for consequences, RISKIND used the atmospheric stability and wind speed data to analyze the dispersion of radioactive materials in the atmosphere that could follow releases in severe accidents. The dispersions were modeled as plumes of gases and particles. Using the results of the dispersion analysis, RISKIND calculated values for radiological consequences (population dose and dose to a maximally exposed individual). These results were placed in order from lowest to highest. Following this order, the probabilities of the atmospheric conditions associated with each set of consequences were accumulated. As the accumulated probability increased and the likelihood of an exceedance of a set of atmospheric conditions decreased, estimated consequences increased. This procedure was followed to identify the level of severe accident and sabotage consequences that would not be exceeded 50 percent and 95 percent of the time. For atmospheric conditions that are called neutral, or average, the consequences would not be exceeded 50 percent of the time. Thus, neutral atmospheric conditions would be the conditions likely to prevail during a severe accident or act of sabotage. Under stable, or quiescent, conditions the consequences would not be exceeded 95 percent of the time. The

Table J-23. Frequency of atmospheric and wind speed conditions – U.S. averages.[a]

Atmospheric stability class	Wind speed condition						
	WS(1)	WS(2)	WS(3)	WS(4)	WS(5)	WS(6)	Total
A	0.00667	0.00444	0.00000	0.00000	0.00000	0.00000	0.01111
B	0.02655	0.02550	0.01559	0.00000	0.00000	0.00000	0.06764
C	0.01400	0.02931	0.05724	0.01146	0.00122	0.00028	0.11351
D	0.03329	0.07231	0.15108	0.16790	0.03686	0.01086	0.47230
E	0.00040	0.04989	0.06899	0.00146	0.00016	0.00003	0.12093
F	0.10771	0.08710	0.00110	0.00000	0.00000	0.00000	0.19591
G	0.01713	0.00146	0.00000	0.00000	0.00000	0.00000	0.01859
F+G	0.12485	0.08856	0.00110	0.00000	0.00000	0.00000	0.21451
Totals	**0.20576**	**0.27000**	**0.29401**	**0.18082**	**0.03825**	**0.01117**	**1.00000**
Wind speed (meters per second)[b]	0.89	2.46	4.47	6.93	9.61	12.52	

a. Source: TRW (1999a, page 40).
b. To convert meters per second to miles per hour, multiply by 2.237.

analysis assumed that these conditions, which would be unlikely, would occur only for maximum reasonably foreseeable accidents that had an annual probability greater than 2 chances in 1 million in a year.

Exposure Pathways

Radiation doses were calculated for an individual who is postulated to be near the scene of an accident and for populations within 80 kilometers (50 miles) of an accident location. Doses were determined for rural, suburban, and urban population groups. Dose calculations considered a variety of exposure pathways, including inhalation and direct exposure (cloudshine and immersion in a plume of radioactive material) from a passing cloud of contaminants; ingestion from contaminated crops; direct exposure from radioactivity deposited on the ground (groundshine); and inhalation of radioactive particles resuspended by wind from the ground.

Emergency Response, Interdiction, Dose Mitigation, and Evacuation

The RADTRAN4 computer program that DOE used to estimate radiological risks includes assumptions about the postaccident remediation of radioactive material contamination of land where people live. The program assumed that, after an accident, contaminants would continue to contribute to population dose through three pathways—groundshine, inhalation of resuspended particulates, and, for accidents in rural areas, ingestion of foods produced on the contaminated lands. It also assumed that medical and other interdiction would not occur to reduce concentrations of radionuclides absorbed or deposited in human tissues as a result of accidents.

Similarly, the RISKIND (Yuan et al. 1995, all) computer program includes assumptions about response, interdiction, dose mitigation, and evacuation for calculating radiological consequences (dose to populations and maximally exposed individuals). In estimating consequences of maximum reasonably foreseeable accidents during the transportation of spent nuclear fuel and high-level radioactive waste to the repository, the analysis assumed the following:

- Populations would continue to live on contaminated land for 1 year.
- There would be no radiological dose to populations from ingestion of contaminated food. Food produced on land contaminated by a maximum reasonably foreseeable accident would be embargoed from consumption.

- Medical and other interdiction would not occur to reduce concentrations of radionuclides absorbed or deposited in human tissues as a result of an accident.

The analysis of radiological risks to populations and estimates of consequences of maximum reasonably foreseeable accidents did not explicitly address local, difficult-to-evacuate populations such as those in prisons, hospitals, nursing homes, or schools. However, the analysis addressed the potential for accidents to occur in urban areas with high population densities and used the assumptions regarding interdiction, evacuation, and other intervention actions discussed above. These assumptions encompass the consequences and risks that could arise from slowness in preventing the consequences of an accident for some population groups.

Health Risk Conversion Factors

The health risk conversion factors used to estimate expected latent cancer fatalities from radiological exposures are presented in International Commission on Radiological Protection Publication 60 (ICRP 1991, page 22). These factors are 0.0005 latent cancer fatality per person-rem for members of the public and 0.0004 latent cancer fatality per person-rem for workers. For accidents in which individuals would receive doses greater than 20 rem over a short period (high dose/high dose rate), the factors would be 0.0010 latent cancer fatality per rem for a member of the public and 0.0008 latent cancer fatality per rem for workers.

Assessment of Accident Risk

The RADTRAN4 computer code (Neuhauser and Kanipe 1992, all) was used in calculating risks from transportation of spent nuclear fuel and high-level radioactive waste. The code determined unit-risk factors (person-rem per curie) for the radionuclides of concern in the inventory being shipped. The unit-risk factors from RADTRAN4 were combined with conditional accident probabilities, state-specific accident rates, release fractions for each of the six accident severity collective categories, and state-specific food transfer factors to obtain risk per shipment for routes. The accident risks were estimated in terms of collective radiation dose to the population within 80 kilometers (50 miles).

The analysis first calculated unit risk factors for a shipment for each state through which shipments would pass. This was done for the three types of population zones in each state (using population density data from the 1990 census) and for each accident severity category. The unit risk factors used actual population densities within 800 meters (0.5 mile) of routes based on 1990 census data to estimate populations within 80 kilometers (50 miles). This yielded values for each transportation mode, for each type of impact, and for each state through which a shipment would pass. The unit risk factors for all the applicable accident severity categories were summed for each population zone for each state. Also, for the three types of population zone in a state, the lengths through areas of each type were summed for the route used in the analysis. This yielded route lengths for each population zone in each state. The sum of the route lengths and the sum of the unit risk factors for each population zone were multiplied together. This was repeated for each population zone in each state through which a shipment would pass. The results were summed to provide estimates of the accident risk for a shipment.

Estimating Consequences of Maximum Reasonably Foreseeable Accident Scenarios

In addition to analyzing the radiological and nonradiological risks that would result from the transportation of spent nuclear fuel and high-level radioactive waste to the repository, DOE assessed the consequences of maximum reasonably foreseeable accidents. This analysis provided information about the magnitude of impacts that could result from the most severe accident that could reasonably be expected to occur, although it could be highly unlikely. DOE concluded that, as a practical matter, events with a probability less than 1×10^{-7} (1 chance in 10 million) per year rarely need to be examined (DOE 1993, page 28). This would be equivalent to about once in the course of 15 billion legal-weight truck shipments. For perspective, an accident this severe in commercial truck transportation would occur about once in 50 years on U.S. highways. Thus, the analysis of maximum reasonably foreseeable accidents postulated to occur during the transportation of spent nuclear fuel and high-level radioactive waste evaluated only consequences for accidents with a probability greater than 1×10^{-7} per year. The consequences were determined for atmospheric conditions that could prevail during accidents and for physical and biological pathways that would lead to exposure of members of the public and workers to radioactive materials and ionizing radiation. The analysis used the RISKIND code (Yuan et al. 1995, all) to estimate doses for individuals and populations.

The analysis assumed maximum reasonably foreseeable accident scenarios could occur anywhere, either in rural or urbanized areas. The probability of such an accident would depend on the amount of exposure to the transportation accident environment. In this case, exposure would be the product of the cumulative shipment distance and the applicable accident rates. However, because of large differences in exposure, principally because of the large differences in the distances traveled in the two types of population areas, a severe accident scenario that might be reasonably foreseeable, in a rural area might not be reasonably foreseeable in an urbanized area. Thus, a reasonably foreseeable accident postulated to occur in a rural area (most travel would occur in rural areas) under meteorological conditions that would be exceeded (resulting in greater consequences) only 5 percent of the time, might not be reasonably foreseeable in an urbanized area where shipments would travel relatively few kilometers. For the mostly legal-weight truck and mostly rail scenarios, Table J-24 lists the probability of a severe accident during national transportation. These probabilities are for accidents that would:

- Occur in urbanized and rural areas
- Occur under median (50-percent) meteorological conditions and 95-percent conditions (95-percent conditions would be exceeded, in terms of dose consequences, only 5 percent of the time)
- Occur for accidents in collective severity categories 5 and 6 that are postulated to result in the largest releases of radioactive materials from shipping casks
- Involve rail and legal-weight truck casks

Table J-24. Annual probability of severe accidents in urbanized and rural areas – category 5 and 6 accidents, national transportation.

Scenario	Meteorologic conditions exceeded	Probability of exceeding threshold for Category 5: Annual probability for urbanized area	Probability of exceeding threshold for Category 5: Annual probability for rural area	Probability of exceeding threshold for Category 6: Annual probability for urbanized area	Probability of exceeding threshold for Category 6: Annual probability for rural area
Mostly rail					
Truck shipments	50%	4×10^{-7}[a]	2×10^{-6}	3×10^{-7}	1×10^{-6}
	95%	$\mathbf{2\times10^{-8}}$[b]	1×10^{-7}	$\mathbf{1\times10^{-8}}$	$\mathbf{7\times10^{-8}}$
Rail shipments	50%	1×10^{-5}	4×10^{-5}	3×10^{-6}	8×10^{-6}
	95%	7×10^{-7}	2×10^{-6}	2×10^{-7}	4×10^{-7}
Mostly legal-weight truck					
Truck shipments	50%	6×10^{-6}	4×10^{-5}	4×10^{-6}	2×10^{-5}
	95%	3×10^{-7}	2×10^{-6}	2×10^{-7}	1×10^{-6}
Rail shipments	50%	$\mathbf{4\times10^{-8}}$	1×10^{-6}	$\mathbf{8\times10^{-9}}$	4×10^{-7}
	95%	$\mathbf{2\times10^{-9}}$	$\mathbf{5\times10^{-8}}$	$\mathbf{4\times10^{-10}}$	$\mathbf{2\times10^{-8}}$

a. Probabilities not in bold are reasonably foreseeable.
b. Probabilities in bold would occur less than one time in 10 million and therefore are not reasonably foreseeable.

For the mostly legal-weight truck scenario, in which only naval spent nuclear fuel would be shipped by rail, the likelihood would be less than 1×10^{-7} per year for the most severe rail accident (severity category 6) to occur in an urbanized area. Thus, the highest severity rail accidents would only be reasonably foreseeable in rural areas under average (50-percent) meteorological conditions (probability greater than 1 in 10 million per year).

Table J-24 also lists the probabilities of other severe accidents the analysis considered. Under the mostly rail scenario, the most severe types of legal-weight truck accidents (collective category 6) in rural and urbanized areas under meteorological conditions that would be exceeded only 5 percent of the time would not be reasonably foreseeable.

In total, 9 sets of accident conditions defined by scenario, shipment mode, meteorology, accident severity category, and location (identified in the table by shaded cells) would not be reasonably foreseeable. Nonetheless, although the probabilities would be remote for some accidents, the RADTRAN4 analysis of radiological dose-risks (discussed above) included risk contributions of all accidents, including ones in categories 1 through 4, regardless of their probability of occurrence or consequences. Thus, the analysis addressed the contributions to risk from the spectrum of accidents that would range from low-consequence, high-probability events to high-consequence, low-probability events.

The analysis of maximum reasonably foreseeable accidents evaluated only accidents from the 23 listed in Table J-24 that would be reasonably foreseeable and that could result in maximum consequences.

From this collection of 23 possible accidents, the analysis evaluated three sets of accident conditions that were determined as those with the greatest consequences—one for the mostly rail scenario and two for the mostly legal-weight truck scenario—to identify the maximum reasonably foreseeable accident that would have the greatest consequences. The results for these cases are listed in Table J-25. Based on these results, the analysis identified one maximum reasonably foreseeable accident each for the mostly rail and mostly legal-weight truck national transportation analysis scenarios. For the mostly legal-weight truck scenario, the maximum reasonably foreseeable accident would be a severity category 6 accident involving a legal-weight truck cask in an urbanized area under stable weather (meteorological conditions that would be exceeded only about 5 percent of the time) conditions. For the mostly rail scenario, the accident would also be a category 6 accident involving a rail cask in an urbanized area under stable weather conditions.

The analysis of consequences of maximum reasonably foreseeable accidents used data from the 1990 census to estimate the size of populations in urbanized areas that could receive exposures to radioactive materials. The analysis used estimated populations in successive 8-kilometer (5-mile)-wide annular rings around the centers of the 21 large urbanized areas (cities and metropolitan areas) in the continental United States (TRW 1999a, page 22). The average population for each ring was used to form a population distribution for use in the analysis. To be conservative in estimating consequences, the analysis assumed that accidents in urbanized areas would occur at the center of the population zone, where the population density would be greatest. This assumption resulted in conservative estimates of collective dose to exposed populations.

J.1.4.2.2 *Methods and Approach for Analysis of Nonradiological Impacts of Transportation Accidents*

Nonradiological accident risks are risks of traffic fatalities. Traffic fatality rates are reported by state and Federal transportation departments as fatalities per highway vehicle- or train-kilometer traveled. The fatalities are caused by physical trauma in accidents. For nonradiological accident risks estimated in this EIS for legal-weight truck transportation, accident fatality risks were based on state-level fatality rates for Interstate Highways (Saricks and Tompkins 1999, all). Accident fatality risks for rail transportation were

Table J-25. Consequences of maximum reasonably foreseeable accidents in national transportation.

Scenario	Meteorologic conditions exceeded	Severity category 5 accidents: Consequences in urbanized area	Severity category 5 accidents: Consequences in rural area	Severity category 6 accidents: Consequences in urbanized area	Severity category 6 accidents: Consequences in rural area
Mostly rail					
Truck accident	50%	+[a]	+	+	+
	95%	--[b]	+	--	--
Rail accident	50% population dose	+	+	+	+
	50% MEI[c] dose	+	+	+	+
	95% population dose	+	+	61,000 (31)[d]	+
	95% MEI dose	+	+	26 (0.013)[e]	+
Mostly legal-weight truck					
Truck accident	50% population dose	++[f]	++	++	++
	50% MEI dose	++	++	++	++
	95% population dose	++	++	9,400 (5)	430 (0.2)
	95% MEI dose	++	++	4 (0.002)	3.9 (0.002)
Rail accident	50%	--	++	--	++
	95%	--	--	--	--

a. + = Consequences of these accidents are bounded by the rail accident in an urbanized area.
b. = probability less than 1 × 10-7 (not reasonably foreseeable).
c. MEI = maximally exposed individual.
d. Population consequence in person-rem (latent cancer fatality).
e. MEI consequences in rem (probability of increasing a latent cancer fatality).
f. ++ = Consequences of these accidents are bounded by the truck accident in an urbanized area.

also calculated using state-specific rates (Saricks and Tompkins 1999, all). Section J.2.1 discusses methods and data used to analyze accidents for barge transportation.

For truck transportation, the rates in Saricks and Tompkins (1999, Table 4) are specifically for heavy combination trucks involved in interstate commerce. Heavy combination trucks are multiaxle tractor-trailer trucks having a tractor and one to three freight trailers connected to each other. This kind of truck with a single trailer would be used to ship spent nuclear fuel and high-level radioactive waste. Truck accident rates were determined for each state based on statistics compiled by the Department of Transportation Office of Motor Carriers for 1994 through 1996. The report presents accident involvement and fatality counts, estimated kilometers of travel by state, and the corresponding average accident involvement, fatality, and injury rates for the 3 years investigated. Fatalities include crew members and all others attributed to accidents. Although escort vehicles would not be heavy combination trucks, the fatality rate data used for truck shipments of loaded and empty spent fuel casks were also used to estimate fatalities from accidents that would involve escort vehicles.

Rail accident rates were computed and presented similarly to truck accident rates, but a railcar is the unit of haulage. The state-specific rail accident involvement and fatality rates are based on statistics compiled by the Federal Railroad Administration for 1994 through 1996. Rail accident rates include both mainline accidents and those occurring in railyards (Saricks and Tompkins 1999, page 9).

The accident rates used to estimate traffic fatalities were computed using data for all interstate shipments, independent of the cargoes. Shippers and carriers of radioactive material generally have a higher-than-average awareness of transport risk and prepare cargoes and drivers accordingly (Saricks and Kvitek 1994, all). These effects were not given credit in the assessment.

J.1.4.2.3 *Data Used To Estimate Incident Rates for Rail and Motor Carrier Accidents*

In analyzing potential impacts of transporting spent nuclear fuel and high-level radioactive waste, DOE considered both incident-free transportation and transportation accidents. Potential incident-free transportation impacts would include those caused by exposing the public and workers to low levels of radiation and other hazards associated with the normal movement of spent nuclear fuel and high-level radioactive waste by truck, rail, or barge. Impacts from accidents would be those that could result from exposing the public and workers to radiation, as well as vehicle-related fatalities.

In its analysis of impacts from transportation accidents, DOE relied on data collected by the U.S. Department of Transportation and others (for example, the American Petroleum Institute) to develop estimates of accident likelihood and their ranges of severity (see Fischer et al. 1987, pages 7-25 and 7-26). Using these data, the analysis estimated that as many as 40 accidents could occur over 24 years in the course of shipping spent nuclear fuel to the repository by legal-weight trucks; 1 or 2 rail accidents that involved a railcar carrying a cask could occur if most shipments were by rail; and no accidents would be likely for the limited use of barges.

Furthermore, in using data collected by the Department of Transportation, the analysis considered the range of accidents, from slightly more than "fender benders" to high-speed crashes, that the DOE carrier would have to report in accordance with the requirements of Department of Transportation regulations. The accidents that could occur would be unlikely to be severe enough to affect the integrity of the shipping casks.

The following paragraphs discuss reporting and definitions for transportation accidents and the relationships of these to data used in analyzing transportation impacts in this EIS.

J.1.4.2.3.1 Transportation Accident Reporting and Definitions. In the United States, the reporting of transportation accidents and incidents involving trucks, railroads, and barges follows requirements specified in various Federal and state regulations.

Motor Carrier Accident Reporting and Definitions

Regulations generally require the reporting of motor carrier accidents (regardless of the cargo being carried) if there are injuries, fatalities, or property damage. These regulations have evolved through the years, mostly in response to increasing values of transportation equipment and commodities. For example, the Federal requirements in the following text box establish a functional threshold for damage to vehicles rather than a value-of-damage threshold, which was used until the 1980s. Nonetheless, many states continue to use value thresholds (for example, Ohio uses $500) for vehicle damage when documenting reportable accidents.

Until March 4, 1993, Federal regulations (49 CFR Part 394) required motor carriers to submit accident reports to the Federal Highway Administration Motor Carrier Management Information System using the so-called "50-T" reporting format. The master file compiled from the data on these reports in the Federal Highway Administration Office of Motor Carriers was the basis of accident, fatality, and injury rates developed for the 1994 study of transportation accident rates (Saricks and Kvitek 1994, all).

The Final Rule of February 2, 1993 (58 *FR* 6726, February 2, 1993), modified the carrier reporting requirement; rather than submitting reports, carriers now must maintain a register of accidents that meet the definition of an accident for 1 year after such an accident occurs. Carriers must make the contents of such a register available to Federal Highway Administration agents investigating specific accidents. They must also give "...all reasonable assistance in the investigation of any accident including providing a full, true, and correct answer to any question of inquiry" to determine if hazardous materials other than spilled

COMMERCIAL MOTOR VEHICLE ACCIDENT
(49 CFR 390.5)

An occurrence involving a commercial motor vehicle operating on a public road in interstate or intrastate commerce that results in:

- A fatality
- Bodily injury to a person who, as a result of the injury, immediately receives medical treatment away from the scene of the accident
- One or more motor vehicles incurring disabling damage as a result of the accident, requiring the motor vehicle to be transported away from the scene by a tow truck or other motor vehicle

The term accident does not include:

- An occurrence involving only boarding and alighting from a stationary motor vehicle
- An occurrence involving only the loading or unloading of cargo
- An occurrence in the course of the operation of a passenger car or a multipurpose passenger vehicle by a motor carrier and is not transporting passengers for hire or hazardous materials of a type and quantity that require the motor vehicle to be marked or placarded in accordance with 49 CFR Part 177, Subpart 823

fuel from the fuel tanks were released, and to furnish copies of all state-required accident reports [49 CFR 390.15]. The reason for this rule change was the emergence of an automated State accident reporting system compiled from law enforcement accident reports that, pursuant to provisions of the Intermodal Surface Transportation Efficiency Act of 1991 [P.L. 102-240, 105 STAT. 1914], was established under the Motor Carrier Safety Assistance Program.

Under Section 408 of Title IV of the Motor Carrier Act of 1991, a component of the Intermodal Surface Transportation Efficiency Act, the Secretary of Transportation is authorized to make grants to states to help them achieve uniform implementation of the police reporting system for truck and bus accidents recommended by the National Governors Association. Under this system, called SAFETYNET, accident data records generated by each state follow identical formatting and content instructions. They are entered in a Federally maintained SAFETYNET data base on approximately a weekly basis. The SAFETYNET data base, in turn, is compiled and managed as part of the Motor Carrier Management Information System.

Accident data compiled from the Bureau of Motor Carrier Safety (now the Office of Motor Carriers in the Federal Highway Administration), American Petroleum Institute, California Highway Patrol, and California Department of Transportation provided the basis used by the Modal Study (Fischer et al. 1987, page B-1) for estimating characteristics of accidents that might involve shipments of spent nuclear fuel using "large trucks." Although reporting requirements have changed, these data were similar to data being compiled by the SAFETYNET system for motor carrier accidents in 1999. Most important, the definition of a motor carrier accident, the basis for reporting and data compilation, has remained basically unchanged over the 40 years of data collection.

Because the Modal Study is the fundamental source for data that describes the severity of transportation accidents used in this EIS, the relative constancy of the definition of *accident* is important in establishing confidence in estimated impact results. Thus, although the transportation environment has changed over the 40 years of data collection, the constancy of the definition of *accident* tends to provide confidence that the distribution of severity for reported accidents has remained relatively the same. That is, low-consequence, fender-bender accidents are the most common, high-consequence, highly energetic accidents are rare, and the proportions of these have remained roughly the same.

Changes in the transportation environment, such as changes in speed limits and safety technology, tend to change the accident rate (accidents per vehicle-kilometer of travel). Overall, however, given that the definition of *accident* does not change, such changes do not greatly affect the distribution of accident severities. For example, recent increases in speed limits from 105 to 121 kilometers (65 to 75 miles) per hour represent about a 25-percent increase in the maximum mechanical energy of vehicles. Other information aside, this increase could lead to the conclusion that the resulting distribution of accidents would show an increase for the most severe accidents in comparison to minor accidents. However, the speed limit increases do not represent a corresponding increase in actual traffic speeds, and would be unlikely to change the distribution of velocities and, thus, mechanical energies, of severe accidents from those reported in the Modal Study. These velocities ranged to faster than 137 kilometers (85 miles) per hour, even though at the time the National speed limit was 89 kilometers (55 miles) per hour.

Rail Carrier Accident Reporting and Definitions

As with regulations governing the reporting of motor carrier accidents, Federal Railroad Administration regulations generally require the reporting of accidents if there are injuries, fatalities, or property damage. These regulations have evolved through the years, mostly in response to increasing values of transportation equipment and commodities. For example, the Federal requirements in the following text box establish a value-based reporting threshold for damage to vehicles; the value has been indexed to inflation since 1975.

RAILROAD ACCIDENT/INCIDENT
(49 CFR 225.11)

- An impact between railroad on-track equipment and an automobile, bus, truck, motorcycle, bicycle, farm vehicle or pedestrian at a highway-rail grade crossing
- A collision, derailment, fire, explosion, act of God, or other event involving operation of railroad on-track equipment (standing or moving) that results in reportable damages greater than the current reporting threshold to railroad on-track equipment, signals, track, track structures, and roadbed
- An event arising from the operation of a railroad which results in:
  - Death to any person
  - Injury to any person that requires medical treatment
  - Injury to a railroad employee that results in:
- A day away from work
- Restricted work activity or job transfer
- Loss of consciousness
- Occupational illness

Rail carriers covered by these requirements must fulfill several bookkeeping tasks. The Federal Railroad Administration requires the submittal of a monthly status report, even if there were no reportable events during the period. This report must include accidents and incidents, and certain types of incidents require immediate telephone notification. Logs of reportable injuries and on-track incidents must be maintained by the railroads on which they occur, and a listing of such events must be posted and made available to employees and to the Federal Railroad Administration, along with required records and reports, on request. The data entries extracted from the reporting format are consolidated into an accident/incident data base that separates reportable *accidents* from grade-crossing *incidents*. These are processed annually into event, fatality, and injury count tables in the Federal Railroad Administration's *Accident/Incident*

Bulletin (Saricks and Tompkins 1999, all), which the Office of Safety publishes on the Internet (*http://safetydata.fra.dot.gov/officeofsafety/Prelim/1999/r01.htm*).

In contrast to the regulations for motor carriers discussed above, the Federal Railroad Administration regulations cited above call for the reporting of accidents and incidents. According to the Modal Study, the Administration defines an *accident* as "any event involving on-track railroad equipment that results in damage to the railroad on-track equipment, signals, track, or track structure, and roadbed at or exceeding the dollar damage threshold." Train *incidents* are defined as "events involving on-track railroad equipment [and non-train incidents arising from the operation of a railroad] that result in the reportable death and/or injury or illness of one or more persons, but do not result in damage at or beyond the damage threshold." The Modal Study, because "damage to casks containing spent nuclear fuel will necessarily involve severe accidents" (hence, substantial damage), used only "train accidents" to form the basis for developing the conditional probabilities of accident severities.

As with motor carrier operations, the constancy of the definition of a train accident is important in establishing confidence in the impact. For rail accidents the transportation environment has not changed dramatically over the years of data collection, and the definition of *accident* has remained essentially unchanged (with adjustments for inflation). The constancy of the definition provides confidence that the distribution of severity for reported accidents has remained relatively the same—low-consequence, limited-damage accidents are the most common and high-consequence, highly energetic accidents are rare, and their proportions have remained about the same. Changes in the rail transportation environment, as in safety and operations technology (for example, shelf-type couplers and tankcar head protection), have resulted in lower accident rates (per railcar-kilometer of travel) and, in some cases, less severe accidents. However, because the definition of *accident* has not changed appreciably, the changes that have occurred are not the kind that would greatly affect the relative proportions of minor and severe accidents.

Reporting and Definitions for Marine Casualties and Incidents

As with the regulations governing the reporting of motor carrier and rail accidents, U.S. law (46 USC 6101-6103) requires operators to report marine casualties and incidents if there are injuries, fatalities, or property damage. In addition, the law requires the reporting of significant harm to the environment.

MARINE CASUALTY AND INCIDENT
(46 USC 6101-6103)

Criteria have been established for the required reporting (by vessel operators and owners) of marine casualties and incidents involving all United States flag vessels occurring anywhere in the world and any foreign flag vessel operating on waters subject to the jurisdiction of the United States. An incident must be reported within five days if it results in:

- The death of an individual
- Serious injury to an individual
- "Material" loss of property (threshold not specified; previously was $25,000)
- Material damage affecting the seaworthiness or efficiency of the vessel
- Significant harm to the environment

The states collect casualty data for incidents occurring in navigable waterways within their borders, and there is a uniform state marine casualty reporting system for transmitting these reports to Federal jurisdiction (the U. S. Coast Guard). Coast Guard Headquarters receives quarterly extracts of the Marine

Safety Information System developed from these sources. This system is a network data base into which Coast Guard investigators enter cases at each marine safety unit. The analysis uses a Relational Database Management System. The Coast Guard Office of Investigations and Analysis compiles and processes the casualty reports into the formats and partitioned data sets that comprise the Marine Safety Information System data base, which includes maritime accidents, fatalities, injuries, and pollution spills dating to 1941 (however, the file is complete only from about 1991 to the present).

Hazardous Material Transportation Accident and Incident Reporting and Definitions

Radioactive material is a subset of the more general term *hazardous material,* which includes commodities such as gasoline and chemical products. The U.S. Department of Transportation Office of Hazardous Materials estimates that there are more than 800,000 hazardous materials shipments per day, of which about 7,700 shipments contain radioactive materials.

Hazardous materials transportation regulations (49 CFR 171) contain no distinction between an *accident* and an *incident*, and *incident* is the term used to describe situations that must be reported. Hazardous materials regulations (49 CFR 171.15) require the reporting of incidents if:

- A person is killed
- A person receives injuries requiring hospitalization
- The estimated property damage is greater than $50,000
- An evacuation of the public occurs lasting one or more hours
- One or more major transportation arteries are closed or shutdown for one or more hours
- The operational flight pattern or routine of an aircraft is altered
- Fire, breakage, spillage, or suspected radioactive contamination occurs involving shipment of radioactive material
- Fire, breakage, spillage, or suspected contamination occurs involving shipment of infectious agents
- There has been a release of a marine pollutant in a quantity exceeding 450 liters (about 120 gallons) for liquids or 400 kilograms (about 880 pounds) for solids
- There is a situation that, in the judgement of the carrier, should be reported to the U.S. Department of Transportation even though it does not meet the above criteria

These criteria apply to loading, unloading, and temporary storage, as well as to transportation. The criteria involving infectious agents or aircraft are unlikely to be used for spent nuclear fuel or high-level radioactive waste shipments. Based on these criteria, reportable motor vehicle and rail transportation situations are far more exclusionary than hazardous material situations.

Carriers (not law enforcement officials) are required to report hazardous materials incidents to the U.S. Department of Transportation. These reports are compiled in the Hazardous Materials Incident Report data base. In addition, U.S. Nuclear Regulatory Commission regulations (20 CFR 20.2201, 20.2202, 20.2203) require the reporting of a loss of radioactive materials, exposure to radiation, or release of radioactive materials.

Sandia National Laboratories maintains the Radioactive Materials Incident Report (RMIR) data base, which contains incident reports from the Hazardous Materials Incident Report data base that involve radioactive material. In addition, RMIR contains data from the U.S. Nuclear Regulatory Commission, state radiation control offices, the DOE Unusual Occurrence Report data base, and media coverage of radioactive materials transportation incidents. DOE (1995, pages I-117) and McClure and Fagan (1998, all) discuss historic incidents involving spent nuclear fuel that are reported in RMIR as well as incidents that took place prior to the existence of this data base. RMIR characterizes incidents in three categories: transportation accidents, handling accidents, and reported incidents. However, the definitions of these categories are not consistent with the definitions used in other U.S. Department of Transportation data bases. For example, from 1971 through 1998, RMIR lists one transportation accident involving a loaded rail shipment of spent nuclear fuel. However, based on current Federal Railroad Administration reporting requirements, this occurrence probably would be listed as a grade-crossing incident, not an accident. For this reason and because of the small number of occurrences in the data base involving spent nuclear fuel, the EIS analysis did not use RMIR to estimate transportation accident rates.

J.1.4.2.3.2 Accident Rates for Transportation by Heavy-Combination Truck, Railcar, and Barge in the United States. Saricks and Tompkins (1999, all) developed estimates of accident rates for heavy-combination trucks, railcars, and barges based on data available for 1994 through 1996. The estimates provide an update for accident rates published in 1994 (Saricks and Kvitek 1994, all) that reflected rates from almost a decade earlier.

Rates for Accidents in Interstate Commerce for Heavy-Combination Trucks

Saricks and Tompkins (1999, all) developed basic descriptive statistics for state-specific rates of accidents involving interstate-registered combination trucks for 1994, 1995, and 1996. The accident rate over all road types for 1994 was 2.98×10^{-7} accident per truck-kilometer (Saricks and Tompkins, 1999, Table 3a); for 1995 it was 2.97×10^{-7} accident per truck-kilometer (Saricks and Tompkins, 1999, Table 3b); and for 1996 it was 3.46×10^{-7} accident per truck-kilometer (Saricks and Tompkins, 1999, Table 3c). The composite mean from 1994 through 1996 was 3.21×10^{-7} accident per truck-kilometer.

During the 24 years of the Proposed Action, the *mostly legal-weight truck* national transportation scenario would involve as many as 50,000 truck shipments of spent nuclear fuel and high-level radioactive waste. Based on the data in Saricks and Tompkins (1999, Table 4), the transportation analysis estimated that those shipments could involve as many as 40 accidents. During the same period, the *mostly rail* scenario would involve about 2,600 truck shipments, and the analysis estimated that as many as two accidents could occur during these shipments. More than 99 percent of these accidents would not generate forces capable of causing functional damage to the casks, and would have no radiological consequences. A small fraction of the accidents could generate forces capable of damaging the cask.

Rates for Freight Railcar Accidents

Results for accident rates for freight railcar shipments from Saricks and Tompkins (1999, all), show that domestic rail freight accidents, fatalities, and injuries on Class 1 and 2 railroads have remained stable or declined slightly since the late 1980s. Based on data from 1994 through 1996, these rates are 5.39×10^{-8}, 8.64×10^{-8}, and 1.05×10^{-8} per railcar-kilometer, respectively (Saricks and Tompkins, 1999, Table 6). This conclusion is based on applying denominators that do *not* include train and car kilometers for intermodal shipments (containers and trailers-on-flatcar) not loaded by the carriers themselves. Thus, the actual denominators are probably higher and the rates consequently lower, by about 20 percent.

During the 24 years of the Proposed Action, the *mostly rail* national transportation scenario would involve as many as 11,000 rail shipments of spent nuclear fuel and high-level radioactive waste. Based on the data in Saricks and Tompkins (1999, Table 6), the analysis estimated that these shipments could involve one or two accidents. More than 99 percent of these accidents would not generate forces capable of causing functional damage to the cask; these accidents would have no radiological consequences. A small fraction of the accidents could generate forces capable of damaging the cask. For the *mostly legal-weight truck* scenario, rail accidents would be unlikely during the 300 railcar shipments of naval spent nuclear fuel.

Rates for Barge Accidents

Waterway results show a general improvement over mid-1980s rates. The respective rates for 450-metric-ton (500-ton) shipments for waters internal to the coast (rivers, lakes, canals, etc.) for accident and incident involvements and fatalities were 1.68×10^{-6} and 8.76×10^{-9} per shipment-kilometer, respectively (Saricks and Tompkins 1999, Table 8b). Rates for lake shipping were lower—2.58×10^{-7} and 0 per shipment-kilometer, for accidents and incidents and for fatalities, respectively. Coastal casualty involvement rates have risen in comparison to the data recorded about 10 years ago, and are comparable to rates for internal waters—5.29×10^{-7} and 8.76×10^{-9} per shipment-kilometer (Saricks and Tompkins 1999, Table 9b).

During the 24 years of the Proposed Action, the *mostly rail* national transportation scenario could involve the use of barges to ship spent nuclear fuel from 14 commercial sites. Based on the data in Saricks and Tompkins (1999, all), the analysis estimated that less than one accident could occur during such shipments. A barge accident severe enough to cause measurable damage to a shipping cask would be highly unlikely.

Rates for Safe Secure Trailer Accidents

DOE uses safe secure trailers to transport hazardous cargoes in the continental United States. The criteria used for reporting accidents involving these trailers are damage in excess of $500, a fire, a fatality, or damage sufficient for the trailer to be towed. From 1975 through 1998, 14 accidents involved safe secure trailers over about 54 million kilometers (about 34 million miles) of travel, which yields a rate of 2.6×10^{-7} accident per kilometer (4.2×10^{-7} per mile). This rate is comparable to the rate estimated by Saricks and Tompkins (1999, Table 4) for heavy combination trucks, 3.2×10^{-7} accident per kilometer (5.1×10^{-7} per mile).

J.1.4.2.3.3 Accident Data Provided by the States of Nevada, California, South Carolina, Illinois, and Nebraska. In May 1998, DOE requested the 48 contiguous states to provide truck and rail transportation accident data for use in this EIS. Five states responded – Nevada, California, Illinois, Nebraska, and South Carolina (Denison 1998, all; Caltrans 1997, all; Wort 1998, all; Kohles 1998, all; SCDPS 1997, all). No states provided rail information.

- *Nevada.* Nevada provided a highway accident rate of 1.1×10^{-6} accident per kilometer (1.8×10^{-6} per mile) for interstate carriers over all road types. This is higher than the accident rate estimated by Saricks and Tompkins (1999, Table 4); 2.5×10^{-7} accident per kilometer (3.9×10^{-7} per mile) for heavy trucks over all road types in Nevada from 1994 to 1996.

  The definition of *accident* used in Saricks and Tompkins (1999, page 4) is the Federal definition (fatality, injury, or tow-away); in Nevada the accident criteria are fatality, injury, or $750 property damage. Based on national data from the U.S. Department of Transportation Office of Motor Carrier Information Analysis (FHWA 1997, page 2; FHWA 1998, pages 1 and 2), using the Federal definition would reduce the accident rate from 1.1×10^{-6} to about 4.1×10^{-7} accident per kilometer (1.8×10^{-6} to 6.7×10^{-7} per mile). The radiological accident risk in Nevada for the mostly legal-weight truck scenario would increase over 24 years from 0.0002 latent cancer fatality to about 0.0005 latent cancer fatality (a likelihood of 5 in 10,000 of one latent cancer fatality) if the accident rate reported by Saricks and Tompkins for Nevada were replaced by the rate of 4.1×10^{-7} per kilometer. Thus, the impacts of the rate for accidents involving large trucks on Nevada highways reported by Nevada (Denison 1998, all) would be comparable to the impacts derived using rate estimated by Saricks and Tompkins.

- *California.* California responded with highway accident rates that included all vehicles (cars, buses, and trucks). The accident rate for Interstate highways was 4.2×10^{-7} accident per kilometer (6.8×10^{-7} per mile) for all vehicles in 1996. This rate is higher than the accident rate estimated by Saricks and Tompkins (1999, Table 4), 1.6×10^{-7} accident per kilometer (2.6×10^{-7} per mile) for heavy trucks on California interstate highways from 1994 to 1996.

  The definition of *accident* in Saricks and Tompkins (1999, page 4) is the Federal definition (fatality, injury, or tow-away); in California the accident criteria are fatality, injury, or $500 property damage. Based on national data from FHWA (1997, page 2) and FHWA (1998, pages 1 and 2), using the Federal definition would reduce the accident rate from 4.2×10^{-7} to about 1.6×10^{-7} accident per kilometer (6.8×10^{-7} to 2.6×10^{-7} per mile). In addition, the rate provided by California was for all vehicles. Based on national data from the U.S. Department of Transportation Bureau of Transportation Statistics, using the accident rate for large trucks would reduce the all-vehicle accident rate from 1.6×10^{-7} to about 1.3×10^{-7} accident per kilometer (2.6×10^{-7} to 2.1×10^{-7} per mile) for large trucks. This rate is slightly less than the rate estimated by Saricks and Tompkins (1999, Table 4), 1.6×10^{-7} accident per kilometer.

- *Illinois.* Illinois provided highway data for semi-trucks from 1991 through 1995 over all road types. Over this period, the accident rate was 1.8×10^{-6} accident per kilometer (2.9×10^{-6} per mile). From 1994 through 1996, Saricks and Tompkins (1999, all) estimated an accident rate of 3.0×10^{-7} accident per kilometer (4.8×10^{-7} per mile) for heavy trucks over all road types in Illinois.

  The definition of *accident* used in Saricks and Tompkins (1999, page 4) is the Federal definition (fatality, injury, or tow-away); in Illinois the accident criteria are fatality, injury, or $500 property damage. Based on national data from the U.S. Department of Transportation Office of Motor Carrier Information Analysis (FHWA 1997, page 2; FHWA 1998, pages 1 and 2), using the Federal definition would reduce the accident rate from 1.8×10^{-6} to about 6.7×10^{-7} accident per kilometer (2.9×10^{-6} to 1.1×10^{-6} per mile). This rate is comparable to the rate estimated by Saricks and Tompkins (1999, all).

- *Nebraska.* Nebraska provided a highway accident rate of 2.4×10^{-7} accident per kilometer (3.8×10^{-7} per mile) for 1997. Nebraska did not specify if the rate was for interstate highways, but it is for interstate truck carriers. This rate is slightly less than the accident rate estimated by Saricks and Tompkins (1999, all) for Nebraska interstates, 3.2×10^{-7} accident per kilometer (5.1×10^{-7} per mile) for heavy trucks from 1994 through 1996.

- *South Carolina.* South Carolina responded with highway accident rates that included all types of tractor/trailers (for example, mobile homes, semi-trailers, utility trailers, farm trailers, trailers with boats, camper trailers, towed motor homes, petroleum tankers, lowboy trailers, auto carrier trailers, flatbed trailers, and twin trailers). The rate was 8.3×10^{-7} accident per kilometer (1.3×10^{-6} per mile), for all road types. [This is higher than the accident rate estimated by Saricks and Tompkins (1999, all), 4.7×10^{-7} accident per kilometer (7.6×10^{-7} per mile) for heavy trucks on all road types in South Carolina from 1994 through 1996].

  The definition of *accident* in Saricks and Tompkins (1999, page 4) is the Federal definition (fatality, injury, or tow-away); in South Carolina the accident criteria are fatality, injury, or $1,000 property damage. Based on national data from the U.S. Department of Transportation Office of Motor Carrier Information Analysis (FHWA 1997, page 2; FHWA 1998, pages 1 and 2), using the Federal definition of an accident would reduce the accident rate from 8.3×10^{-7} to about 3.1×10^{-7} accident per kilometer (1.3×10^{-6} to 5.0×10^{-7} per mile), which is slightly less than the rate estimated by Saricks and Tompkins (1999, all), 4.7×10^{-7} accident per kilometer (7.6×10^{-7} per mile). In addition, the accident rate estimated by Saricks and Tompkins (1999, all) was based on Motor Carrier Management Information System vehicle configuration codes 4 through 8 (truck/trailer, bobtail, tractor/semi-trailer, tractor/double, and tractor/triple), while the rate obtained from South Carolina included all truck/trailer combinations. Including all of the combinations tends to increase accident rates; for example, light trucks have higher accident rates than heavy trucks (BTS 1999, Table 3-22).

DOE evaluated the effect of using the data provided by the five states on radiological accident risk for the mostly legal-weight truck national transportation scenario. If the data used in the analysis for the five states (Saricks and Tompkins 1999, Table 4) were replaced by the data provided by the states with the adjustments discussed, the change in the resulting estimate of radiological accident risk would be small, increasing from 0.067 to 0.071 latent cancer fatality. Using the unadjusted data provided by those states would result in an increase in accident risk from 0.067 to 0.093 latent cancer fatality.

J.1.4.2.4 *Transportation Accidents Involving Nonradioactive Hazardous Materials*

The analysis of impacts of transportation accidents involving the transport of nonradioactive hazardous materials to and from Yucca Mountain used information presented in two U.S. Department of Transportation reports (DOT 1998b, Table 1; BTS 1996, page 43) on the annual number of hazardous materials shipments in the United States and the number of deaths caused by hazardous cargoes in 1995. In total, there are about 300 million annual shipments of hazardous materials; only a small fraction involve radioactive materials. In 1995, 6 fatalities occurred because of hazardous cargoes. These data suggest a rate of 2 fatalities per 100 million shipments of hazardous materials. DOE anticipates about 40,000 shipments of nonradioactive hazardous materials (including diesel fuel and laboratory and industrial chemicals) to and from the Yucca Mountain site during construction, operation and monitoring, and closure of the repository. Assuming that the rate for fatalities applies to the transportation of nonradioactive hazardous materials to and from Yucca Mountain, DOE does not expect fatalities from 40,000 shipments of these materials.

J.2 Evaluation of Rail and Intermodal Transportation Options

DOE could use several modes of transportation to ship spent nuclear fuel from the 77 sites. Legal-weight trucks could be used to transport spent nuclear fuel and high-level radioactive waste contained in truck casks that would weigh approximately 22,500 kilograms (25 tons) when loaded. For sites served by railroads, rail casks placed on railcars could be used to ship directly to the Yucca Mountain site if a branch rail line was constructed in Nevada or to ship to an intermodal transfer station in Nevada if heavy-haul trucks were used.

For sites not served by a railroad that nonetheless have the capability to load rail casks, DOE could use heavy-haul trucks or, for sites located on navigable waterways, barges to transport the casks between the generating sites and nearby railheads.

For rail shipments, DOE could request the railroads provide dedicated trains to transport casks from sites to a destination in Nevada or could deliver railcars with loaded casks to the railroads as general freight for delivery in Nevada.

J.2.1 IMPACTS OF THE SHIPMENT OF COMMERCIAL SPENT NUCLEAR FUEL BY BARGE AND HEAVY-HAUL TRUCK FROM 19 SITES NOT SERVED BY A RAILROAD

An alternative to truck or rail transport of commercial spent nuclear fuel, barge transportation, was evaluated. Nineteen commercial sites that have the capability to handle and load rail casks are not served by a railroad. Accordingly, under the mostly rail transportation scenario the 19 sites were assumed to use heavy-haul trucks to move the rail casks to nearby railheads. However, because 14 of the sites are on navigable waterways (see Figure J-9), some could use barges to ship to nearby railheads. The following sections present the analysis of impacts of using barges and compares these impacts from one of the fourteen sites located on a navigable waterway (Turkey Point) to the impacts based on the use of heavy-haul trucks and legal-weight truck. The analysis assumed that all five of the DOE sites would have railroad service.

Unlike previous sections, where impacts were presented for all shipments by mode (mostly legal-weight truck and mostly rail), impacts are reported on a per shipment basis and compared on that basis to shipments via heavy-haul truck and legal-weight truck for the same reactor site.

J.2.1.1 Routes for Barges and Heavy-Haul Trucks

The heavy-haul truck-to-railhead distances for the 19 sites range from about 6 to 75 kilometers (4 to 47 miles). Routing for heavy-haul trucks was estimated using the HIGHWAY computer code (Johnson et al. 1993a, all). The INTERLINE computer code (Johnson et al. 1993b, all) was used to generate route-specific distances that would be traveled by barges. The resulting estimates for route lengths for barges and heavy-haul trucks are listed in Table J-26. Table J-27 lists the number of shipments from each site.

J.2.1.2 Analysis of Incident-Free Impacts for Barge and Heavy-Haul Truck Transportation

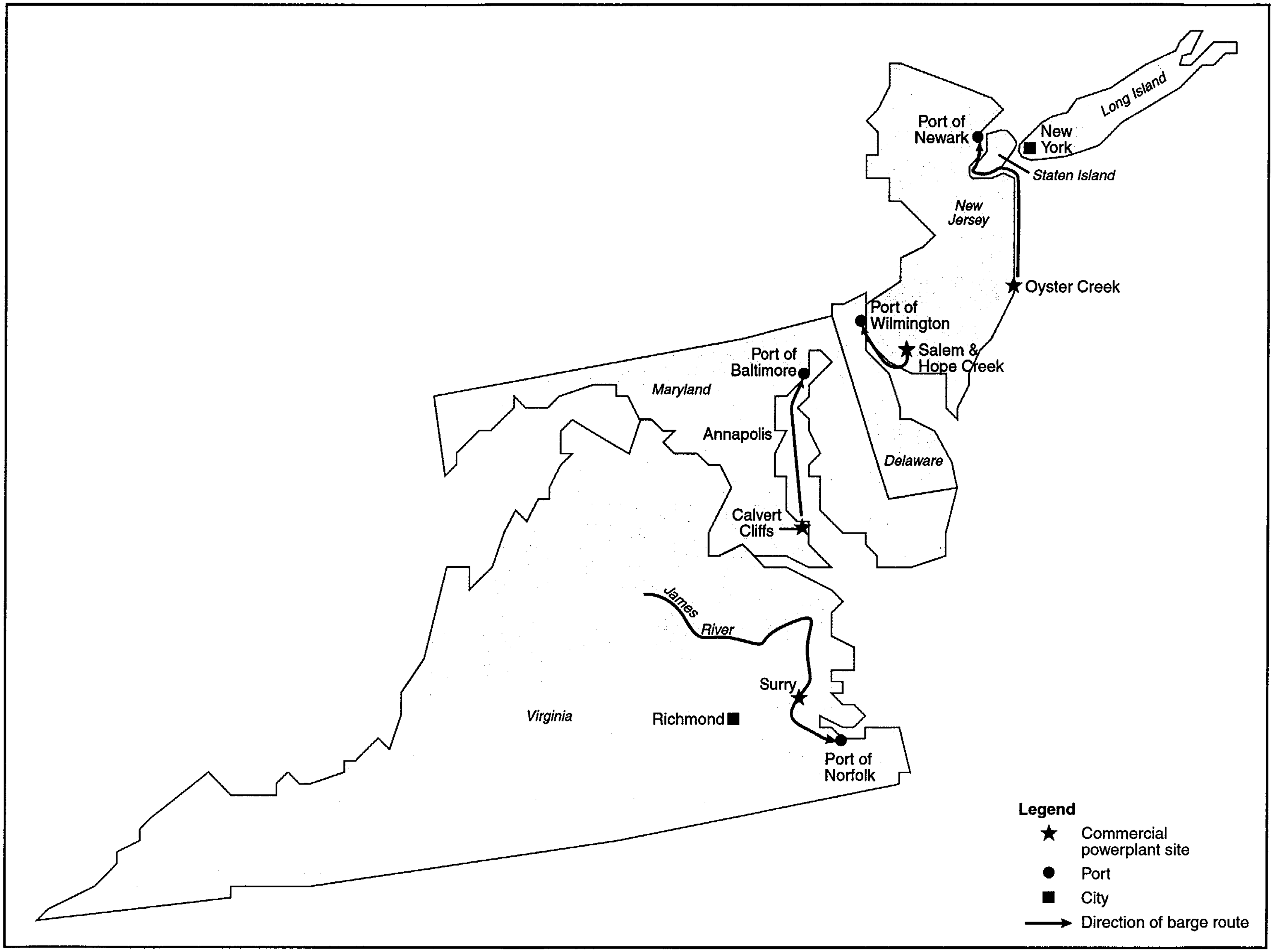
J.2.1.2.1 *Radiological Impacts of Incident-Free Transportation*

This section compares the radiological and nonradiological impacts to populations and maximally exposed individuals of incident-free transportation of spent nuclear fuel from one commercial spent nuclear fuel site (Turkey Point) for:

- Shipments using heavy-haul trucks to the nearest railhead and then to the Nevada Caliente node by rail and finally to the Yucca Mountain site by rail using the Caliente-Chalk Mountain corridor.
- Shipments using barge to a nearby railhead (Port of Miami for the Turkey Point site) and then to the Nevada Caliente node by rail and finally to the Yucca Mountain site by rail using the Caliente-Chalk Mountain corridor.
- Shipments using legal-weight trucks to the Yucca Mountain site.

The radiological impacts of intermodal transfers at the interchange from heavy-haul trucks to railcars or barges to railcars were included in the analysis. Workers would be exposed to radiation from casks during transfer operations. However, because the transfers would occur in terminals and berths that are remote from public access, public exposures would be small. Impacts of constructing intermodal transfer facilities were not included because intermodal transfers were assumed to take place at existing facilities.

The analysis assumed that heavy-haul trucks, though they would be slower moving vehicles, would result in the same types of impacts as, although somewhat higher than, an equal number of legal-weight truck shipments over the same routes. Because travel distances to nearby railheads would be short, impacts of



Fi J-9. Routes for barges from sites to nearby railheads (page 1).

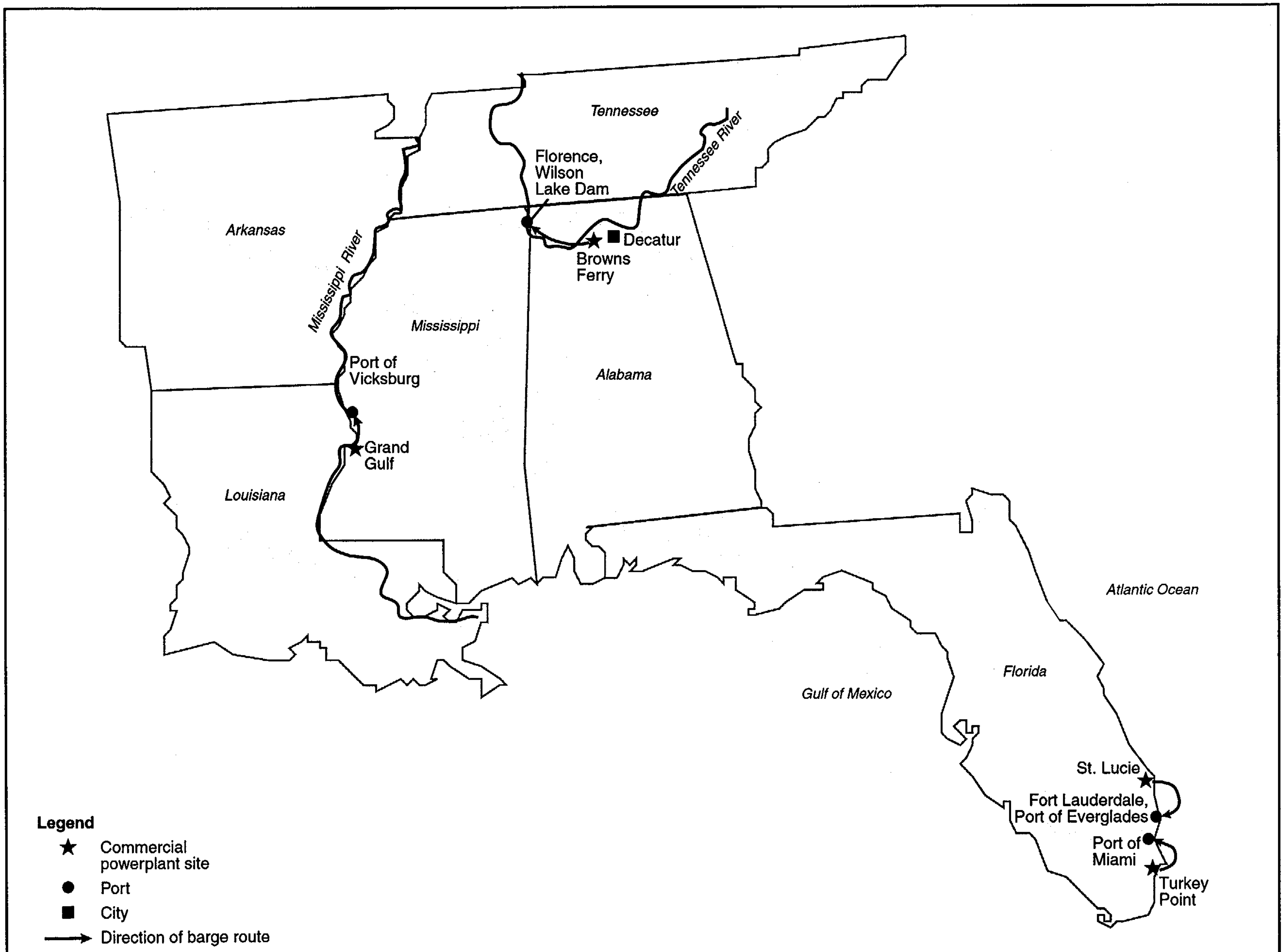


Figure J-9. Routes for barges from sites to nearby railheads (page 2 of 3).

Figure J-9. Routes for barges from sites to nearby railheads (page 3 of 3).

Table J-26. National transportation distances from commercial sites to Nevada ending rail nodes (kilometers)[a,b] (page 1 of 2).

Site (intermodal rail node)[c]	State	Destination	Rail transportation Total[d]	Rural	Suburban	Urban	Barge transportation Total[d]	Rural	Suburban	Urban
Browns Ferry NP[e]	AL	Apex	3,596	3,269	281	46	57	52	5	0
		Caliente	3,423	3,095	281	46	57	52	5	0
		Beowawe	3,278	2,990	254	34	57	52	5	0
		Jean	3,678	3,333	293	51	57	52	5	0
Diablo Canyon NP	CA	Apex	644	420	124	100	143	143	0	0
		Caliente	817	594	124	100	143	143	0	0
		Beowawe	1,439	1,005	291	141	143	143	0	0
		Jean	562	355	112	94	143	143	0	0
St. Lucie NP	FL	Apex	5,203	4,293	812	97	140	50	52	39
		Caliente	5,029	4,119	812	97	140	50	52	39
		Beowawe	4,885	4,014	784	86	140	50	52	39
		Jean	5,284	4,358	823	103	140	50	52	39
Turkey Point NP	FL	Apex	5,245	4,296	820	127	54	53	0	1
		Caliente	5,071	4,123	820	127	54	53	0	1
		Beowawe	4,927	4,017	793	116	54	53	0	1
		Jean	5,326	4,361	832	133	54	53	0	1
Calvert Cliffs NP	MD	Apex	4,344	3,558	645	140	99	98	2	0
		Caliente	4,170	3,385	645	140	99	98	2	0
		Beowawe	4,026	3,279	618	129	99	98	2	0
		Jean	4,425	3,623	657	145	99	98	2	0
Palisades NP	MI	Apex	3,375	2,895	391	90	256	256	0	0
		Caliente	3,202	2,722	391	90	256	256	0	0
		Beowawe	3,058	2,616	363	78	256	256	0	0
		Jean	3,457	2,960	402	95	256	256	0	0
Grand Gulf NP	MS	Apex	3,686	3,355	291	39	51	51	0	0
		Caliente	3,512	3,181	291	39	51	51	0	0
		Beowawe	3,368	3,076	264	28	51	51	0	0
		Jean	3,767	3,419	303	44	51	51	0	0
Cooper NP	NE	Apex	2,345	2,193	119	33	117	100	16	1
		Caliente	2,171	2,020	119	33	117	100	16	1
		Beowawe	2,027	1,914	92	21	117	100	16	1
		Jean	2,426	2,258	130	38	117	100	16	1
Salem/Hope Creek NP	NJ	Apex	4,423	3,410	818	194	30	30	0	0
		Caliente	4,250	3,236	818	194	30	30	0	0
		Beowawe	4,106	3,131	791	183	30	30	0	0
		Jean	4,505	3,475	830	200	30	30	0	0
Oyster Creek NP	NJ	Apex	4,532	3,371	933	227	130	77	36	17
		Caliente	4,358	3,198	933	227	130	77	36	17
		Beowawe	4,214	3,092	906	216	130	77	36	17
		Jean	4,613	3,436	944	232	130	77	36	17
Surry NP	VA	Apex	4,583	3,982	532	68	71	60	8	3
		Caliente	4,409	3,809	532	68	71	60	8	3
		Beowawe	4,265	3,703	505	57	71	60	8	3
		Jean	4,664	4,047	544	73	71	60	8	3
Kewaunee NP	WI	Apex	3,180	2,789	312	79	293	285	2	7
		Caliente	3,007	2,616	312	79	293	285	2	7
		Beowawe	2,863	2,510	285	68	293	285	2	7
		Jean	3,262	2,854	323	84	293	285	2	7
Point Beach NP	WI	Apex	3,180	2,789	312	79	301	293	2	7
		Caliente	3,007	2,616	312	79	301	293	2	7
		Beowawe	2,863	2,510	285	68	301	293	2	7
		Jean	3,262	2,854	323	84	301	293	2	7
Callaway NP HH – 18.5 kilometers	MO	Apex	2,796	2,625	140	31	--[f]	--	--	--
		Caliente	2,624	2,452	140	31	--	--	--	--
		Beowawe	2,491	2,358	113	20	--	--	--	--
		Jean	2,878	2,689	151	37	--	--	--	--
Fort Calhoun NP HH – 6.0 kilometers	NE	Apex	2,301	2,177	102	21	--	--	--	--
		Caliente	2,129	2,005	102	21	--	--	--	--
		Beowawe	1,996	1,911	75	10	--	--	--	--
		Jean	2,383	2,242	114	27	--	--	--	--

Table J-26. National transportation distances from commercial sites to Nevada ending rail nodes (kilometers)[a,b] (page 2 of 2).

Site (intermodal rail node)[c]	State	Destination	Rail transportation Total[d]	Rural	Suburban	Urban	Barge transportation Total[d]	Rural	Suburban	Urban
Peach Bottom NP[e]	PA	Apex	4,294	3,324	779	191	--[f]	--	--	--
HH – 58.9 kilometers		Caliente	4,121	3,151	779	191	--	--	--	--
		Beowawe	3,988	3,057	752	179	--	--	--	--
		Jean	4,375	3,388	790	196	--	--	--	--
Oconee NP	SC	Apex	4,247	3,651	534	61	--	--	--	--
HH – 17.5 kilometers		Caliente	4,074	3,479	534	61	--	--	--	--
		Beowawe	3,941	3,385	507	50	--	--	--	--
		Jean	4,328	3,716	546	66	--	--	--	--

a. To convert kilometers to miles, multiply by 0.62137.
b. Distances estimated using INTERLINE computer program.
c. Intermodal rail nodes selected for purpose of analysis. Source: TRW (1999a, all).
d. Totals might differ from sums of rural, suburban, and urban distances due to method of calculation and rounding.
e. NP = nuclear plant.
f. -- = the four sites that are not located on a navigable waterway.

Table J-27. Barge shipments and ports.

Plant name	State	Number of shipments: Proposed Action	Number of shipments: Modules 1 and 2	Barge ports assumed for barge-to-rail intermodal transfer
Browns Ferry 1	AL	176	253	Wilson L/D
Browns Ferry 3	AL	67	114	Wilson L/D
Diablo Canyon 1	CA	64	129	Port Huememe
Diablo Canyon 2	CA	59	149	Port Huememe
St. Lucie 2	FL	56	103	Port Everglades
Turkey Point 3	FL	56	80	Port of Miami
Turkey Point 4	FL	57	89	Port of Miami
Calvert Cliffs 1	MD	144	204	Port of Baltimore
Palisades	MI	70	70	Port of Muskegan
Grand Gulf 1	MS	79	154	Port of Vicksburg
Cooper Station	NE	103	159	Port of Omaha
Hope Creek	NJ	59	146	Port of Wilmington
Oyster Creek 1	NJ	87	87	Port of Newark
Salem 1	NJ	63	104	Port of Wilmington
Salem 2	NJ	57	112	Port of Wilmington
Surry 1	VA	102	128	Port of Norfolk
Kewaunee	WI	57	70	Port of Milwaukee
Point Beach 1	WI	90	102	Port of Milwaukee
Totals		**1,833**	**2,970**	

heavy-haul truck transportation would be much less than the impacts of national rail shipments. The analysis of impacts for barge shipments assumed the transport would employ commercial vessels operated by maritime carriers on navigable waterways and that these shipments would follow direct routing from the sites to nearby railheads. For both modes, intermodal transfers would be necessary to transfer rail casks to railcars.

Radiological impacts were estimated for workers and the general population. For heavy-haul truck shipments, workers included vehicle drivers and escorts. For barge shipments, the work crew included five members on board during travel and workers close to the shipping casks during inspections or intermodal transfers. The general population for truck shipments included persons within 800 meters (about 2,600 feet) of the road (offlink), persons sharing the road (onlink), and persons at stops. The general population for barging included persons within a range of 200 to 1,000 meters (about 660 to 3,300 feet) of the route, and persons at stops. On-link exposures to members of the public during barging were assumed to be small. Incident-free unit risk factors were developed to calculate occupational and general population collective doses. Table J-28 lists the unit risk factors for heavy-haul truck and barge shipments. The unit risk factors for heavy-haul truck shipments reflect the effects of slower operating speeds for those vehicles in comparison to those for legal-weight trucks.

Table J-28. Risk factors for incident-free heavy-haul truck and barge transportation of spent nuclear fuel and high-level radioactive waste.

		Incident free risk factors (person-rem per kilometer)[a]		
Mode	Exposure group	Rural	Suburban	Urban
Heavy-haul truck	*Occupational*	1.1×10^{-5}	1.1×10^{-5}	1.9×10^{-5}
	General population			
	Offlink[b]	7.3×10^{-8}	7.7×10^{-8}	8.3×10^{-8}
	Onlink[c]	1.1×10^{-4}	1.2×10^{-4}	5.5×10^{-4}
	Stops	1.9×10^{-4}	1.9×10^{-4}	1.9×10^{-4}
	Storage[d]	1.9×10^{-3}	1.9×10^{-3}	1.9×10^{-3}
	Totals	$\mathbf{2.2\times10^{-3}}$	$\mathbf{2.3\times10^{-3}}$	$\mathbf{2.7\times10^{-3}}$
Barge	*Occupational*[d]	9.4×10^{-7}	1.9×10^{-6}	4.8×10^{-6}
	General population			
	Offlink[b]	8.6×10^{-8}	1.7×10^{-7}	4.3×10^{-7}
	Onlink[c]	0.0	0.0	0.0
	Stops	5.4×10^{-3}	5.4×10^{-3}	5.4×10^{-3}
	Totals	$\mathbf{5.4\times10^{-3}}$	$\mathbf{5.4\times10^{-3}}$	$\mathbf{5.5\times10^{-3}}$

a. The methodology, equations, and data used to develop the unit dose factors are discussed in Madsen et al. (1986, all) and Neuhauser and Kanipe (1992, all). Cashwell et al. (1986, all) contains a detailed explanation of the use of unit factors.
b. Offlink general population included persons within 800 meters (about 2,600 feet) of the road or railway.
c. Onlink general population included persons sharing the road or railway.
d. The storage unit risk factor is only applied for heavy-haul truck shipments requiring an overnight stop.

Table J-29 lists the incident-free impacts on a per shipment basis from the Turkey Point nuclear power plant using the three shipment scenarios listed above. This is presented to compare the impacts on a per shipment basis using barge, heavy-haul truck or legal weight truck. Impacts of intermodal transfers are included in the results. Occupational impacts would include the estimated radiological exposures of security escorts.

Table J-29. Comparison of population doses and impacts from incident-free national transportation for heavy-haul-to-rail, barge-to-rail, and legal-weight truck options.[a,b]

Category	Heavy-haul to rail	Barge to rail	Legal-weight truck
Involved worker			
Collective dose (person-rem)	0.15	0.13	0.32
Estimated LCFs[e]	0.00006	0.00005	0.00013
Public			
Collective dose (person-rem)	0.12	0.41	1
Estimated LCFs	0.00006	0.0002	0.0005
Maximally exposed individual	Impacts would be the same as those in Chapter 6, Tables 6-9 and 6-12		

a. Rail impacts are presented for the Caliente-Chalk Mountain rail implementing alternative.
b. Impacts presented on a per shipment basis for the Turkey Point site.
c. LCF = latent cancer fatality.

As indicated in Table J-29, differences in radiological impacts between the use of heavy-haul trucks and barges would be small. The impacts to maximally exposed individuals would be the same because both cases use the same assumptions for locations of such individuals in relation to shipments and times of exposure.

J.2.1.2.2 *Nonradiological Impacts of Incident-Free Transportation (Vehicle Emissions)*

Table J-30 compares the estimated number of fatalities from vehicle emissions from shipments, assuming the use of heavy-haul trucks or barges to ship to nearby railheads.

Table J-30. Population health impacts from vehicle emissions during incident-free national transportation for mostly legal-weight truck scenario.[a]

Category	Heavy-haul to rail	Barge to rail	Legal-weight truck
Estimated fatalities	0.00004	0.00004	0.00003

a. Impacts are presented on a per shipment basis for the Turkey Point site.

J.2.1.3 Analysis of Impacts of Accidents for Barge and Heavy-Haul Truck Transportation

J.2.1.3.1 *Radiological Impacts of Accidents*

The analysis of risks from accidents during heavy-haul truck, rail, and legal-weight truck transport of spent fuel and high-level radioactive waste used the RADTRAN4 computer code (Neuhauser and Kanipe 1992, all) and the analysis approach discussed in Section J.1.4.2. The analysis of risks due to barging used the same methodology with the exception of conditional probabilities. For barge shipments, the conditional accident probabilities (Table J-31) for each cask response category were based on a review of other barge accident analyses.

Table J-31. Conditional probabilities for barge transportation.

Severity category	1	2	3	4	5	6
Conditional probability	0.93794	0.005	0.000	0.057	0.000051	0.0000058

When radioactive material is shipped by barge, it is possible to have both water and land contamination. The analysis assumed that airborne releases could occur in accidents involving barges. Any portion of a release plume over water would result in water contamination. Thus, there are two mechanisms for contaminating water and one, the airborne release, for contaminating land surfaces.

For accident scenarios that result in releases of radioactive material, part of the plume would be deposited on water and part on land. For coastal and lake shipping, the analysis assumed that, 50 percent of the time, the plume would be entirely deposited on water. For the other 50 percent, the analysis assumed that the accident would occur about 200 meters (660 feet) from the shore and any material deposited in the first 200 meters would be into water. The analysis used the methods used by the RISKIND computer program (Yuan et al. 1995 all) to estimate plume depletion into water for D stability and a wind speed of 3 meters per second. For these conditions, about 20 percent of the plume would be depleted in the first 200 meters. Based on this information, the analysis assumed that for coastal and lake shipping, 60 percent of the plume would be deposited on water and for river transport only 20 percent of the release would occur over water.

The analysis accommodated this split by allocating 60 percent of coastal and lake shipping to what was called a "water" state and the remaining 40 percent to an adjoining state (Florida in the case of Turkey

Point). For river transport, 20 percent of the mileage was allocated to the water state representing the river and the remaining 80 percent of the mileage was allocated to the adjacent state (Mississippi in the case of Browns Ferry).

The dose from plume release to water was limited to an ingestion dose. The transfer coefficients that were used in the calculation are listed in Table J-32. The selection of isotopes and the transfer coefficients was based on models used in the Foreign Spent Nuclear Fuel EIS (DOE 1996a, page E-126). The same water uptake models were used. Both the freshwater and ocean models considered fish consumption. The freshwater model included irrigation and domestic water consumption by both the general population and livestock. The ocean model included uptake from eating shellfish.

Table J-32. Food transfer factors used in the barge analysis.

Isotope	Ocean release	Freshwater release
Hydrogen-3 (tritium)		0.000020
Niobium-95	0.080	
Ruthenium-106	0.00014	
Cesium-134	0.00037	0.000022
Cesium-137	0.00037	0.000022

In addition, the analysis of barge accident risks used the following assumptions:

- Release fractions that determine the source term for dispersion to the waterway are the same as those developed for airborne release scenarios

For freshwater river systems, the analysis assessed the following exposure pathways:

- Drinking water
- Ingestion of fish by humans
- Ingestion of irradiated foods
- Shoreline deposits
- External irradiation from immersion during swimming

For marine coastal systems, the following exposure pathways were assessed:

- Ingestion of fish and invertebrates by humans
- External irradiation from shoreline deposits
- External irradiation from immersion during swimming

Route-specific collective doses were calculated using population distributions along the routes developed from 1990 Census data. As an example, Table J-33 presents the dose risk per shipment for the Turkey Point nuclear power plant.

Table J-33. Accident risks for shipping spent nuclear fuel from Turkey Point.

Category	Heavy-haul to rail	Barge to rail	Legal-weight truck
Dose risk (person-rem)	0.0038	0.0019	0.0023
Dose risk (LCF)[a]	0.000002	0.0000009	0.000001
Traffic fatalities	0.00039	0.00039	0.00011

a. LCF = latent cancer fatality.

J.2.1.3.2 *Nonradiological Accident Risks*

The fatalities per shipment for heavy-haul truck, barge, and legal-weight truck transport from Turkey Point would be 3.9×10^{-4}, 3.9×10^{-4} and 1.1×10^{-4} , respectively.

J.2.1.3.3 *Maximum Reasonably Foreseeable Accidents*

With the relatively short barging distance relative to the rail distance traveled, the probability of a barge accident is much lower than the 1×10^{-7}-criteria used for accidents that are reasonably foreseeable.

J.2.2 EFFECTS OF USING DEDICATED TRAINS OR GENERAL FREIGHT SERVICE

The Association of American Railroads recommends that only special (dedicated) trains move spent nuclear fuel and certain other forms of radioactive materials (DOT 1998b, page 2-6). In developing its recommendation, the Association concluded that the use of special trains would provide operational (for railroads and shippers) and safety advantages over shipments that used general freight service. Notwithstanding this recommendation, the Department of Transportation study (DOT 1998b, all) compared dedicated and regular freight service using factors that measure impacts to overall public safety. The results of this study indicated that dedicated trains could provide advantages over regular trains for incident-free transportation but could be less advantageous for accident risks. However, available information does not indicate a clear advantage for the use of either dedicated trains or general freight service. Thus, DOE has not determined the commercial arrangements it would request from railroads for shipment of spent nuclear fuel and high-level radioactive waste. Table J-34 compares the dedicated and general freight modes. These comparisons are based on the findings of the Department of Transportation study and the Association of American Railroads.

J.3 Nevada Transportation

With the exceptions of the possible construction of a branch rail line or upgrade of highways for use by heavy-haul trucks and the construction of an intermodal transfer station, the characteristics of the transportation of spent nuclear fuel and high-level radioactive waste in Nevada would be similar to those for transportation in other states across the nation. Unless the State of Nevada designated alternative or additional preferred routes as prescribed under regulations of the Department of Transportation (49 CFR 397.103), Interstate System Highways (I-15) would be the preferred routes used by legal-weight trucks carrying spent nuclear fuel and high-level radioactive waste. Unless alternative or non-Interstate System routes have been designated by states, Interstate system Highways would also be the preferred routes used by legal-weight trucks in other states during transit to Nevada.

In Nevada as in other states, rail shipments would, for the most part, be transported on mainline tracks of major railroads. Operations over a branch rail line in Nevada would be similar to those on a mainline railroad, except the frequency of train travel would be much lower. Shipments in Nevada that used heavy-haul trucks would use Nevada highways in much the same way that other overdimensional, overweight trucks use the highways along with other commercial vehicle traffic.

In some cases State-specific assumptions were used to analyze human health and safety impacts in Nevada. A major difference would be that much of the travel in the State would be in rural areas where population densities are much lower than those of many other states. Another difference would be for travel in an urban area in the state. The most populous urban area in Nevada is the Las Vegas metropolitan area, which is also a major resort area with a high percentage of nonresidents. The analysis also addressed the channeling of shipments from the commercial and DOE sites into the transportation arteries in the southern part of the State. Finally, the analysis addressed the commuter and commercial

Table J-34. Comparison of general freight and dedicated train service.

Attribute	General freight	Dedicated train
Overall accident rate for accidents that could damage shipping casks	Same as mainline railroad accident rates	Expected to be lower than general freight service because of operating restrictions and use of the most up-to-date railroad technology.
Grade crossing, trespasser, worker fatalities	Same as mainline railroad rates for fatalities	Uncertain. Greater number of trains could result in more fatalities in grade crossing accidents. Fewer stops in classification yards could reduce work related fatalities and trespasser fatalities.
Security	Security provided by escorts required by NRC[a] regulations	Security provided by escorts required by NRC regulations; fewer stops in classification yards than general freight service.
Incident-free dose to public	Low, but more stops in classification yards than dedicated trains. However, classification yards would tend to be remote from populated areas.	Lower than general freight service. Dedicated trains could be direct routed with fewer stops in classification yards for crew and equipment changes.
Radiological risks from accidents	Low, but greater than dedicated trains	Lower than general freight service because operating restrictions and equipment could contribute to lower accident rates and reduced likelihood of maximum severity accidents.
Occupational dose	Duration of travel influences dose to escorts	Shorter travel time would result in lower occupational dose to escorts.
Utilization of resources	Long cross-country transit times could result in least efficient use of expensive transportation cask resources; best use of railroad resources; least reliable delivery scheduling; most difficult to coordinate state notifications.	Direct through travel with on-time deliveries would result in most efficient use of cask resources; least efficient use of railroad resources. Railroad resource demands from other shippers could lead to schedule and throughput conflicts. Easiest to coordinate notification of state officials.

a. NRC = U.S. Nuclear Regulatory Commission.

travel that would occur on highways in the southern part of the State as a consequence of the construction, operation and monitoring, and closure of the proposed repository.

This section presents information specific to Nevada that DOE used to estimate impacts for transportation activities that would take place in the State. It includes results for cumulative impacts that would occur in Nevada for transportation associated with Inventory Modules 1 and 2.

J.3.1 TRANSPORTATION MODES, ROUTES, AND NUMBER OF SHIPMENTS

J.3.1.1 Routes in Nevada for Legal-Weight Trucks

The analysis of impacts that would occur in Nevada used the characteristics of (1) highways in Nevada that would be used for shipments of spent nuclear fuel and high-level radioactive waste by legal-weight trucks, (2) rail routes from the border to rail nodes where the implementing alternatives would connect, and (3) rail corridors and highway routes analyzed for the rail and heavy-haul truck implementing alternatives in the State.

Figure J-10 shows the routes in Nevada that legal-weight trucks would use unless the State designated alternative or additional preferred routes. The figure shows estimates for the number of legal-weight truck shipments that would travel on each route segment for the mostly legal-weight truck and mostly rail transportation scenarios. The inset on Figure J-10 shows the proposed Las Vegas Beltway and the routes DOE anticipates legal-weight trucks traveling to the repository would use.

J.3.1.2 Routes in Nevada for Transporting Rail Casks

The rail and heavy-haul truck implementing alternatives for transportation in Nevada include five possible rail corridors and five possible routes for heavy-haul trucks; the corridors and routes for these implementing alternatives are shown in Figures J-11 and J-12. These figures also show the estimated number of rail shipments that would enter the State on mainline railroads. These numbers indicate shipments that would arrive from the direction of the bordering state for each of the implementing alternatives for the mostly rail transportation scenario.

Table J-35 lists the total length and cumulative distance in rural, suburban, and urban population zones in the State of Nevada used to analyze impacts of the implementing alternatives. Table J-36 lists the total population that lives within 800 meters (0.5 mile) of rail lines in Nevada. The estimated population that would live along each branch rail line was based on population densities along existing mainline railroads in Nevada.

Nevada Heavy-Haul Truck Scenario

Tables J-37 through J-41 summarize the road upgrades for each of the five possible routes for heavy-haul trucks that DOE estimates would be needed before routine use of a route to ship casks containing spent nuclear fuel and high-level radioactive waste.

Nevada Rail Corridors

Under the mostly rail scenario, DOE could construct and operate a branch rail line in Nevada. Based on the studies listed below, DOE has narrowed its consideration for a new branch rail line to five potential rail corridors—the Carlin, Caliente, Caliente-Chalk Mountain, Jean, and Valley Modified routes. DOE identified the five rail corridors through a process of screening potential rail alignments that it had studied in past years. Several studies evaluated rail options.

- The *Feasibility Study for Transportation Facilities to Nevada Test Site* study (Holmes & Narver 1962, all) determined the technical and economic feasibility of constructing and operating a railroad from Las Vegas to Mercury.

- The *Preliminary Rail Access Study* (Tappen and Andrews 1990, all) identified 13 and evaluated 10 rail corridor alignment options. This study recommended the Carlin, Caliente, and Jean corridors for detailed evaluation.

- *The Nevada Railroad System: Physical, Operational, and Accident Characteristics* (DOE 1991, all) described the operational and physical characteristics of the current Nevada railroad system.

- The *High Speed Surface Transportation Between Las Vegas and the Nevada Test Site (NTS)* report (Raytheon 1994, all) explored the rationale for a potential high-speed rail corridor between Las Vegas and the Nevada Test Site to accommodate personnel.

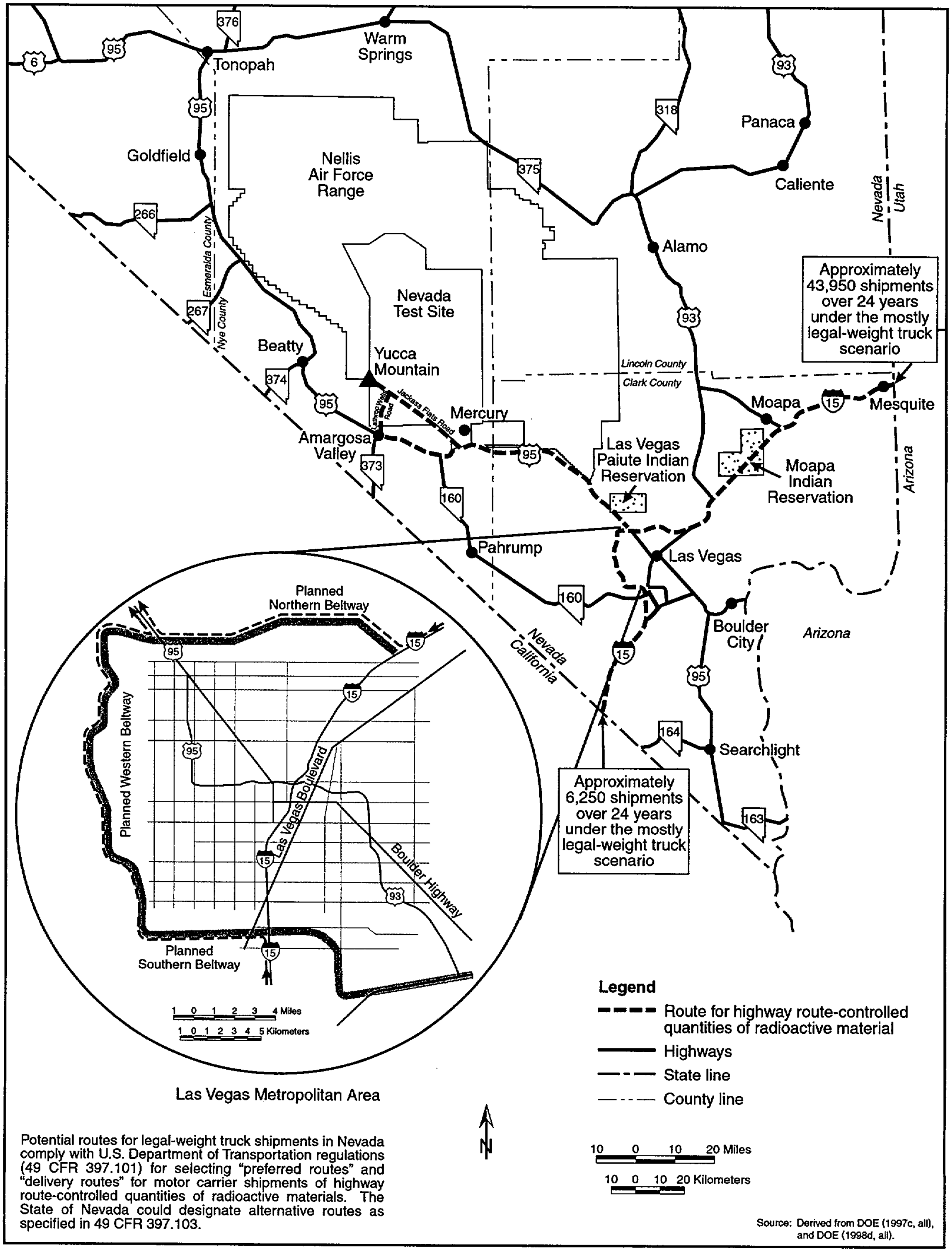


Figure J-10. Potential Nevada routes for legal-weight truck shipments of spent nuclear fuel and high-level radioactive waste to Yucca Mountain.

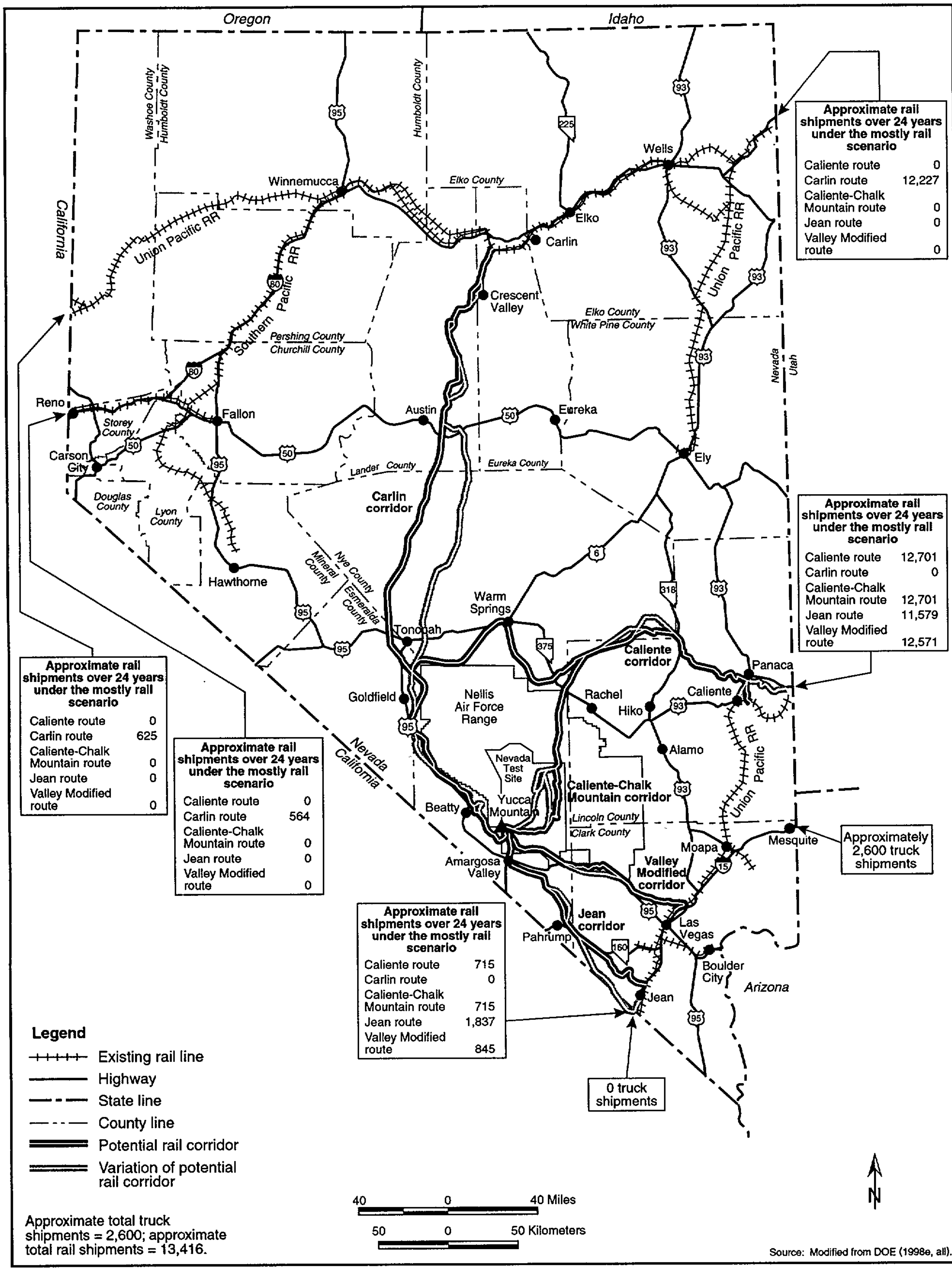


Figure J-11. Potential Nevada rail routes to Yucca Mountain and approximate number of shipments for each route.

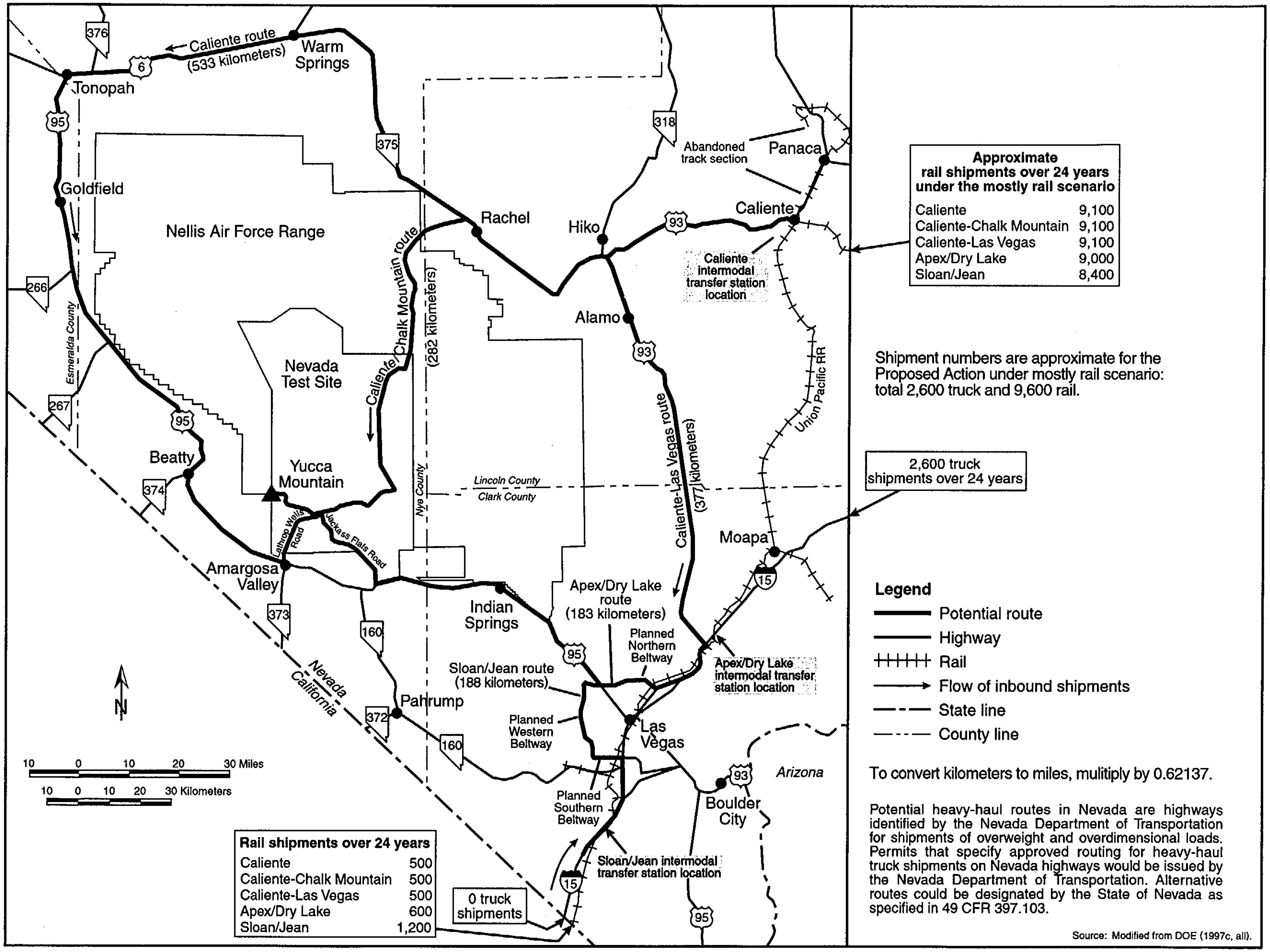


Figure J-12. Nevada routes for heavy-haul truck shipments of spent nuclear fuel and high-level radioactive waste to Yucca Mountain.

Table J-35. Route characteristics for rail and heavy-haul truck implementing alternatives.

Alternative	Rail node	Distance (kilometers)[a]			
		Rural	Suburban	Urban	Total[b]
Rail					
Caliente	Caliente	513	0	0	513
Carlin	Beowawe	520	0	0	520
Caliente-Chalk Mountain	Caliente	345	0	0	345
Jean	Jean	181	0	0	181
Valley Modified	Apex	159	0	0	159
Heavy-haul[c]					
Caliente	Caliente	533	0	0	533
Caliente-Chalk Mountain	Caliente	282	0	0	282
Caliente-Las Vegas	Caliente	356	21	0	377
Apex/Dry Lake	Apex	162	21	0	183
Sloan/Jean	Jean	145	43	0	188

a. To convert kilometers to miles, multiply by 0.62137.
b. Rounded to the nearest kilometer.
c. Heavy-haul distances are based on using the Northern, Western, and Southern Beltways in the Las Vegas area. These beltways are assumed to have suburban population density.

Table J-36. Populations in Nevada within 800 meters (0.5 mile) of routes.

Transportation scenario	Population 1990 Census
Legal-weight truck routes[a]	60,000
Rail routes Nevada border to branch rail line[b]	
Caliente	30,000
Carlin	52,000
Caliente-Chalk Mountain	30,000
Jean	30,000
Valley Modified	30,000
Branch rail lines[c]	
Caliente	2,600
Carlin	2,700
Caliente-Chalk Mountain	1,800
Jean	900
Valley Modified	800

a. Source: TRW (1999a, Table 5-1).
b. Source: TRW (1999a, Table 5-2).
c. Estimated using 3.2 persons per square kilometer – the highest value for rural populations along mainline railroads in Nevada (TRW 1999a, Table 5-2).

- The *Nevada Potential Repository Preliminary Transportation Strategy, Study 1* (TRW 1995, all), reevaluated 13 previously identified rail routes and evaluated a new route called the Valley Modified route. This study recommended four rail routes for detailed evaluation—the Caliente, Carlin, Jean, and Valley Modified routes.

- The *Nevada Potential Repository Preliminary Transportation Strategy, Study 2* (TRW 1996, all), further refined the analyses of potential rail corridor alignments presented in Study 1.

Public comments submitted to DOE during hearings on the scope of this environmental impact statement resulted in addition of a fifth potential rail corridor—Caliente-Chalk Mountain.

Table J-37. Potential road upgrades for Caliente route.[a]

Route	Upgrades
Intermodal transfer station to U.S. 93	Pave existing gravel road.
U.S. 93 to State Route 375	Asphalt overlay on existing pavement, truck lanes where grade is greater than 4 percent (minimum distance of 460 meters[b] per lane), turnout lanes every 32 kilometers[c] (distance of 305 meters per lane), widen road.
State Route 375 to U.S. 6	Remove existing pavement, increase road base and overlay to remove frost restrictions, truck lanes where grade is greater than 4 degrees (minimum distance of 460 meters per lane), turnout lanes every 32 kilometers (distance of 305 meters per lane), widen road.
U.S. 6 to U.S. 95	Same as State Route 375 to U.S. 6.
U.S. 95 to Lathrop Wells Road	Remove existing pavement on frost restricted portion, increase base and overlay to remove frost restrictions, turnout lanes every 8 kilometers (distance of 305 meters per lane), construct bypass around intersection at Beatty, bridge upgrade near Beatty.
Lathrop Wells Road to Yucca Mountain site	Asphalt overlay on existing roads.

a. Source: TRW (1999b, Heavy-Haul Truck Files, Item 4).
b. To convert meters to feet, multiply by 3.2808.
c. To convert kilometers to miles, multiply by 0.62137.

Table J-38. Potential road upgrades for Caliente-Chalk Mountain route.[a]

Route	Upgrades
Intermodal transfer station to U.S. 93	Pave existing gravel road.
U.S. 93 to State Route 375	Asphalt overlay on existing pavement, truck lanes where grade is greater than 4 percent (minimum distance 460 meters[b] per lane), turnout lanes every 32 kilometers[c] (distance of 305 meters per lane), widen road.
State Route 375 to Rachel	Remove existing pavement, increase road base and overlay to remove frost restrictions, turnout lanes every 32 kilometers (distance of 305 meters per lane), widen road.
Rachel to Nellis Air Force Range	Pave existing gravel road.
Nellis Airforce Range Roads	Rebuild existing road.
Nevada Test Site Roads	Asphalt overlay on existing roads.

a. Source: TRW (1999b, Heavy-Haul Truck Files, Item 9).
b. To convert meters to feet, multiply by 3.2808.
c. To convert kilometers to miles, multiply by 0.62137.

DOE has identified 0.4-kilometer (0.25-mile)-wide corridors along each route within which it would need to obtain a right-of-way to construct a rail line and an associated access road. A corridor defines the boundaries of the route by identifying an established "zone" for the location of the railroad. For this analysis, DOE identified a single alignment for each of the corridors. These single alignments are representative of the range of alignments that DOE has considered for the corridors from engineering design and construction viewpoints. The following paragraphs describe the alignments that have been identified for the corridors. Before siting a branch rail line, DOE would conduct engineering studies in each corridor to determine a specific alignment for the roadbed, track, and right-of-way for a branch rail line.

Carlin Rail Corridor Implementing Alternative. The Carlin corridor originates at the Union Pacific main line railroad near Beowawe in north-central Nevada. The corridor is about 520 kilometers (331

Table J-39. Potential road upgrades for Caliente-Las Vegas route.[a]

Route	Upgrades
Intermodal transfer station to U.S. 93	Pave existing gravel road.
U.S. 93 to Interstate 15	Asphalt overlay on existing pavement, truck lanes where grade is greater than 4 percent (minimum distance 460 meters[b] per lane), turnout lanes every 32 kilometers[c] (distance of 305 meters per lane), widen road, rebuild Interstate 15 interchange.
Interstate 15 to U.S. 95	Increase existing two-lane Las Vegas Beltway to four lanes, asphalt overlay on U.S. 95.
U.S. 95 to Mercury	Asphalt overlay on U.S. 95.
Mercury Exit to Yucca Mountain site	Asphalt overlay on Jackass Flats Road, rebuild road when required.

a. Source: TRW (1999b, Heavy-Haul Truck Files, Item 4).
b. To convert meters to feet, multiply by 3.2808.
c. To convert kilometers to miles, multiply by 0.62137.

Table J-40. Potential road upgrades for Apex/Dry Lake route.[a]

Route	Upgrades
Intermodal transfer station to Interstate 15	Rebuild frontage road to U.S. 93. Rebuild U.S. 93/Interstate 15 interchange.
Interstate 15 to U.S. 95	Increase existing two-lane Las Vegas Beltway to four lanes.
U.S. 95 to Mercury Exit	Asphalt overlay on U.S. 95.
Mercury Exit to Yucca Mountain site	Asphalt overlay on Jackass Flats Road, rebuild road when required.

a. Source: TRW (1999b, Heavy-Haul Truck Files, Item 4).

Table J-41. Potential road upgrades for Sloan/Jean route.[a]

Route	Upgrades
Intermodal transfer station to Interstate 15	Overlay and widen existing road to Interstate 15 interchange, rebuild Interstate 15 interchange.
Interstate 15 to U.S. 95	Increase existing two-lane Las Vegas Beltway to four lanes.
U.S. 95 to Mercury Exit	Asphalt overlay on U.S. 95.
Mercury Exit to Yucca Mountain site	Asphalt overlay on Jackass Flats Road, rebuild road when required.

a. Source: TRW (1999b, Heavy-Haul Truck Files, Item 4).

miles) long from the tie-in point with the Union Pacific line to the Yucca Mountain site. Table J-42 lists possible variations in the alignment of this corridor.

Caliente Rail Corridor Implementing Alternative. The Caliente corridor originates at an existing siding to the Union Pacific mainline railroad near Caliente, Nevada. The Caliente and Carlin corridors converge near the northwest boundary of the Nellis Air Force Range. Past this point, they are identical. The Caliente corridor would be 513 kilometers (320 miles) long from the Union Pacific line connection to the Yucca Mountain site. Table J-43 lists possible alignment variations for this corridor.

Caliente-Chalk Mountain Rail Corridor Implementing Alternative. The Caliente-Chalk Mountain corridor is identical to the Caliente corridor until it approaches the northern boundary of the Nellis Air Force Range. At this point the Caliente-Chalk Mountain corridor turns south through the Nellis Air Force Range and the Nevada Test Site to the Yucca Mountain site. The corridor would be 345 kilometers (214 miles) long from the tie-in point at the Union Pacific line to the Yucca Mountain Site. Table J-44 lists possible alignment variations for this corridor.

Table J-42. Possible alignment variations of the Carlin corridor.[a]

Corridor	Description
Crescent Valley	Would diverge from the analyzed alignment near Cortez Mining Operation; would travel through nonagricultural lands adjacent to alkali flats but would affect larger area of private land.
Wood Spring	Would diverge from the analyzed alignment and use continuous 2-percent grade to descend from Dry Canyon Summit in Toiyabe range; would be shorter than the analyzed alignment but would have steeper grade.
Rye Patch	Would travel through Rye Patch Canyon, which has springs, riparian areas, and game habitats; would divert from the analyzed alignment, maintaining distance of 420 meters[b] from Rye Patch Spring and at least 360 meters from riparian areas throughout Rye Patch Canyon, except at crossing of riparian area near south end of canyon; would avoid game habitat (sage grouse strutting area).
Steiner Creek	Would diverge from the analyzed alignment at north end of Rye Patch Canyon. Would avoid crossing private lands, two known hawk-nesting areas, and important game habitat (sage grouse strutting area) in the analyzed alignment.
Monitor Valley	Would travel through less populated Monitor Valley (in comparison to Big Smokey Valley).
Mud Lake[c]	Would travel farther from west edge of Mud Lake, which has known important archaeological sites.
Goldfield[c]	Would avoid crossing Nellis Air Force Range boundary near Goldfield, avoiding potential land-use conflicts with Air Force.
Bonnie Claire[c]	Would avoid crossing Nellis Air Force Range boundary near Scotty's Junction, avoiding potential land-use conflicts with Air Force.
Oasis Valley[c]	Would enable flexibility in crossing environmentally sensitive Oasis Valley area. If DOE selected route through this area, further studies would ensure small environmental impacts.
Beatty Wash[c]	Would provide a corridor through Beatty Wash that was longer, but required less severe earthwork than the analyzed alignment.

a. Source: TRW (1999b, Rail Files, Item 6).
b. To convert meters to feet, multiply by 3.2808.
c. Common with Caliente corridor.

Table J-43. Possible alignment variations of the Caliente corridor.[a]

Corridor	Description
Caliente[b]	Would connect with Union Pacific line at existing siding in Town of Caliente.
Crestline[b]	Would connect with Union Pacific line near east end of existing siding at Crestline.
White River	Would avoid potential conflict with Weepah Spring Wilderness Study Area.
Garden Valley	Would put more distance between rail corridor and private lands in Garden Valley and Coal Valley.
Mud Lake[c]	Would travel farther from west edge of Mud Lake, which has known important archaeological sites.
Goldfield[c]	Would avoid crossing Nellis Air Force Range boundary near Goldfield, avoiding potential land-use conflicts with Air Force.
Bonnie Claire[c]	Would avoid crossing Nellis Air Force Range boundary near Scotty's Junction, avoiding potential land-use conflicts with Air Force.
Oasis Valley[c]	Would enable flexibility in crossing environmentally sensitive Oasis Valley area. If DOE selected route through this area, further studies would ensure small environmental impacts.
Beatty Wash[c]	Would provide corridor through Beatty Wash that was longer, but required less severe earthwork than the analyzed alignment.

a. Source: TRW (1999b, Rail Files, Item 6).
b. Common with Caliente-Chalk Mountain corridor.
c. Common with Carlin corridor.

Table J-44. Possible alignment variations of the Caliente-Chalk Mountain corridor.[a]

Corridor	Description
Mercury Highway	To provide flexibility in choosing path, would travel north through center of Nevada Test Site.
Tonopah	To provide flexibility in choosing path through Nevada Test Site; would travel north along western boundary of Nevada Test Site.
Mine Mountain	Would provide flexibility in minimizing impacts to local archaeological sites.
Area 4	Would provide flexibility in choosing path through Nevada Test Site.

a. Source: TRW (1999b, Rail Files, Item 8).

Jean Rail Corridor Implementing Alternative. The Jean corridor originates at the existing Union Pacific mainline railroad near Jean, Nevada. The corridor would be 181 kilometers (112 miles) long from the tie-in point at the Union Pacific line to the Yucca Mountain site. Table J-45 lists possible variations for this corridor.

Table J-45. Possible alignment variations of the Jean corridor.[a]

Corridor	Description
North Pahrump	Would minimize impacts to approximately 4 kilometers[b] of private land on northeast side of Pahrump.
Stateline Pass	Would provide option to crossing Spring Mountains at Wilson Pass; would diverge from analyzed alignment in Pahrump Valley; would parallel Nevada-California border, traveling along southwestern edge of Spring Mountains and crossing border twice.

a. Source: TRW (1999b, Rail Files, Item 6).
b. 4 kilometers = 2.5 miles (approximate).

Valley Modified Rail Corridor Implementing Alternative. The Valley Modified corridor originates at an existing rail siding off the Union Pacific mainline railroad northeast of Las Vegas. The corridor is about 159 kilometers (98 miles) long from the tie-in point with the Union Pacific line to the Yucca Mountain site. Table J-46 lists the possible variations in alignment for this corridor.

Table J-46. Possible alignment variations of the Valley Modified corridor.[a]

Corridor	Description
Indian Hills	Would avoid entrance to Nellis Air Force Range north of Town of Indian Springs by traveling south of town.
Sheep Mountain	Would increase distance from private land in Las Vegas and proposed 30-square-kilometer[b] Bureau of Land Management land exchange with city.
Valley Connection	Would locate transfer operations at Union Pacific Valley Yard rather than Dike siding. Overflights of Dike siding from Nellis Air Force Base could conflict with switching operations.

a. Source: TRW (1999b, Rail Files, Item 6).
b. 30 square kilometers = 7,410 acres (approximate).

J.3.1.3 Sensitivity of Analysis Results to Routing Assumptions

In addition to analyzing the impacts of using highway routes that would meet Department of Transportation requirements for transporting spent nuclear fuel, DOE evaluated how the estimated impacts would differ if legal-weight trucks used other routes in Nevada. Six other routes identified in a 1989 study by the Nevada Department of Transportation (Ardila-Coulson 1989, pages 36 and 45) were selected for this analysis. The Nevada Department of Transportation study described the routes as follows:

Route A. Minimum distance and minimum accident rate.
South on U.S. 93A, south on U.S. 93, west on U.S. 6, south on Nevada 318, south on U.S. 93, south on I-15, west on Craig Road, north on U.S. 95

Route B. Minimum population density and minimum truck accident rate.
South on U.S. 93A, south on U.S. 93, west on U.S. 6, south on U.S. 95.

Both of these two routes use the U.S. 6 truck bypass in Ely.

Alternative route possibilities were identified between I-15 at Baker, California and I-40 at Needles, California to Mercury. These alternative routes depend upon the use of U.S. 95 in California, California 127 and the Nipton Road.

Route C. From Baker with California 127.
North on California 127, north on Nevada 373, south on U.S. 95

Route D. From Baker without California 127.
North on I-15, west on Nevada 160, south on U.S. 95

Route E. From Needles with U.S. 95, California 127, and the Nipton Road.
North on U.S. 95, west on Nevada 164, west on I-15, north on California 127, north on Nevada 373, south on U.S. 95

Route F. From Needles without California 127 and the Nipton Road.
West on I-40, east on I-15, west on Nevada 160, south on U.S. 95

Table J-47 identifies the sensitivity cases evaluated based on the Nevada Department of Transportation routes. Table J-48 lists the range of impacts in Nevada of using these different routes for the mostly legal-weight truck analysis scenario. The tables compare the impacts estimated for the highways identified in the Nevada study to those estimated for shipments that would follow routes allowed by current Department of Transportation regulations for Highway Route-Controlled Quantities of Radioactive Materials. Because the State of Nevada has not designated alternative or additional preferred routes for use by these shipments, as permitted under Department of Transportation regulations (49 CFR 397.103), DOE has assumed that shipments of spent nuclear fuel and high-level radioactive waste would

Table J-47. Nevada routing sensitivity cases analyzed for a legal-weight truck.

Case	Description
Case 1	To Yucca Mountain via Barstow, California, using I-15 to Nevada 160 to Nevada 160 (Nevada D and F)
Case 2	To Yucca Mountain via Barstow using I-15 to California route 127 to Nevada 373 to US 95 (Nevada C)
Case 3	To Yucca Mountain via Needles using U.S. 95 to Nevada 164 to I-15 to California 127 to Nevada 373 and U.S. 95 (Nevada E)
Case 4	To Yucca Mountain via Needles using U.S. 95 to Nevada 164 to I-15 to Nevada 160 (variation of Nevada E)
Case 5	To Yucca Mountain via Wendover using U.S. 93 Alternate to U.S. 93 to US 6 to U.S. 95 (Nevada B)
Case 6	To Yucca Mountain via Wendover using U.S. 93 Alternate to U.S. 93 to Nevada 318 to U.S. 93 to I-15 to the Las Vegas Beltway to U.S. 95 (Nevada A)

Table J-48. Comparison of impacts from the sensitivity analyses (national and Nevada).

	Base case		Barstow via Nevada 160		Barstow via U.S. 95		Needles via Nevada 160		Needles via U.S. 95		Wendover via U.S. 95		Wendover via Las Vegas Beltway	
	National	Nevada	National	Nevada	National	Nevada	National	Nevada	National	Nevada	National	Nevada	National	Nevada
Public incident-free dose (person-rem)	35,000	2,700	39,000	2,500	38,000	710	39,000	2,900	37,000	1,100	38,000	7,100	38,000	7,600
Occupational incident-free dose (person-rem)	11,000	1,600	12,000	1,500	12,000	1,100	12,000	1,600	12,000	1,200	12,000	2,600	12,000	2,700
Pollution health effects nonradioactive	0.60	0.006	0.68	0.005	0.68	0.004	0.64	0.003	0.64	0.001	0.61	0.011	0.61	0.011
Public incident-free risk of latent cancer fatality	17	1.4	19	1.2	19	0.4	18	1.4	19	0.6	19	3.5	19	3.8
Occupational incident-free risk of latent cancer fatality	4.5	0.6	4.9	0.6	4.8	0.4	4.7	0.6	4.7	0.5	4.7	1.0	4.8	1.1
Radiological accident risk (person-rem)	130	0.5	100	0.4	100	0.0	98	0.4	98	0.1	140	1.0	140	1.0
Radiological accident risk of latent cancer fatality	0.067	0.00024	0.0	0.00020	0.050	0.00001	0.049	0.00021	0.049	0.00003	0.069	0.0005	0.069	0.0005
Traffic fatalities	3.9	0.5	4.3	0.4	4.0	0.1	4.2	0.5	4.0	0.2	4.7	1.2	4.8	1.3

enter Nevada on I-15 from either the northeast or southwest. The analysis assumed that shipments traveling on I-15 from the northeast would use the northern Las Vegas Beltway to connect to U.S. 95 and continue to the Nevada Test Site. Shipments from the southwest on I-15 would use the southern and western Las Vegas Beltway to connect to U.S. 95 and continue to the Nevada Test Site.

J.3.2 ANALYSIS OF INCIDENT-FREE TRANSPORTATION IN NEVADA

The analysis of incident-free impacts to populations in Nevada addressed transportation through urban, suburban, and rural population zones. The population densities that were assumed for the analysis were determined using the HIGHWAY and INTERLINE computer programs. The population in the 800-meter (0.5-mile) region of influence used to evaluate the impacts of incident-free transportation for both legal-weight truck and rail shipments is listed in Table J-36.

Results for incident-free transportation of spent nuclear fuel and high-level radioactive waste for Inventory Modules 1 and 2 are presented in Section J.3.4.

J.3.3 ANALYSIS OF TRANSPORTATION ACCIDENT SCENARIOS IN NEVADA

Section J.1.4 discusses the methodology for estimating the risks of accidents that could occur during rail and truck transportation of spent nuclear fuel and high-level radioactive waste. Section J.3.5 describes the results of the accident risk analysis for Inventory Modules 1 and 2.

J.3.3.1 Intermodal Transfer Station Accident Methodology

Shipping casks would arrive at an intermodal transfer station in Nevada by rail, and a gantry crane would transfer them from the railcars to heavy-haul trucks for transportation to the repository. The casks, which would not be opened or altered in any way at the intermodal transfer station, would be certified by the Nuclear Regulatory Commission and would be designed for accident conditions specified in 10 CFR Part 71. Impact limiters, which would protect casks against collisions during transportation, would remain in place during transfer operations at the intermodal transfer station.

DOE performed an accident screening process to identify credible accidents that could occur at an intermodal transfer station with the potential for compromising the integrity of the casks and releasing radioactive material. The external events listed in Table J-49 were considered, along with an evaluation of their potential applicability.

As indicated from Table J-49, the only accident-initiating event identified from among the feasible external events was the aircraft crash. Such events would be credible only for casks being handled or on transport vehicles at an intermodal transfer station in the Las Vegas area (Apex/Dry Lake or Sloan/Jean). For a station in the Las Vegas area, an aircraft crash would be from either commercial aircraft operations at McCarran airport or military operations from Nellis Air Force Base.

Among the internal events, the only potential accident identified was a drop of the cask during transfer operations. This accident would bound the other events considered, including drops from the railcar or truck (less fall height would be involved than during the transfer operations). Collisions, derailments, and other accidents involving the transport vehicles at the intermodal transfer would not damage the casks due to the requirement that they be able to withstand high-speed impacts and the low velocities of the transport vehicles at the intermodal transfer station.

Sabotage events were also considered as potential accident-initiating events at an intermodal transfer station. Section J.1.5 evaluates such events.

Table J-49. Screening analysis of external events considered potential accident initiators at intermodal transfer station.

Event	Applicability
Aircraft crash	Retained for further evaluation
Avalanche	(a)
Coastal erosion	(a)
Dam failure	See flooding
Debris avalanching	(a)
Dissolution	(b)
Epeirogenic displacement (tilting of the earth's crust)	(c)
Erosion	(b)
Extreme wind	(c)
Extreme weather	(e)
Fire (range)	(b)
Flooding	(d)
Denudation	(b)
Fungus, bacteria, algae	(b)
Glacial erosion	(b)
High lake level	(b)
High tide	(a)
High river stage	See flooding
Hurricane	(a)
Inadvertent future intrusion	(b)
Industrial activity	Bounded by aircraft crash
Intentional future intrusion	(b)
Lightning	(c)
Loss of off/on site power	(c)
Low lake level	(b)
Meteorite impact	(e)
Military activity	Retained for further evaluation
Orogenic diastrophism	(e)
Pipeline accident	(b)
Rainstorm	See flooding
Sandstorm	(c)
Sedimentation	(b)
Seiche	(a)
Seismic activity, uplifting	(c)
Seismic activity, earthquake	(c)
Seismic activity, surface fault	(c)
Seismic activity, subsurface fault	(c)
Static fracturing	(b)
Stream erosion	(b)
Subsidence	(c)
Tornado	(c)
Tsunami	(a)
Undetected past intrusions	(b)
Undetected geologic features	(b)
Undetected geologic processes	(c)
Volcanic eruption	(e)
Volcanism, magmatic activity	(e)
Volcanism, ash flow	(c)
Volcanism, ash fall	(b)
Waves (aquatic)	(a)

a. Conditions at proposed sites do not allow event.
b. Not a potential accident initiator.
c. Bounded by cask drop accident considered in the internal events analysis.
d. Shipping cask designed for event.
e. Not credible; see evaluation for repository.

Accident Analysis

1. *Cask Drop Accident.* The only internal event retained after the screening process was a failure of the gantry crane (due to mechanical failure or human error) during the transfer of a shipping cask from a railcar to a heavy-haul truck. The maximum height between the shipping cask and the ground during the transfer operation would be less than 6 meters (19 feet) (TRW 1999a, Heavy-Haul Files, Item 11). The casks would be designed to withstand a 9-meter (30-foot) drop. Therefore, the cask would be unlikely to fail during the event, especially because the impact energy from the 6-meter drop would be only 65 percent of the minimum design requirement.

2. *Aircraft Crash Accident.* Two of the three intermodal transfer station locations are near airports that handle large volumes of air traffic. The Apex/Dry Lake location is about 16 kilometers (10 miles) northeast of the Nellis Air Force Base runways. Between 60,000 and 67,000 takeoffs and landings occur at Nellis Air Force Base each year (Luedke 1997, all). The Sloan/Jean intermodal transfer area begins about 16 kilometers southwest of McCarran International Airport in Las Vegas. In 1996, McCarran had an average of 1,300 daily aircraft operations (Best 1998, all). Because of the large number of aircraft operations at these airports, the probability of an aircraft crash on the proposed intermodal transfer station could be within the credible range. To assess the consequences of an aircraft crash, an analysis evaluated the ability of large aircraft projectiles [jet engines and jet engine shafts (DOE 1996b, page 58)] to penetrate the shipping casks. The analysis used a recommended formula (DOE 1996b, page 69) for predicting the penetration of steel targets, as follows:

$$T^{1.5} = 0.5 \times M \times V^2 \div 17{,}400 \times K_s \times D^{1.5}$$

where:

- T = predicted thickness to just perforate a steel plate (inches)
- M = projectile mass (weight/gravitational acceleration)
- V = projectile impact velocity (feet per second)
- K_s = constant depending on the grade of steel (usually about 1.0)
- D = projectile diameter (inches)

The projectile characteristics listed in Table J-50 are from Davis, Strenge, and Mishima (1998, all). The velocity used is about 130 meters (427 feet) per second, which is representative of aircraft velocities near airports (maximum velocity during takeoff and landing operations). A higher velocity [about 180 meters (590 feet) per second] was assumed for the projectile found to be limiting in terms of ability to penetrate (commercial engine shaft) to provide perspective on the influence of velocity on the penetration thickness. Table J-51 lists the results of the penetration calculation.

Table J-50. Projectile characteristics.[a]

Aircraft	Engine weight (kilograms)[b]	Engine diameter (centimeters)[c]
Small military	420	71
Commercial	3,900	270

a. Source: Davis, Strenge, and Mishima (1998, Table 1).
b. To convert kilograms to pounds, multiply by 2.2046.
c. To convert centimeters to inches, multiply by 0.3937.

The results indicate that none of the aircraft projectiles considered would penetrate the shipping casks, which would have metal shield walls about 18 centimeters (7 inches) thick (JAI 1996, all).

This evaluation found no credible accidents with the potential for radioactive release at an intermodal transfer station.

Table J-51. Results of aircraft projectile penetration analysis.[a]

Projectile	Velocity (meters per second)[b]	Penetration thickness (centimeters)[c,d]
Small military engine	130	2.5
Small military shaft	130	2.5
Commercial engine	130	3.0
Commercial shaft	130	3.7
Commercial shaft	180	5.9

a. Source: Davis, Strenge, and Mishima (1998, Table 2).
b. To convert meters to feet, multiply by 3.2808.
c. To convert centimeters to inches, multiply by 0.3937.
d. Penetration through steel plate.

J.3.4 IMPACTS IN NEVADA FROM INCIDENT-FREE TRANSPORTATION FOR INVENTORY MODULES 1 AND 2

This section presents the analysis of impacts to occupational and public health and safety in Nevada from incident-free transportation of spent nuclear fuel and high-level radioactive waste in Inventory Modules 1 and 2. The analysis assumed that the routes, population densities, and shipment characteristics (for example, radiation from shipping casks) for shipments under the Proposed Action and Inventory Modules 1 and 2 would be the same. The only difference was the projected number of shipments that would travel to the repository.

The following sections provide detailed information on the range of potential impacts to occupational and public safety and health from incident-free transportation of Modules 1 and 2 that result from legal-weight trucks and the 10 alternative transportation routes considered in Nevada. National impacts of incident-free transportation of Modules 1 and 2 incorporating Nevada impacts are discussed together with other cumulative impacts in Chapter 8.

J.3.4.1 Mostly Legal-Weight Truck Scenario

Tables J-52 and J-53 list estimated incident-free impacts in Nevada for the mostly legal-weight truck scenario for shipments of materials included in Inventory Modules 1 and 2.

J.3.4.2 Nevada Rail Implementing Alternatives

Table J-54 lists the range of estimated incident-free impacts in Nevada for the operation of a branch rail line to ship the materials included in Inventory Modules 1 and 2. It lists impacts that would result from operations for a branch line in each of the five possible rail corridors DOE is evaluating. These include the impacts of about 2,600 legal-weight truck shipments from commercial sites that could not use rail casks to ship spent nuclear fuel.

J.3.4.3 Nevada Heavy-Haul Truck Implementing Alternatives

Radiological Impacts

Intermodal Transfer Station Impacts. Involved worker exposures (the analysis assumed that the noninvolved workers would receive no radiation exposure and thus required no further analysis) would occur during both inbound (to the repository) and outbound (to the 77 sites) portions of the shipment campaign. DOE used the same involved worker level of effort it used in the analysis of intermodal transfer station worker industrial safety impacts to estimate collective involved worker radiological impacts (that is, 16 full-time equivalents per year). The collective worker radiation doses were adapted from a study (Smith, Daling and Faletti 1992, all) of a spent nuclear fuel transportation system, which

Table J-52. Population doses and radiological impacts from incident-free Nevada transportation for mostly legal-weight truck scenario – Modules 1 and 2.[a]

Category	Legal-weight truck shipments	Rail shipments of naval spent nuclear fuel[b]	Total[c]
Module 1			
Involved worker			
Collective dose (person-rem)	2,900	30	2,900
Estimated latent cancer fatalities	1.2	0.01	1.2
Public			
Collective dose (person-rem)	5,100	26	5,100
Estimated latent cancer fatalities	2.5	0.01	2.5
Module 2			
Involved worker			
Collective dose (person-rem)	3,000	40	3,000
Estimated latent cancer fatalities	1.2	0.02	1.2
Public			
Collective dose (person-rem)	5,300	30	5,300
Estimated latent cancer fatalities	2.6	0.02	2.6

a. Impacts are totals for shipments over 38 years.
b. Includes impacts at intermodal transfer stations.
c. Totals might differ from sums due to rounding.

Table J-53. Population health impacts from vehicle emissions during incident-free Nevada transportation for the mostly legal-weight truck scenario – Modules 1 and 2.[a]

Vehicle emission-related fatalities	Legal-weight truck shipments	Rail shipments of naval spent nuclear fuel[b]	Total[c]
Module 1	0.01	0.0004	0.01
Module 2	0.01	0.0005	0.01

a. Impacts are totals for shipments over 38 years.
b. Includes heavy-haul truck shipments in Nevada.
c. Totals might differ from sums due to rounding.

Table J-54. Radiological and nonradiological impacts from incident-free Nevada transportation for the mostly rail scenario – Modules 1 and 2.[a]

Category	Legal-weight truck shipments	Rail shipments	Total[b]
Module 1			
Involved worker			
Collective dose (person-rem)	370	280 - 460	650 - 830
Estimated latent cancer fatalities	0.15	0.11 - 0.18	0.26 - 0.33
Public			
Collective dose (person-rem)	430	190 - 270	620 -700
Estimated latent cancer fatalities	0.22	0.09 - 0.14	0.31 - 0.36
Estimated vehicle emission-related fatalities	0.00019	0.004	0.0042

a. Impacts are totals for 38 years (2010 to 2048).
b. Totals might differ from sums due to rounding.

was also performed for the commercial sites. That study found that the collective worker doses that could be incurred during similar inbound and outbound transfer operations of a single loaded (with commercial spent nuclear fuel) and unloaded cask were approximately 0.027 and 0.001 person-rem per cask, respectively, as listed in Table J-55.

The analysis used these inbound and outbound collective dose factors to calculate the involved worker impacts listed in Table J-56 for Module 1 and Module 2 inventories in the same manner it used for

Table J-55. Collective worker doses (person-rem) from transportation of a single cask.[a,b]

Inbound	Inbound CD[b]	Outbound	Outbound CD
Receive transport vehicle and loaded cask. Monitor, inspect, unhook offsite drive unit, and attach onsite drive unit.	6.3×10^{-3}	Receive transport vehicle and empty cask. Monitor, inspect, unhook offsite drive unit, and attach onsite drive unit.	0.0
Move cask to parking area and wait for wash down station. Attach to carrier puller when ready.	1.4×10^{-3}	Move cask to parking area and wait for wash down station. Attach to carrier puller when ready.	5.4×10^{-4}
Move cask to receiving and handling area.	9.2×10^{-5}	Move cask to receiving and handling area.	8.0×10^{-5}
Remove cask from carrier and place on cask cart.	4.3×10^{-3}	Remove cask from carrier and place on cask cart.	2.2×10^{-4}
Connect onsite drive unit and move cask to inspection area; disconnect onsite drive unit.	7.0×10^{-4}	Connect onsite drive unit and move cask to inspection area; disconnect onsite drive unit.	3.3×10^{-5}
Hook up offsite drive unit, move to gatehouse, perform final monitoring and inspection of cask.	1.4×10^{-2}	Hook up offsite drive unit, move to gatehouse, perform final monitoring and inspection of cask.	8.3×10^{-5}
Notify appropriate organizations of the shipment's departure.	0.0	Notify appropriate organizations of the shipment's departure.	0.0
Total	**2.7×10^{-5}**	**Total**	**8.8×10^{-5}**

a. Adapted from Smith, Daling and Faletti (1992, Table 4.2).
b. Values are rounded to two significant figures; therefore, totals might differ from sums of values.
c. CD = collective dose (person-millirem per cask).

Table J-56. Doses and radiological health impacts to involved workers from intermodal transfer station operations – Modules 1 and 2.[a,b]

	Module 1		Module 2	
Group	Dose	Latent cancer fatality	Dose	Latent cancer fatality
Maximally exposed individual worker[c]	12	0.005	12	0.005
Involved worker population[d]	530	0.21	550	0.22

a. Includes estimated impacts from handling 300 shipments of U.S. Navy fuel that would be shipped by rail under the mostly legal-weight truck transportation scenario. DOE estimated the impacts from these shipments by adjusting the impacts from the approximately 19,300 shipments (9,650 x 2) that would pass through the intermodal transfer station under the mostly rail scenario.
b. Totals for 24 years of operations.
c. The estimated probability of a latent cancer fatality in an exposed individual.
d. The estimated number of latent cancer fatalities in an exposed involved worker population.

commercial power reactor spent nuclear fuel impacts. The number of inbound and outbound shipments for Module 1 and Module 2 inventories is from Section J.1.2. The worker impacts reflect two-way operations.

Incident-Free Transportation. Table J-57 lists the range of estimated incident-free impacts in Nevada for the use of heavy-haul trucks to ship the materials included in Inventory Modules 1 and 2. It lists impacts that would result from operations on each of the five possible highway routes in Nevada DOE is evaluating. These include impacts of about 2,600 legal-weight truck shipments from commercial sites that could not ship spent nuclear fuel using rail casks.

Table J-57. Radiological and nonradiological health impacts from incident-free transportation for the heavy-haul truck implementing alternatives – Modules 1 and 2.[a]

Category	Legal-weight truck shipments	Rail and heavy-haul truck shipments[b]	Total[c]
Involved worker			
Collective dose (person-rem)	370	830 - 1,000	1,200 - 1,400
Estimated latent cancer fatalities	0.15	0.33 - 0.40	0.48 - 0.55
Public			
Collective dose (person-rem)	430	1,200 - 3,200	1,600 - 3,700
Estimated latent cancer fatalities	0.22	0.60 - 1.6	0.82 -1.8
Estimated vehicle emission-related fatalities	0.00019	0.03	0.05

a. Impacts are totals for 38 years (2010 to 2048).
b. Includes impacts to workers at an intermodal transfer station.
c. Totals might differ from sums due to rounding.

J.3.5 IMPACTS IN NEVADA FROM TRANSPORTATION ACCIDENTS FOR INVENTORY MODULES 1 AND 2

The analysis assumed that the routes, population densities, and shipment characteristics (for example, assumed radioactive material contents of shipping casks) for the Proposed Action and Inventory Modules 1 and 2 would be the same. The only difference would be the projected number of shipments that would travel to the repository. As listed in Table J-1, Module 2 would include about 3 percent more shipments than Module 1.

J.3.5.1 Mostly Legal-Weight Truck Scenario

Radiological Impacts

The analysis estimated the radiological impacts of accidents in Nevada for the mostly legal-weight truck scenario for shipments of the materials included in Inventory Modules 1 and 2. The radiological health impacts associated with Module 1 would be 0.86 person-rem and for Module 2 would be 0.88 person-rem (see Table J-58). These impacts would occur over 34 years in a population of more than 1 million people who lived within 80 kilometers (50 miles) of the Nevada routes that DOE would use. This dose risk would lead to about 1 chance in 1,000 of an additional cancer fatality in the exposed population. For comparison, about 220,000 in a population of 1 million people would suffer fatal cancers from other causes (ACS 1998, page 10).

Traffic Fatalities

The analysis estimated traffic fatalities from accidents involving the transport of spent nuclear fuel and high-level radioactive waste by legal-weight trucks in Nevada for the mostly legal-weight truck scenario for shipments of the materials included in Inventory Modules 1 and 2. It estimated that there would be 0.9 fatality over 34 years for Module 1 and 0.93 fatality for Module 2 (see Table J-58). The estimate of traffic fatalities includes the risk of fatalities from 300 shipments of naval spent nuclear fuel.

J.3.5.2 Nevada Rail Implementing Alternatives

Industrial Safety Impacts

Table J-59 lists the estimated industrial safety impacts in Nevada for the operation of a branch rail line to ship the materials included in Inventory Modules 1 and 2. The table lists impacts that would result from operations for a branch line in each of the five possible rail corridors in Nevada that DOE is evaluating.

The representative workplace loss incidence rate for each impact parameter (as compiled by the Bureau of Labor Statistics) was used as a multiplier to convert the operations crew level of effort to expected

Table J-58. Accident radiological health impacts for Modules 1 and 2 – Nevada transportation.[a]

Transportation scenario	Dose risk (person-rem)	Latent cancer fatalities	Traffic fatalities
Legal-weight truck	0.88[b]	0.0004	0.9
Legal-weight truck for the mostly rail scenario	0.1	0.00006	0.1
Mostly rail (Nevada rail implementing alternatives)			
Caliente	0.02	8.7×10^{-6}	0.13
Carlin	0.03	1.6×10^{-5}	0.17
Sloan/Jean	0.11	5.3×10^{-5}	0.10
Apex/Dry Lake	0.01	7.0×10^{-6}	0.08
Caliente-Chalk Mountain	0.01	6.9×10^{-6}	0.09
Mostly rail (Nevada heavy-haul implementing alternatives)			
Caliente	0.34	1.7×10^{-4}	1.2
Caliente-Chalk Mountain	0.28	1.4×10^{-4}	0.65
Caliente-Las Vegas	1.02	5.1×10^{-4}	0.90
Apex/Dry Lake	0.94	4.7×10^{-4}	0.46
Jean	6.5	3.2×10^{-3}	0.49

a. Impacts over 38 years.
b. Estimates of dose risk are for the transportation of the materials included in Module 2. Estimates of dose risk for transportation of the materials in Module 1 would be slightly (about 3 percent) lower.

Table J-59. Rail corridor operation worker physical trauma impacts (Modules 1 and 2).

Worker group and impact category	Corridor				
	Caliente	Carlin	Chalk Mountain	Jean	Valley Modified
Involved workers					
TRC[a]	200	200	200	150	150
LWC[b]	110	110	110	82	82
Fatalities	0.4	0.4	0.4	0.3	0.3
Noninvolved workers[c]					
TRC	9	9	9	7	7
LWC	5	5	5	3	3
Fatalities	0.01	0.01	0.01	0.01	0.01
All workers (totals)[d]					
TRC	210	210	210	160	160
LWC	120	120	120	85	85
Fatalities	0.4	0.4	0.4	0.3	0.3
Traffic fatalities[e]	1.1	1.1	1.1	0.8	0.8

a. TRC = total recordable cases (injury and illness).
b. LWC = lost workday cases.
c. Noninvolved worker impacts are based on 25 percent of the involved worker level of effort.
d. Totals might differ from sums due to rounding.
e. Fatalities from accidents during commutes to and from jobs for involved and noninvolved workers.

industrial safety losses. The involved worker full-time equivalent multiples that DOE would assign to operate each rail corridor each year was estimated to be 36 to 47 full-time equivalents, depending on the corridor for the period of operations (scaled from cost data in TRW 1996, Appendix E). Noninvolved worker full-time equivalent multiples were unavailable, so DOE assumed that the noninvolved worker level of effort would be similar to that for the repository operations work force—about 25 percent of that for involved workers. The Bureau of Labor Statistics loss incidence rate for each total recordable case, lost workday, and fatality trauma category (for example, the number of total recordable cases per full-time equivalent) was multiplied by the involved and noninvolved worker full-time equivalent multiples to project the associated trauma incidence.

The involved worker total recordable case incidence rate, 170,000 total recordable cases in a workforce of 1,620,000 workers (0.11 total recordable case per full-time equivalent) reflects losses in the Trucking and Warehousing sector during 1996. The same Bureau of Labor Statistics period of record and industry sector was used to select the involved worker lost workday case incidence rate [96,000 lost workday cases in a workforce of 1,620,000 workers (0.06 lost workday case per full-time equivalent)]. The involved worker fatality incidence rate, 22 fatalities in a workforce of 100,000 workers (0.0002 fatality per full-time equivalent) reflects losses in the Transportation and Material Moving Occupations sector during the Bureau of Labor Statistics 1994-to-1995 period of record.

The noninvolved worker incidence rate of 53,000 total recordable cases in a workforce of 2,870,000 workers (0.02 total recordable case per full-time equivalent) reflects losses in the Engineering and Management Services sector during the Bureau of Labor Statistics 1996 period of record. DOE used the same period of record and industry sector to select the noninvolved worker lost workday case incidence rate [22,000 lost workday cases in a workforce of 2,870,000 workers (0.01 lost workday case per full-time equivalent)]. The noninvolved worker fatality incidence rate, 1.5 fatalities in a workforce of 100,000 workers (0.00002 fatality per full-time equivalent) reflects losses in the Managerial and Professional Specialties sector during the 1994-to-1995 period of record.

Table J-59 lists the results of these industrial safety calculations for the five candidate corridors under Inventory Modules 1 and 2. The table also lists estimates of the number of traffic fatalities that would occur in the course of commuting by workers to and from their construction and operations jobs. These estimates used national statistics for average commute distances [18.5 kilometers (11.5 miles) one-way (ORNL 1999, all)] and fatality rates for automobile traffic [1 per 100 million kilometers (1.5 per 100 million miles) (BTS 1998, all)].

Radiological Impacts of Accidents

The analysis estimated the radiological impacts of accident scenarios in Nevada for the Nevada rail implementing alternatives for shipments of the materials included in Inventory Modules 1 and 2. Table J-58 lists the radiological dose-risk and associated risk of latent cancer fatalities. The risks include accident risks in Nevada from approximately 2,600 legal-weight truck shipments from commercial sites that could not ship spent nuclear fuel in rail casks. The risks would occur over 34 years.

Traffic Fatalities

Traffic fatalities from accidents involving transport of spent nuclear fuel and high-level radioactive waste by rail in Nevada were estimated for the Nevada rail implementing alternatives for shipments of materials included in Inventory Modules 1 and 2. Table J-58 lists the estimated number of fatalities that would occur over 34 years for a branch rail line along each of the five possible rail corridors. These estimates include the risk of fatalities from about 2,600 legal-weight truck shipments from commercial generators that could not ship spent nuclear fuel in rail casks.

J.3.5.3 Nevada Heavy-Haul Truck Implementing Alternatives

Industrial Safety Impacts

Tables J-60 and J-61 list the estimated industrial safety impacts in Nevada for operations of heavy-haul trucks (principally highway maintenance safety impacts) and operation of an intermodal transfer station that would transfer loaded and unloaded rail casks between rail cars and heavy-haul trucks for shipments of the materials included in Inventory Modules 1 and 2. Table J-60 lists the estimated industrial safety impacts in Nevada for the operation of a heavy-haul route to the Yucca Mountain site. Table J-61 lists impacts that would result from the operation of an intermodal transfer station for any of the five possible routes DOE is evaluating that heavy-haul trucks could use in Nevada.

Table J-60. Industrial health impacts from heavy-haul truck route operations (Modules 1 and 2).

Worker group and impact category	Corridor				
	Caliente	Caliente-Chalk Mountain	Caliente-Las Vegas	Sloan/ Jean	Apex/Dry Lake
Involved workers					
TRC[a]	460	460	420	250	250
LWC[b]	250	250	230	140	140
Fatalities	0.8	0.8	0.8	0.5	0.5
Noninvolved workers[c]					
TRC	21	21	19	11	11
LWC	11	11	10	6	6
Fatalities	0.02	0.02	0.02	0.01	0.01
All workers (totals)[d]					
TRC	480	480	440	260	260
LWC	260	260	240	150	150
Fatalities	0.82	0.82	0.82	0.5	0.5
Traffic fatalities[e]	2.0	2.0	1.9	1.3	1.3

a. TRC = total recordable cases (injury and illness).
b. LWC = lost workday cases.
c. Noninvolved worker impacts are based on 25 percent of the involved worker level of effort.
d. Totals might differ from sums due to rounding.
e. Fatalities from accidents during commutes to and from jobs for involved and noninvolved workers.

Table J-61. Annual physical trauma impacts to workers from intermodal transfer station operations (Module 1 or 2).

Involved workers			Noninvolved workers[a]			All workers		
TRC[b]	LWC[c]	Fatalities	TRC	LWC	Fatalities	TRC	LWC	Fatalities
112	60	0.2	5	2	0.0	116	62	0.2

a. The noninvolved worker impacts are based on 25 percent of the involved worker level of effort.
b. TRC = total recordable cases of injury and illness.
c. LWC = lost workday cases.

Radiological Impacts of Accidents

The analysis estimated the radiological impacts of accidents in Nevada for the Nevada heavy-haul truck implementing alternatives for shipments of the materials included in Inventory Modules 1 and 2.

Table J-58 lists the radiological dose-risk and associated risk of latent cancer fatalities. The risks include accident risks in Nevada from approximately 2,600 legal-weight truck shipments from commercial generating sites that could not ship spent nuclear fuel in rail casks. The risk would occur over 34 years.

Traffic Fatalities

The analysis estimated traffic fatalities from accidents involving the transport of spent nuclear fuel and high-level radioactive waste (including the rail portion of transportation to and from an intermodal transfer station) in Nevada for the heavy-haul truck implementing alternatives for shipments of the materials included in Inventory Modules 1 and 2. Table J-58 lists the estimated number of fatalities that would occur over 34 years for a branch rail line and for each of the five possible routes for heavy-haul trucks. The estimate for traffic fatalities includes the risk of fatalities from about 2,600 legal-weight truck shipments from commercial generators that could not ship spent nuclear fuel in rail casks.

J.3.6 IMPACTS FROM TRANSPORTATION OF OTHER MATERIALS

Other types of transportation activities associated with the Proposed Action would involve shipments of materials other than the spent nuclear fuel and high-level radioactive waste discussed in previous sections. These activities would include the transportation of people. This section evaluates occupational and public health and safety and air quality impacts from the shipment of:

- Construction materials, consumables, and personnel for repository construction and operation, including disposal containers
- Waste including low-level waste, construction and demolition debris, sanitary and industrial solid waste, and hazardous waste
- Office and laboratory supplies, mail, and laboratory samples

The analysis includes potential impacts of transporting these materials for the case in which DOE would not build a rail line to the proposed repository, because the larger number of truck shipments would lead to higher impacts than those for rail shipments, as discussed above. In addition, because the construction schedule for a new rail line would coincide with the schedule for the construction of repository facilities, trucks would deliver materials for repository construction.

Rail service would benefit the delivery of 10,000 disposal containers from manufacturers. Two 33,000-kilogram (about 75,000-pound) disposal containers and their 700-kilogram (about 1,500-pound) lids (TRW 1999b, Request #027) would be delivered on a railcar—a total of 5,000 railcar deliveries over the 24-year period of the Proposed Action. These containers would be delivered to the repository along with shipments of spent nuclear fuel and high-level radioactive waste or separately on supply trains along with shipments of materials and equipment.

If rail service was not available, disposal container components that would weigh as much as 34 metric tons (37.5 tons) would be transported to Nevada by rail and transferred to overweight trucks for shipment to the repository site. In this event, 10,000 overweight truck shipments would move the containers from a railhead to the site. The State of Nevada routinely provides permits to motor carriers for overweight, overdimension loads if the gross vehicle weight does not exceed 58.5 metric tons (64.5 tons) (TRW 1999b, Request #046).

J.3.6.1 Transportation of Personnel and Materials to Repository

The following paragraphs describe impacts that would result from the transportation of construction materials, consumables, disposal containers, supplies, mail, laboratory samples, and personnel to the repository site during the construction, operation and monitoring, and closure phases.

Human Health and Safety

Most construction materials, construction equipment, and consumables would be transported to the Yucca Mountain site on legal-weight trucks. Heavy and overdimensional construction equipment would be delivered by trucks under permits issued by the Nevada Department of Transportation. DOE estimates that about 42,000 truck shipments over 5 years would be necessary to transport materials, supplies, and equipment to the site during the construction phase.

In addition to construction materials, supplies, equipment, and disposal containers, trucks would deliver consumables to the repository site. These would include diesel fuel, cement, and other materials that would be consumed in daily operations. About 13,000 semitrailer truck shipments would occur during each year of operation. Similarly, there would be an estimated 1,000 semitrailer truck shipments during each year of monitoring and 1,200 each year during closure operations.

Over the 24-year period of the Proposed Action, the repository would receive about 300,000 truck shipments of supplies, materials, equipment, disposal containers, and consumables, including cement and other materials used in underground excavation. Most of these shipments would originate in the Las Vegas metropolitan area. In addition, an estimated 54,000 shipments of office and laboratory supplies and equipment, mail, and laboratory samples would occur during the 24 years of operation. A total of about 21 million vehicle kilometers (13 million vehicle miles) of travel would be involved. Impacts would include vehicle emissions, consumption of petroleum resources, increased truck traffic on regional highways, and fatalities from accidents. Similarly, there would be about 76,000 shipments during the 76-year monitoring period after emplacement operations and 15,000 shipments during closure activities. The number of shipments during shorter or longer monitoring periods would be proportionately fewer or larger. Table J-62 summarizes these impacts.

Table J-62. Human health and safety impacts from shipments of material to the repository.[a]

Phase	Kilometers[b] traveled (millions)	Traffic fatalities	Fuel consumption (thousands of liters)[c]	Vehicle emissions-related fatalities
Construction	8.2 - 9.9	0.14 - 0.17	1,900 - 2,300	0.0006 - 0.0007
Operation and monitoring				
Emplacement and development	29 - 66	0.5 - 1.1	7,000 - 15,000	0.002 - 0.005
Monitoring				
26 years	6.5	0.1	1,500	0.0005
76 years	19	0.3	4,500	0.0014
276 years	69	1.2	16,000	0.005
Closure	4.1	0.1	1,000	0.0003

a. Impacts are totals for 24 years of operations.
b. To convert kilometers to miles, multiply by 0.62137.
c. To convert liters to gallons, multiply by 0.26418.

During the construction phase, many employees would use their personal automobiles to travel to construction areas on the repository site and to highway or rail line construction sites. The estimated peak level of direct employment during 5 years of repository construction would be 1,035 workers. Current Nevada Test Site employees can ride DOE-provided buses to and from work; similarly, buses probably would be available for repository construction workers, which would reduce the number of vehicles traveling to the site each day by approximately a factor of 8. Table J-63 summarizes the anticipated number of traffic-accident-related injuries and fatalities and the estimated consumption of gasoline that would occur from this travel activity. The greatest impact of this traffic would be added congestion at the northwestern Las Vegas Beltway interchange with U.S. Highway 95. Current estimates call for traffic at this interchange during rush hours to be as high as 1,000 vehicles an hour (Clark County 1997, Table 3-12, page 3-43). The additional traffic from repository construction, an estimated 500 vehicles per hour, would add about 50 percent to traffic volume at peak rush hour and would contribute to congestion although congestion in this area would be generally low.

The average level of employment during repository operations would be about 2,700 workers. As mentioned above, DOE provides bus service from the Las Vegas area to and from the Nevada Test Site. Table J-63 summarizes the anticipated number of traffic-accident-related fatalities and the estimated consumption of gasoline that would occur from this travel activity. The greatest impact of this traffic would be increased congestion at the northwestern Las Vegas Beltway interchange with U.S. 95. As many as 500 vehicles an hour at peak rush hour would contribute to the congestion. Approximately

Table J-63. Health impacts from transportation of construction and operations workers.[a]

Phase	Kilometers[b] traveled (in millions)	Traffic fatalities	Fuel consumption (thousands of liters)[c]	Vehicle emissions-related fatalities
Construction	36.3 - 44.4	0.5 - 0.6	400 - 500	0.0026 - 0.0032
Operation and monitoring				
Emplacement and development	240 -300	3.2 - 4.0	2,600 - 3,300	0.017 - 0.022
Monitoring (76 years)	62.2	0.8	680	0.0045
Closure	20.2 - 42.7	0.3 - 0.6	220 - 470	0.0015 - 0.0031

a. Impacts are totals for 24 years for operations.
b. To convert kilometers to miles, multiply by 0.62137.
c. To convert liters to gallons, multiply by 0.26418.

150 people would be employed during monitoring and about 500 would be employed during closure. The number of vehicles associated with these levels of employment would contribute negligibly to congestion.

Table J-64 lists the impacts associated with the delivery of fabricated disposal container components from a manufacturing site to the repository. A total of 10,000 containers would be delivered; if a rail line to Yucca Mountain was not available, the mode of transportation would be a combination of rail and overweight truck. The analysis assumes that the capacity of each railcar would be two containers and that the capacity of a truck would be one container, so there would be 5,000 railcar shipments to Nevada and 10,000 truck shipments to the Yucca Mountain site. The analysis estimated impacts for one national rail route representing a potential route from a manufacturing facility to a Nevada rail siding. The analysis estimated the impacts of transporting the containers from this siding over a single truck route—the Apex/Dry Lake route analyzed for the transportation of spent nuclear fuel and high-level radioactive waste by heavy-haul trucks. Although the actual mileage from a manufacturing facility could be shorter, DOE decided to select a distance that represents a conservative estimate [4,439 kilometers (2,758 miles)]. The impacts are split into two subcategories—health effects from vehicle emissions and fatalities from transportation accidents.

Table J-64. Impacts of disposal container shipments for Proposed Action.[a]

Type of shipment	Number of shipments	Vehicle emissions-related health effects	Traffic fatalities
Rail and truck	5,000 rail/10,000 truck	0.14	0.8

a. Impacts are totals for 24 years of operations.

Air Quality

The exhaust from vehicles involved in the transport of personnel and materials to the repository would emit carbon monoxide, nitrogen dioxide, and particulate matter (PM_{10}). Because carbon monoxide is the principal pollutant of interest for evaluating impacts caused by motor vehicle emissions, the analysis focused on it.

The analysis assumed that most of the personnel who would commute to the repository would reside in the Las Vegas area and that most of the materials would travel to the repository from the Las Vegas area. To estimate maximum potential emissions to the Las Vegas Valley airshed, which is in nonattainment for carbon monoxide (FHWA 1996, pages 3-53 and 3-54), the analysis assumed that all personnel and material would travel from the center of Las Vegas to the repository. Table J-65 lists the estimated annual amount of carbon monoxide that would be emitted to the valley airshed during the phases of the repository project and the percent of the corresponding threshold level.

As listed in Table J-65, the annual amount of carbon monoxide emitted to the nonattainment area would be below the threshold level during all phases of the repository. In the operation phase, the estimated annual amount of carbon monoxide emitted would be close (93 percent) to the threshold level. So, a more

Table J-65. Annual amount of carbon monoxide emitted to Las Vegas Valley airshed from transport of personnel and material to repository (kilograms per year)[a] for the Proposed Action.

Phase	Annual emission rate	GCR threshold level[b]
Construction	47,000	51
Operation and monitoring		
Operation period	85,000	93
Monitoring period	6,700	7.4
Closure	17,000	19

a. To convert kilograms to tons, multiply by 0.0011023.
b. GCR = General Conformity Rule emission threshold level for carbon monoxide is 91,000 kilograms (100 tons) per year.

detailed analysis and conformity analysis might be required to determine if mitigation would be needed to ensure that the additional emissions did not impede efforts in Nevada to bring the Las Vegas area into attainment for carbon monoxide.

For areas that are in attainment, pollutant concentrations in the ambient air probably would increase due to the additional traffic but, given the relatively small amount of traffic that passes through these areas, the additional traffic would be unlikely to cause the ambient air quality standards to be exceeded.

Noise

Traffic-related noise on major transportation routes used by the workforce would likely increase. The analysis of impacts from traffic noise assumed that the workforce would come from Nye County (20 percent) and Clark County (80 percent). During the period of maximum employment in 2015, an estimated daily maximum of 576 vehicles would pass through the Gate 100 entrance at Mercury during rush hour (DOE 1996c, page 4-45), compared to a baseline of 232 vehicles per hour. This would result in an increase in rush hour noise from 65.5 dBA to 69.5 dBA for the communities of Mercury and Indian Springs. The 4.4-dBA increase could be perceptible to the communities but, because of the short duration, would be unlikely to result in an adverse response.

J.3.6.2 Impacts of Transporting Wastes from the Repository

During repository construction and operations, DOE would ship waste and sample material from the repository. The waste would include hazardous, mixed, and low-level radioactive waste. Samples would include radioactive and nonradioactive hazardous materials shipped to laboratories for analysis. In addition, nonhazardous solid waste could be shipped from the repository site to the Nevada Test Site for disposal. However, as noted in Chapter 2, DOE proposes to include an industrial landfill on the repository site. Table J-66 summarizes the maximum quantities of waste (generally from the uncanistered packaging scenario and the low thermal load scenario) that DOE would ship from the repository and the number of truck shipments.

Occupational and Public Health and Safety

The quantities of hazardous waste that DOE would ship to approved facilities off the Nevada Test Site would be relatively small and would present little risk to public health and safety. This waste could be shipped by rail (if DOE built a rail line to the repository site) or by legal-weight truck to permitted disposal facilities. The principal risks associated with shipments of these materials would be related to traffic accidents. These risks would include 0.01 fatality for the combined construction, operation and monitoring, and closure phases for hazardous wastes.

DOE probably would ship low-level radioactive waste by truck to existing disposal facilities on the Nevada Test Site. Although these shipments would not use public highways, DOE estimated their risks. As with shipments of hazardous waste, the principal risk in transporting low-level radioactive waste would be related to traffic accidents. Because traffic on the Nevada Test Site is regulated by the Nye County Sheriffs Department, DOE assumed that accident rates on the site are similar to those of secondary highways in Nevada. Low-level radioactive waste would not be present during the construction of the repository. Therefore, accidents involving such waste could occur only during the

Table J-66. Shipments of waste from the Yucca Mountain Repository.[a]

Waste	Construction		Operation and monitoring		Closure	
	Volume (cubic meters)[b]	Number of shipments	Volume (cubic meters)	Number of shipments	Volume (cubic meters)	Number of shipments
Hazardous[c]	990	60	6,100	340	630	8
Low-level radioactive[d]	0	0	68,000	1,800	3,500	2
Dual-purpose canisters[e]	0	0	30,000	6,600	0	0
Mixed[c]	0	0	23	2	0	0
Nonhazardous solid[f,g]	13,000	120	90,000	810	160,000	1,400

a. Source: Chapter 4, Section 4.1.12.
b. To convert cubic meters to cubic yards, multiply by 1.3079.
c. Shipment numbers based on 16.64 cubic meters per shipment.
d. Shipment numbers based on 38 cubic meters per shipment.
e. Shipment numbers based on 23 metric tons per shipment.
f. Shipment numbers based on cubic meters per shipment.
g. Includes construction and demolition debris and sanitary and industrial solid waste.

operation and monitoring and the closure phases, although most of this waste would be generated during the operation and monitoring phase. DOE estimates 0.05 traffic fatality from the transportation of low-level radioactive waste during the repository operation and monitoring and closure phases.

Air Quality

The quantities of hazardous waste that DOE would ship to approved facilities off the Nevada Test Site would be relatively small. Vehicle emissions due to these shipments would present little risk to public health and safety.

Biological Resources and Soils

The transportation of people, materials, and wastes during the construction, operation and monitoring, and closure phases of the repository would involve more than 1.6 billion vehicle-kilometers (1 billion vehicle-miles) of travel on highways in southern Nevada. This travel would use existing highways that pass through desert tortoise habitat. Individual desert tortoises probably would be killed. However, because populations of the species are low in the vicinity of the routes (Bury and Germano 1994, pages 57 to 72), few would be lost. Thus, the loss of individual desert tortoises due to repository traffic would not be likely to be a threat to the conservation of this species. In accordance with requirements of Section 7 of the Endangered Species Act, DOE would consult with the Fish and Wildlife Service and would comply with mitigation measures resulting from that consultation to limit losses of desert tortoises from repository traffic.

J.3.6.3 Impacts from Transporting Other Materials and People in Nevada for Inventory Modules 1 and 2

The analysis evaluated impacts to occupational and public health and safety in Nevada from the transport of materials, wastes, and workers (including repository-related commuter travel) for construction, operation and monitoring, and closure of the repository that would occur for the receipt and emplacement of materials in Inventory Modules 1 and 2. The analysis assumed that the routes and transportation characteristics (for example, accident rates) for transportation associated with the Proposed Action and Inventory Modules 1 and 2 would be the same. The only difference would be the projected number of trips for materials, wastes, and workers traveling to the repository.

Table J-67 lists estimated incident-free (vehicle emissions) impacts and traffic (accident) fatality impacts in Nevada for the transportation of materials, wastes, and workers (including repository-related commuter travel) for the construction, operation and monitoring, and closure of the repository that would occur for the receipt and emplacement of the materials in Inventory Modules 1 and 2.

Table J-67. Impacts from transportation of materials, consumables, personnel, and waste for Modules 1 and 2.[a]

Category	Kilometers traveled[b]	Fatalities	Emission-related health effects
Materials	90 - 160	1.7 - 2.9	0.07 - 0.01
Personnel	490 - 650	4.9 - 6.5	0.04 - 0.05
Waste material (Module 1/Module 2)			
Hazardous	0.17/0.20	0.018/0.021	0.00001/0.00001
Low-level radioactive	0.75/0.86	0.10/0.12	0.001
Nonhazardous solid	0.66	0.066	0.00005
Dual-purpose canisters	35	1.5	0.24

a. Numbers are rounded.
b. To convert kilometers to miles, multiply by 0.62137.

Even with the increased transportation of the other materials included in Module 1 or 2, DOE expects that the transportation of materials, consumables, personnel, and waste to and from the repository would be minor contributors to all transportation on a local, state, and national level. Public and worker health impacts would be small from transportation accidents involving nonradioactive hazardous materials. On average, in the United States there is about 1 fatality caused by the hazardous material being transported for each 30 million shipments by all modes (DOT 1998a, page 1; DOT undated, Exhibit 2b).

J.3.6.4 Environmental Justice

The impacts of transporting people and materials other than spent nuclear fuel and high-level radioactive waste would be small and random. Because the number of shipments and commuter trips would be small in comparison to other commercial and commuter travel in southern Nevada and would use existing transportation facilities in the area, impacts to land use; air quality; hydrology; biological resources and soils; occupational and public health and safety; cultural resources; socioeconomics; noise; aesthetics; utilities, energy, and materials; and waste management would be small. In addition, due to the nearly random nature of accidents that would involve the transportation of materials and people, the probability of such an accident would be small in any location, minimizing the risk at a specific location. Furthermore, because potential accidents would be nearly random, impacts to minority or low-income populations and to Native Americans along the routes in Nevada would be unlikely to be disproportionately high and adverse.

Because there would be no adverse or disproportionate impacts from transportation of people and materials, a detailed environmental justice study is not required.

J.3.6.5 Summary of Impacts of Transporting Other Materials

Table J-68 summarizes the impacts of transporting other materials to the repository site for the Proposed Action.

Table J-68. Health impacts from transportation of materials, consumables, personnel, and waste for the Proposed Action.[a]

Category	Distance traveled (kilometers)[b]	Impact
Human health and safety		
Construction		
Materials	8,200,000 - 9,900,000	0.14 - 0.17 fatality
Personnel	36,300,000 - 44,400,000	0.5 - 0.6 fatality
Waste		
Hazardous	14,500	0.002 fatality
Low-level waste	--[c]	--
Nonhazardous	29,000	0.003 fatality
Canisters	--	--
Operation and monitoring		
Materials	57,000,000 - 94,000,000	1.0 - 1.6 fatalities
Personnel	300,000,000 - 360,000,000	4.0 - 4.8 fatalities[d]
Waste		
Hazardous	90,000	0.002 fatality
Low-level waste	435,000	0.008 fatality
Nonhazardous	196,000	0.003 fatality
Canisters	1,590,000	0.028 fatality
Closure		
Materials	4,400,000	0.1 fatality
Personnel	20,200,000 - 42,700,000	0.3 - 0.6 fatality
Waste		
Hazardous	9,200	0.001 fatality
Low-level waste	22,200	0.002 fatality
Nonhazardous	338,000	0.04 fatality
Canisters	0	--
Air quality		
Construction traffic	74,000,000	75 percent of Air Quality General Conformity Rule threshold for PM_{10}
Operation and monitoring traffic		
Operations	860,000,000	170 percent of carbon monoxide threshold
Monitoring	170,000,000	9 percent of carbon monoxide threshold
Closure traffic	1,000,000,000	30 percent of carbon monoxide threshold
Biological resources	1,000,000,000	Individual desert tortoises would be killed but kills would not be likely to be a threat to conservation of species
Noise	--	Small impacts unlikely to affect communities
Environmental justice	--	Traffic impacts unlikely to be high and disproportionate for minority or low income populations or populations of Native Americans

a. Numbers are rounded.
b. To convert kilometers to miles, multiply by 0.62137.
c. -- = none.
d. Monitoring for 76 years.

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Appendix K

Long-Term Radiological Impact Analysis for the No-Action Alternative

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APPENDIX K. LONG-TERM RADIOLOGICAL IMPACT ANALYSIS FOR THE NO-ACTION ALTERNATIVE

K.1 Introduction

This appendix provides detailed information related to the radiological impact analysis for No-Action Alternative Scenario 2, including descriptions of the conceptual models used for facility degradation, spent nuclear fuel and high-level radioactive waste material degradation, and data input parameters. In addition, this appendix discusses the computer programs and exposure calculations used. The methods described include summaries of models and programs used for radioactive material release, environmental transport, radiation dose, and radiological human health impact assessment. Although the appendix describes No-Action Scenario 1, it focuses primarily on the long-term (100 to 10,000 years) radiological impacts associated with Scenario 2.

NO-ACTION ALTERNATIVE SCENARIOS 1 AND 2

Under the Nuclear Waste Policy Act, the Federal Government has the responsibility to provide permanent disposal of spent nuclear fuel and high-level radioactive waste to protect the public's health and safety and the environment. DOE intends to comply with the terms of existing consent orders and compliance agreements on the management of spent nuclear fuel and high-level radioactive waste. However, the course that Congress, DOE, and the commercial nuclear utilities would take if there was no recommendation to use Yucca Mountain as a repository is highly uncertain.

In light of these uncertainties, it would be speculative to attempt to predict precise consequences. To illustrate one set of possibilities, however, DOE decided to focus the analysis of the No-Action Alternative on the potential impacts of two scenarios:

Scenario 1: Long-term storage of spent nuclear fuel and high-level radioactive waste at the current storage sites, with effective institutional control for at least 10,000 years.

Scenario 2: Long-term storage of spent nuclear fuel and high-level radioactive waste, with the assumption of no effective institutional control after approximately 100 years.

DOE recognizes that neither of these scenarios is likely to occur if there was a decision to not develop a repository at Yucca Mountain. However, the Department selected these two scenarios for analysis because they provide a baseline for comparison to the impacts from the Proposed Action and because they reflect a range of the potential impacts that could occur.

To permit a comparison of the impacts between the construction, operation and monitoring, and eventual closure of a proposed repository at Yucca Mountain and No-Action Scenario 2, the U.S. Department of Energy (DOE) took care to maintain consistency, where possible, with the modeling techniques used to conduct the *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998, all) and in the *Total System Performance Assessment – Viability Assessment (TSPA-VA) Analyses Technical Basis Document* (TRW 1998a,b,c,d,e,f,g,h,i,j,k, all) for the proposed repository (see Appendix I, Section I.1, for details). In pursuit of this goal, DOE structured this analysis to facilitate an impact comparison with the repository impact analysis. Important consistencies include the following:

- Identical evaluation periods (100 years and 10,000 years)

- Identical spent nuclear fuel and high-level radioactive waste inventories at the reference repository:
  - Proposed Action: 63,000 metric tons of heavy metal (MTHM) of commercial spent nuclear fuel; 2,333 MTHM of DOE spent nuclear fuel; 8,315 canisters of high-level radioactive waste; and 50 MTHM of surplus weapons-usable plutonium
  - Module 1: All Proposed Action materials, plus an additional 42,000 MTHM of commercial spent nuclear fuel; 167 MTHM of DOE spent nuclear fuel; and 13,965 canisters of high-level radioactive waste. This would result in a total of approximately 105,000 MTHM of commercial spent nuclear fuel; 2,500 MTHM of DOE spent nuclear fuel; and 22,280 canisters of high-level radioactive waste, plus 50 MTHM of surplus weapons-usable plutonium (see Appendix A, Figure A-2).

DEFINITION OF METRIC TONS OF HEAVY METAL

Quantities of spent nuclear fuel are traditionally expressed in terms of *metric tons of heavy metal* (typically uranium), without the inclusion of other materials such as cladding (the tubes containing the fuel) and structural materials. A metric ton is 1,000 kilograms (1.1 tons or 2,200 pounds). Uranium and other metals in spent nuclear fuel (such as thorium and plutonium) are called *heavy metals* because they are extremely dense; that is, they have high weights per unit volume. One metric ton of heavy metal disposed of as spent nuclear fuel would fill a space approximately the size of a typical household refrigerator.

- Consistent spent nuclear fuel and high-level radioactive waste corrosion and dissolution models
- Identical radiation dose and risk conversion factors
- Similar assumptions regarding the future habits and behaviors of population groups (that is, that they will not be much different from those of populations today)

For commercial facilities, the No-Action analysis estimated short- and long-term radiological impacts for Scenario 1 and short-term impacts for Scenario 2 during the first 100 years for facility workers and the public based on values provided by the U.S. Nuclear Regulatory Commission (NRC 1991a, page 21). For DOE facilities, radiological impacts for these periods under Scenarios 1 and 2 were estimated based on analysis by Orthen (1999, all). To ensure consistency with the repository impact analysis, the long-term facility degradation and environmental releases of radioactive materials were estimated by adapting Total System Performance Assessment process models developed to predict the behavior of spent nuclear fuel and high-level radioactive waste in the repository (Battelle 1998, pages 2.4 to 2.9).

Because DOE did not want to unduly influence the results to favor the repository, it used assumptions were that generally resulted in lower predicted impacts (rather than applying the bounding assumptions used in many of the repository impact analyses) if Total System Performance Assessment models were not available or not appropriate for this continuous storage analysis. For example, the No-Action Scenario 2 analysis took into account the protectiveness of the stainless-steel waste canister when estimating releases of radioactive material from the vitrified high-level radioactive waste; the Total System Performance Assessment assumed no credit for material protection or radionuclide retardation by the intact canister. This approach dramatically reduced the release rate of high-level radioactive waste materials to the environment, thereby resulting in lower estimated total doses and dose rates to the exposed populations. Conversely, in many instances the Total System Performance Assessment selected values for input parameters that defined ranges to ensure that there would be no underestimation of the associated impacts. Section K.4 discusses other consistencies and inconsistencies between the Total System Performance Assessment and the No-Action analysis.

The long-term impact analysis used recent climate and meteorological data, assuming they would remain constant throughout the evaluation period (Poe and Wise 1998, all). DOE recognizes that there could be considerable changes in the climate over 10,000 years (precipitation patterns, ice ages, global warming, etc.) but, to simplify the analysis, did not attempt to quantify climate changes. Section K.4.1.2 discusses the difficulties of modeling these changes and the potential effect on outcomes resulting from uncertainties associated with predicting potential future climatic conditions.

Although the repository Total System Performance Assessment used probabilistic process models to evaluate the transport of radioactive materials within Yucca Mountain and underlying groundwater aquifers, DOE used the deterministic computer program Multimedia Environmental Pollutant Assessment System (MEPAS; Buck et al. 1995, all) for the No-Action Scenario 2 analysis because of the need to model the transport of radioactive material. In addition, it discusses environmental pathways not present at the repository (for example, the movement of contaminants through surface water). The MEPAS program has been accepted and used by DOE and the Environmental Protection Agency for long-term performance assessments (Rollins 1998a, pages 1, 10, and 19).

PROBABILISTIC AND DETERMINISTIC ANALYSES

A *probabilistic* analysis represents data input to a model as a range of values that represents the uncertainty associated with the actual or true value. The probabilistic model randomly samples these input parameter distributions many times to develop a possible range of results. The range of results provides a quantitative estimate of the uncertainty of the results.

A *deterministic* analysis uses a best estimate single value for each model input and produces a single result. The deterministic analysis will usually include a separate analysis that addresses the uncertainty associated with each input and provides an assessment of impact these uncertainties could have on the model results.

Analyses can use both approaches to provide similar information regarding the uncertainty of the results.

K.2 Analytical Methods

This section describes the methodology used to evaluate the long-term degradation of the concrete facilities, steel storage containers, and spent nuclear fuel and high-level radioactive waste materials. In addition, it discusses the eventual release and transport of radioactive materials under Scenario 2. The institutional control assumed under Scenario 1 would ensure ongoing maintenance, repair and replacement of storage facilities, and containment of spent nuclear fuel and high-level radioactive waste. For this reason, assuming the degradation of engineered barriers and the release and transport of radioactive materials is not appropriate for Scenario 1. The Scenario 2 analysis assumed that the degradation process would begin at the time when there was no effective institutional control (that is, after approximately 100 years) and the facilities would no longer be maintained. This section also describes the models and assumptions used to evaluate human exposures and potential health effects, and cost impacts.

K.2.1 GENERAL METHODOLOGY

For the No-Action analysis, the facilities, dry storage canisters, cladding, spent nuclear fuel, and high-level radioactive waste material, collectively known as the *engineered barrier system*, were modeled using an approach consistent (to the extent possible) with that developed for the Viability Assessment (DOE 1998, Volume 3). These process models were developed to evaluate, among other things, the performance of the repository engineered barrier system in the underground repository environment. In this analysis, the process models were adapted whenever feasible to evaluate surface environmental conditions at commercial and DOE sites. These models are described below.

Figure K-1 shows the modeling of the degradation of spent nuclear fuel and high-level radioactive waste and the release of radioactive materials over long periods. Five steps describe the process of spent nuclear fuel and high-level radioactive waste degradation; a sixth step, facility radioactive material release, describes the amount and rate of precipitation that would transport the radioactive material or *dissolution products* to the environment. This section describes each process and the results. Additional details are provided in reference documents (Poe 1998a, all; Battelle 1998, all).

Environmental parameters important to the degradation processes include temperature, relative humidity, precipitation chemistry (pH and chemical composition), precipitation rates, number of rain-days, and freeze/thaw cycles. Other parameters considered in the degradation process describe the characteristics and behavior of the engineered barrier system, including barrier material composition and thickness. To simplify the analysis, the United States was divided into five regions (as shown in Figure K-2) for the purposes of estimating degradation rates and human health impacts (see Section K.2.1.6 for additional details).

Under the No-Action Alternative, commercial utilities would manage their spent nuclear fuel at 72 nuclear power generating facilities. DOE would manage its spent nuclear fuel and high-level radioactive waste at five DOE facilities [the Hanford Site (Region 5), the Idaho National Engineering and Environmental Laboratory (Region 5), Fort St. Vrain (Region 5), the West Valley Demonstration Project (Region 1), and the Savannah River Site (Region 2)]. The No-Action analysis evaluated DOE spent nuclear fuel and high-level radioactive waste at the commercial and DOE sites or at locations where Records of Decision have placed or will place these materials (for example, West Valley Demonstration Project spent nuclear fuel was evaluated at the Idaho National Engineering and Environmental Laboratory (60 *FR* 28680, June 1, 1995). Therefore, the No-Action analysis evaluated DOE aluminum-clad spent nuclear fuel at the Savannah River Site and DOE non-aluminum-clad fuel at the Idaho National Engineering and Environmental Laboratory. DOE evaluated most of the Fort St. Vrain spent nuclear fuel at the Colorado site. In addition, the analysis evaluated high-level radioactive waste at the West Valley Demonstration Project, the Idaho National Engineering and Environmental Laboratory, the Hanford Site, and the Savannah River Site.

K.2.1.1 Concrete Storage Module Degradation

The first process model analyzed degradation mechanisms related to failure of the concrete storage module. *Failure* is defined as the time when precipitation would infiltrate the concrete and reach the spent nuclear fuel or high-level radioactive waste storage canister. The analysis (Poe 1998a, Section 2.0) considered degradation due to exposure to the surrounding environment.

The primary cause of failure of surface-mounted concrete structures is freeze/thaw cycles that cause the concrete to crack and spall (break off in layers), which allows precipitation to enter the concrete, causing more freeze damage. *Freeze/thaw failure* is defined as the time when half of the thickness of the concrete is cracked and spalled. Some regions (coastal California, Texas, Florida, etc.) are essentially without the freeze/thaw cycle. In these locations the primary failure mechanism is chlorides in precipitation, which decompose the chemical constituents of the concrete into sand-like materials. This process progresses more slowly than the freeze/thaw process. Figure K-3 shows estimated concrete storage module failure times.

Below-grade concrete structures, such as those used to store some of the DOE spent nuclear fuel and most of the high-level radioactive waste, would be affected by the same concrete degradation mechanisms as surface facilities. Below grade, the freeze/thaw degradation would not be as great because the soil would moderate temperature fluctuations. The primary failure mechanism for below-grade facilities would be the loss of the above-grade roof, which would result in precipitation seeping around shield plugs. The

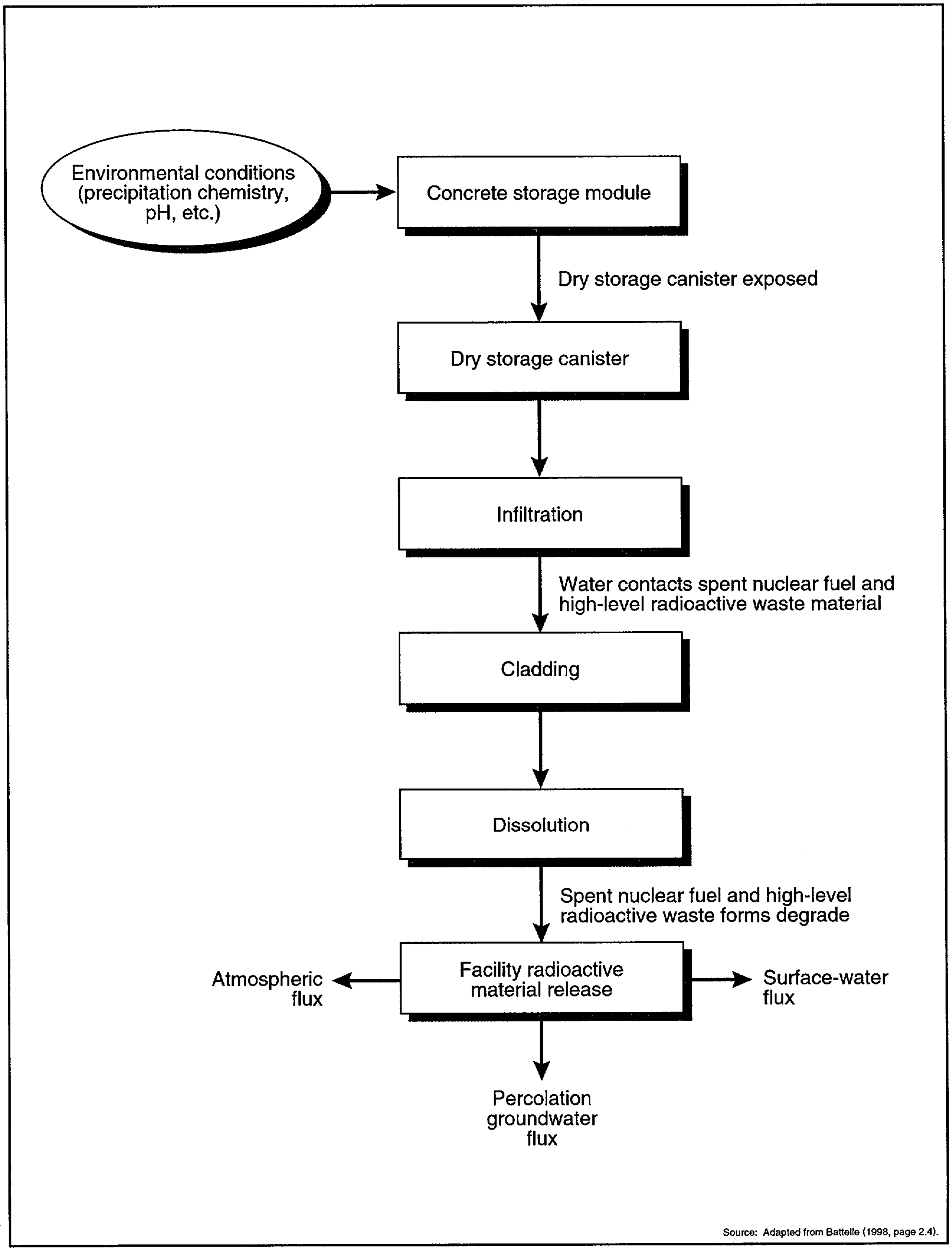
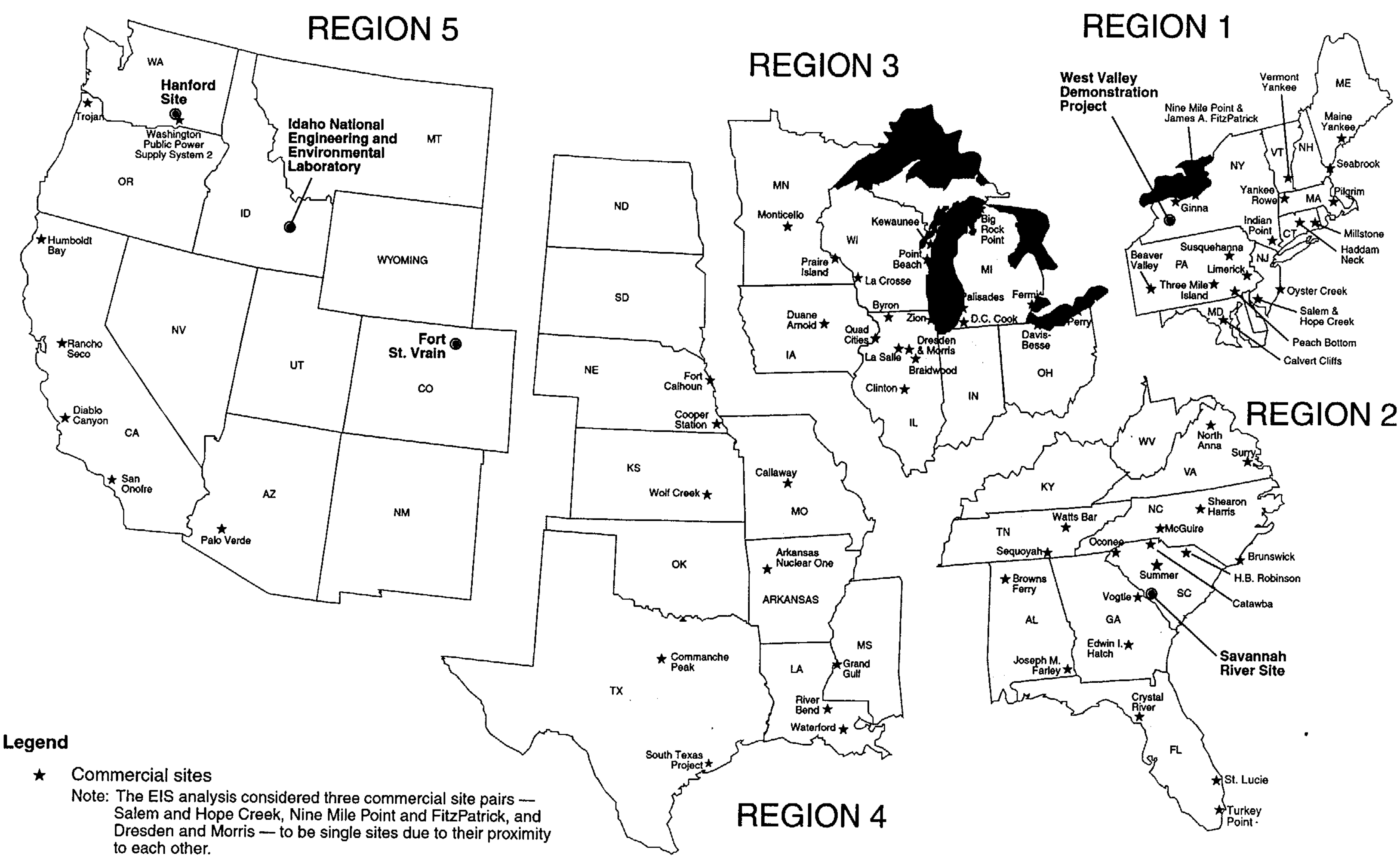


Figure K-1. Primary steps and processes involved in the degradation of the engineered barrier system.



Legend

★ Commercial sites

Note: The EIS analysis considered three commercial site pairs — Salem and Hope Creek, Nine Mile Point and FitzPatrick, and Dresden and Morris — to be single sites due to their proximity to each other.

◉ DOE sites

Note: The EIS analysis included the high-level radioactive waste and spent nuclear fuel at West Valley. The State of New York owns the high-level radioactive waste and the site. Under the West Valley Demonstration Project Act, DOE is responsible for solidifying and transporting the high-level radioactive waste off the West Valley site. DOE owns and is responsible for the spent nuclear fuel at the site.

Symbols do not reflect precise locations.

Note: None of the facilities evaluated are located in Alaska or Hawaii.

Source: Modified from Poe (1998b, page 2).

K-2. No-Action Alternative analysis regions.

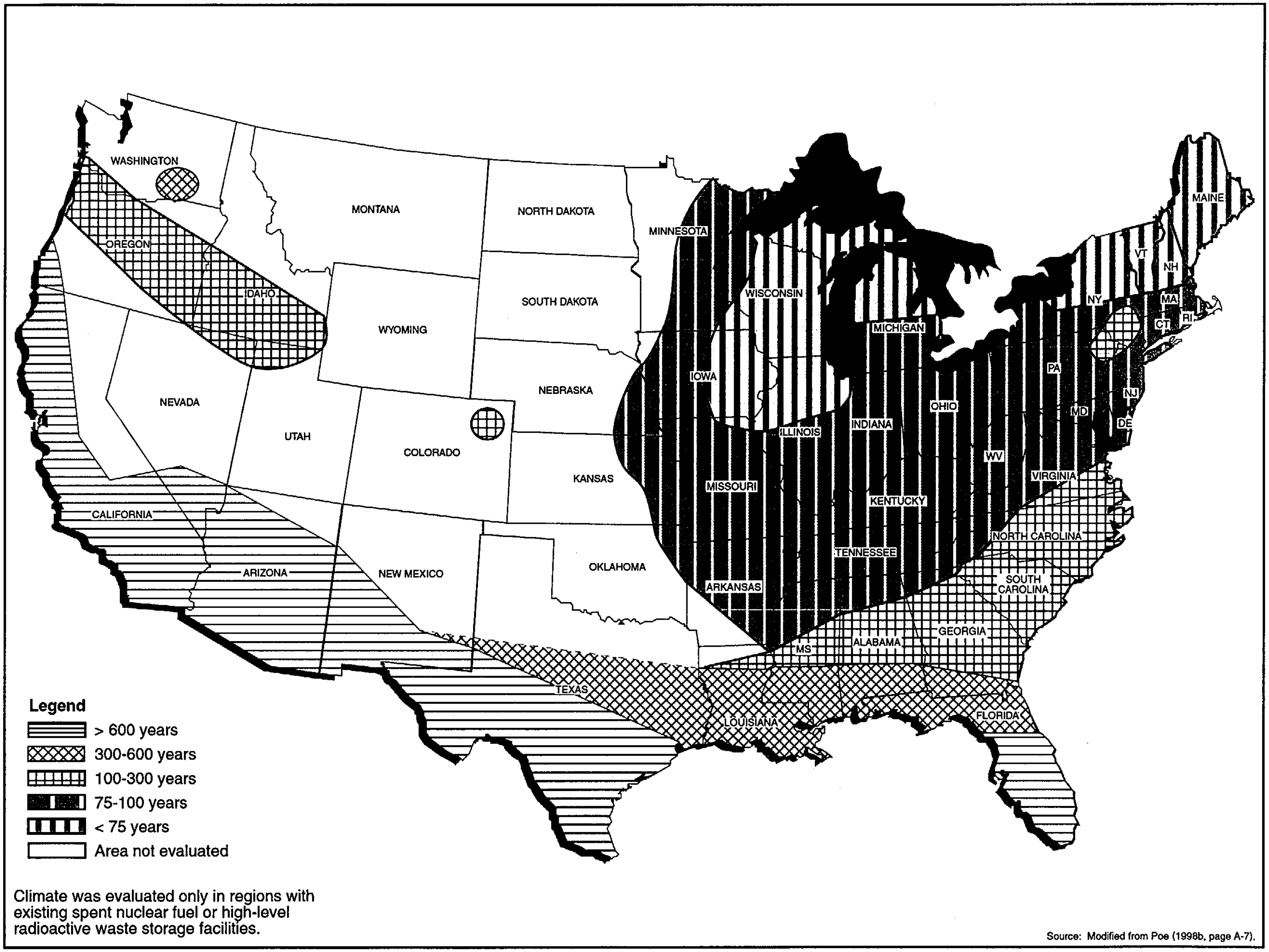


Figure K-3. Failure times for above-ground concrete storage modules.

analysis assumed that this would occur 50 years after the end of facility maintenance, and that this would be the reasonable life expectancy of a facility without maintenance and periodic repair (Poe 1998a, pages 4-6 to 4-19).

K.2.1.2 Storage Canister Degradation

The second process analyzed was spent nuclear fuel and high-level radioactive waste storage canister degradation. For commercial and DOE spent nuclear fuel, the analysis defined failure of the stainless-steel dry storage canister as the time at which precipitation penetrated the canister and wet the spent nuclear fuel. The analysis defined failure for the high-level radioactive waste as the time at which precipitation penetrated the canister. This is consistent with the repository definition that failure of the waste package would occur when water penetrated the package and came in contact with the contents. The stainless-steel model used for the No-Action analysis was consistent with the waste package inner layer corrosion model used for the repository Total System Performance Assessment (DOE 1998, Volume 3, Section 3.4) with the functional parameters modified to incorporate stainless-steel corrosion data (Section K.4.3.1 discusses the sensitivity of outcome to carbon-steel dry storage containers). In addition, the analysis used parameters appropriate for above-ground conditions, including temperature, meteorological data, and chemical constituents in the atmosphere and precipitation. Although inconsistent with the assumptions used for the Total System Performance Assessment, the analysis took credit for the protectiveness of the high-level radioactive waste canister because (1) it is the only container between the waste material and the environment and, (2) to ignore the protectiveness of this barrier would have resulted in a considerable overestimation of impacts. This approach is consistent with the decision, in the case of the No-Action Scenario 2 analysis, to provide a realistic radionuclide release rate where possible and to preclude the overestimation of the associated radiological human health impacts.

The primary determinants of stainless-steel corrosion for the different regions are the amount, the acidity, and the chloride concentration of the precipitation. The storage canisters degrade faster in the below-grade storage configuration than on the surface due to the higher humidity in the below-grade environment. The storage canisters degrade faster in the below-grade storage configuration than on the surface due to the higher humidity in the below-grade environment. The high-level radioactive waste canisters degrade faster than the spent nuclear fuel canisters because they are not as thick. The analysis evaluated three corrosion mechanisms—general corrosion, pitting corrosion, and crevice corrosion (Battelle 1998, Appendix A). Of the three, crevice corrosion would be the dominant failure mechanism for the regions analyzed. Corrosion rates and penetration times vary among the different regions of the country. The analysis calculated regional penetration times from the time at which it assumed that precipitation first would come in contact with the stainless steel. Table K-1 lists the results.

K.2.1.3 Infiltration

The third process analyzes infiltration of water to the spent nuclear fuel and high-level radioactive waste. The amount of water in contact with these materials would be directly related to the size of the dry storage canister footprint and the mean (average) annual precipitation at each storage site. The rate of precipitation varies throughout the Unites States from extremely low (less than 25 centimeters [10 inches] per year) in the arid portions of the west to high (more than 150 centimeters [60 inches] per year) along the Gulf Coast in the southeast (Table K-2, Figure K-4). Local precipitation rates were used to determine the amount of water available that could cause dry storage canister and cladding failure, and spent nuclear fuel and high-level radioactive waste material dissolution.

Table K-1. Time (years) after the assumed loss of effective institutional control at which first failures would occur and radioactive materials could reach the accessible environment.

Material	Region	Storage facility	Weather[a] protection lost	Canister[b] breached (initial material release)
Commercial spent nuclear fuel	1	Surface	100	1,400
	2	Surface	700	1,500
	3	Surface	170	1,100
	4	Surface	750	1,600
	5	Surface	3,500	5,400
DOE spent nuclear fuel	2	Surface	700	1,400
	5	Surface	50	1,400
	5	Below grade	50	800
High-level radioactive waste	1	Surface	100	1,200
	2	Below grade	50	500
	5	Below grade	50	700

a. Source: Adapted from Poe (1998b, Appendix A).
b. Source: Battelle (1998, data files, all); spent nuclear fuel dry storage or high-level radioactive waste canister.

Table K-2. Average regional precipitation.[a]

Region	Annual precipitation (centimeters)[b]	Percent of days with precipitation
1	110	30
2	130	29
3	80	33
4	110	31
5	30	24

a. Source: Adapted from Poe (1998b, Appendix A, pages A-13 to A-16).
b. To convert centimeters to inches, multiply by 0.3937.

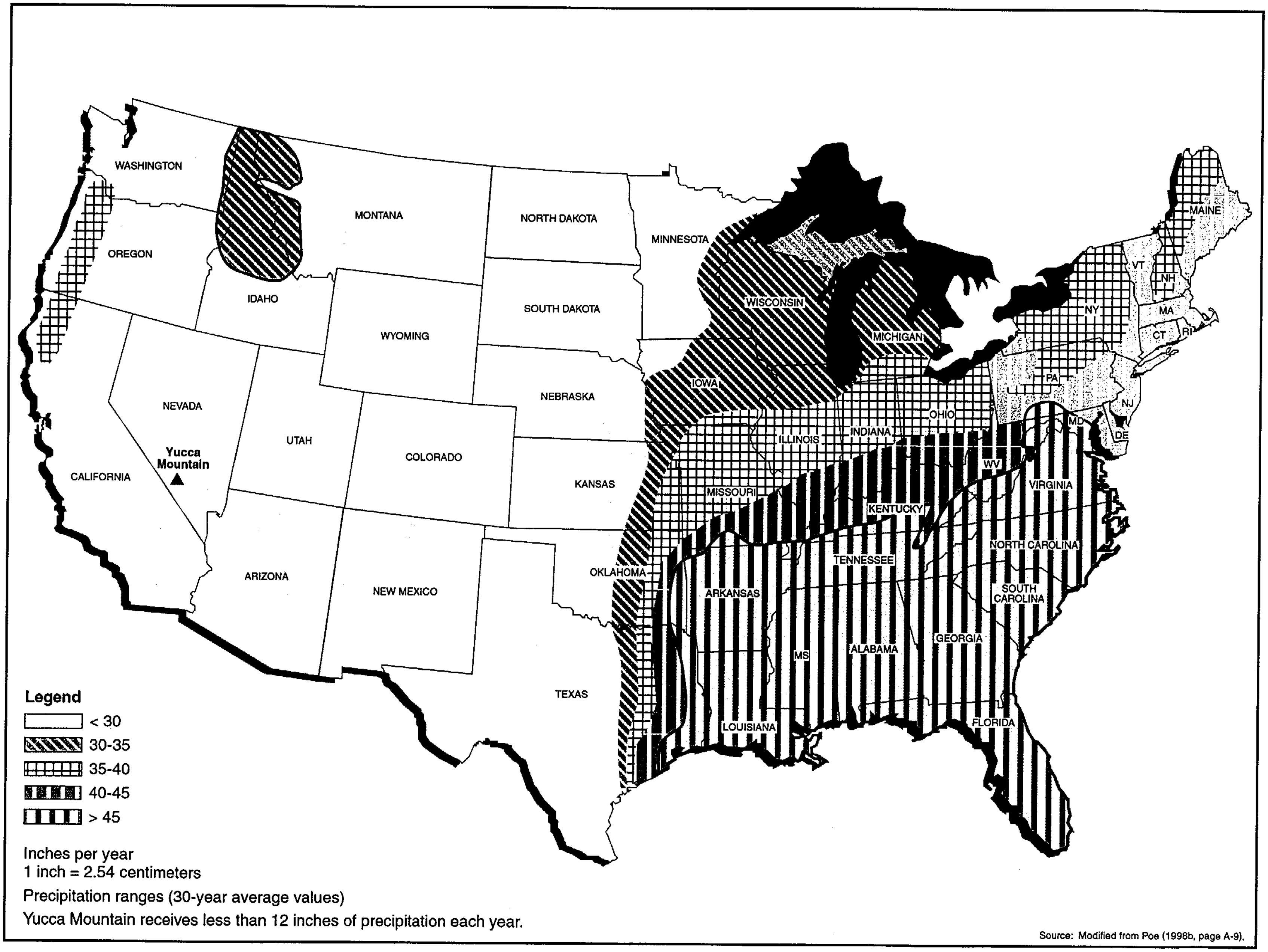
K.2.1.4 Cladding

The fourth process analyzed was failure of the cladding, which is a protective barrier, usually metal (aluminum, zirconium alloy, stainless steel, nickel-chromium, Hastalloy, tantalum, or graphite), surrounding the spent nuclear fuel material to contain radioactive materials. For spent nuclear fuel, cladding is the last engineered barrier to be breached before the radioactive material can begin to be released to the environment.

K.2.1.4.1 *Commercial Spent Nuclear Fuel Cladding*

The principal cladding material used on commercial spent nuclear fuel is zirconium alloy. About 1.2 percent (of MTHM) of commercial spent nuclear fuel is stainless-steel clad (Appendix A, Section A.2.1.5.3). To be consistent with the Total System Performance Assessment, this analysis evaluated two cladding failure mechanisms: (1) so-called *juvenile failures* (failures existing at the start of the analysis period), and (2) *new failures* (failures that occur during the analysis period due to conditions in the storage container). The analysis assumed that juvenile failures existed in 0.1 percent of the zirconium alloy-clad spent nuclear fuel and in all of the stainless-steel-clad fuel at the beginning of the analysis period, and that after failure the cladding would offer no further protection to the radioactive material [this is consistent with the Viability Assessment assumption (DOE 1998, Volume 3, page 3-97)].

Figure K-5 shows new failures (expressed as percent of commercial spent nuclear fuel over time) of zirconium alloy cladding, which were modeled using the median value assumed in the Total System Performance Assessment–Viability Assessment cladding abstraction (TRW 1998f, pages 6-19 to 6-54)



F[illegible] K-4. Precipitation ranges for regions with existing spent nu[illegible] fuel and high-level radioactive waste storage facilities.

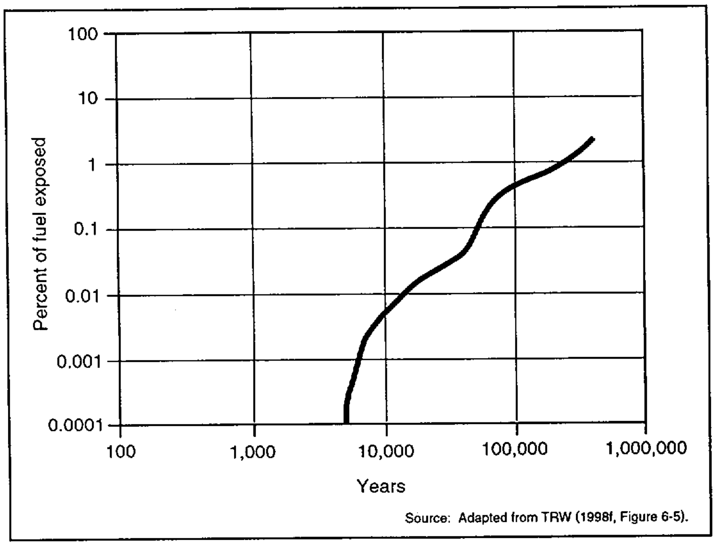


Figure K-5. Percent of commercial spent nuclear fuel exposed over time due to new failures.

for zirconium alloy corrosion. The Viability Assessment (DOE 1998, Volume 3, all) defines this information as a "fractional multiplier," which is calculated from the fraction of the failed fuel pin surface area. In the No-Action analysis, this corrosion is assumed to commence when weather protection afforded by the waste package is lost and the cladding is exposed to environmental precipitation. The Total System Performance Assessment-Viability Assessment also considers cladding failure from creep strain, delayed hydride cracking, and mechanical failure from rock falls. These additional mechanisms normally occur after the 10,000-year analysis period and are therefore not considered in the No-Action analysis. As shown in Figure K-5, during the 10,000-year analysis period, less than 0.01 percent of the zirconium alloy-clad spent nuclear fuel would be expected to fail. If the upper limit curve from Figure 4 of the Total System Performance Assessment-Viability Assessment cladding abstraction (TRW 1998f, pages 6-19 to 6-54) was used, the value could be as high as 0.5 percent of the zirconium alloy-clad spent nuclear fuel. The lower limit value from the Total System Performance Assessment-Viability Assessment cladding abstraction curve would be much less than 0.001 percent.

K.2.1.4.2 *DOE Spent Nuclear Fuel Cladding*

The composition and cladding materials of DOE spent nuclear fuel vary widely. The cladding assumption for the surrogate material used in this analysis is identical (no cladding credit) to the assumption used in the Total System Performance Assessment analysis (see Section K.4.3.2 for the discussion of uncertainty in relation to cladding).

K.2.1.5 Dissolution of Spent Nuclear Fuel and High-Level Radioactive Waste

The fifth process analyzed was the dissolution of the spent nuclear fuel and high-level radioactive waste. The rate of release of radionuclides from these materials would be related directly to the amount of surface area exposed to moisture, the quantity and chemistry of available water, and temperature. The Total System Performance Assessment process model, modified to reflect surface environmental conditions (temperature, relative humidity, etc.), was used to estimate release rates from the exposed spent nuclear fuel and high-level radioactive waste. The model and application to surface conditions is described in detail in Battelle (1998, pages 2.9 to 2.11).

K.2.1.5.1 *Commercial Spent Nuclear Fuel Dissolution*

Consistent with the repository impact analysis, this analysis estimated that new zirconium alloy failures would begin late in the 10,000-year period (see Figure K-5). As discussed in Section K.2.1.4.1, only 0.01 percent of the zirconium alloy-clad spent nuclear fuel would be likely to fail during the 10,000-year analysis period. Therefore, most of the exposed material considered in this analysis would result from juvenile failures of zirconium alloy- and stainless-steel-clad spent nuclear fuel.

K.2.1.5.2 *DOE Spent Nuclear Fuel Dissolution*

The analysis assumed that DOE spent nuclear fuel would be a metallic uranium fuel with zirconium alloy cladding (a representative or surrogate fuel that consisted primarily of N-Reactor fuel). Consistent with the repository input analysis, the No-Action Scenario 2 analysis takes no credit for the cladding. The analysis used the Total System Performance Assessment model for metallic uranium fuel, modified for surface environmental conditions, to predict releases of the DOE spent nuclear fuel.

K.2.1.5.3 *High-Level Radioactive Waste Dissolution*

Most high-level radioactive waste would be stored in below-grade concrete vaults. As discussed in Section K.2.1.1, these vaults would be exposed to precipitation as soon as weather protection was lost (the model assumed this would occur 50 years after loss of institutional control). After the loss of weather protection and failure of the stainless-steel canisters, the high-level radioactive waste would be exposed to precipitation. The environment in the underground vault would be humid and deterioration would occur. Thus, the material would be exposed to either standing water or humid conditions in the degrading vaults after the canister failed. The borosilicate glass deterioration model used in this analysis was the same as the Total System Performance Assessment model modified to reflect surface conditions (temperature and precipitation chemistry).

K.2.1.6 Regionalization of Sites for Analysis

The climate of the contiguous United States varies considerably across the country. The release rate of the radionuclide inventory would depend primarily on the interactions between environmental conditions (rainfall, freeze-thaw cycles) and engineered barriers. To simplify the analysis, DOE divided the country into five regions (see Figure K-2) (Poe 1998b, page 2).

The analysis assumed that a single hypothetical site in each region would store all the spent nuclear fuel and high-level radioactive waste existing in that region. Such a site does not exist but is a mathematical construct for analytical purposes. To ensure that the calculated results for the regional analyses reflect appropriate inventory, facility and material degradation, and radionuclide transport, the spent nuclear fuel and high-level radioactive waste inventories, engineered barriers, and environmental conditions for the hypothetical sites were developed from data for each of the existing sites in the given region. Weighting criteria to account for the amount and types of spent nuclear fuel and high-level radioactive waste at each site were used in the development of the environmental data for the regional site, such that the results of the analyses for the hypothetical site were representative of the sum of the results of each actual site if they had been modeled independently (Poe 1998b, page 1). If there are no storage facilities in a particular area of the country, the environmental parameters of that area were not evaluated.

Table K-3 lists the Proposed Action and Module 1 quantities of commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste in each of the five regions. The values in Table K-1 are the calculated results of failures of the various components of the protective engineered barriers and release of radioactive material in each region.

K.2.2 RADIONUCLIDE RELEASE

The sixth and final step in the process is the release of radioactive materials to the environment. The anticipated release rates (fluxes) were estimated in terms of grams per 70-year period (typical human life expectancy in the United States) of uranium dioxide, uranium metal, or borosilicate glass for commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste, respectively. To assess potential lifetime impacts on human receptors, the amount of fission products and transuranics associated

Table K-3. Proposed Action and Module 1 quantities of spent nuclear fuel (metric tons of heavy metal) and canisters of high-level radioactive waste in each geographic region.[a,b]

	Commercial spent nuclear fuel[c]								
	Region total[d]		With juvenile cladding failure		Stainless-steel cladding	DOE spent nuclear fuel[e]		High-level radioactive waste[f]	
Region	Proposed Action (MTHM)	Module 1 (MTHM)	Proposed Action (MTHM)	Module 1 (MTHM)	Proposed Action and Module 1[g] (MTHM)	Proposed Action (MTHM)	Module 1 (MTHM)	Proposed Action (canisters)	Module 1 (canisters)
1	17,000	27,000	16	27	410			300	300
2	19,000	32,000	19	32	0	30	45	6,000	6,200
3	15,000	23,000	15	23	170				
4	7,200	14,000	7	14	0				
5	5,400	10,000	5	9	140	2,300	2,455	2,000	15,500
Totals	**63,600**	**106,000**	**62**	**105**	**720**	**2,300**	**2,500**	**8,300**	**22,000**

a. Source: Appendix A.
b. Totals might differ from sums due to rounding.
c. All analyzed as stored on surface as shown on Chapter 2, Figures 2-36, 2-37, and 2-38.
d. Includes plutonium in mixed-oxide spent nuclear fuel, which is assumed to behave like other commercial spent nuclear fuel.
e. A representative or surrogate fuel that consisted primarily of N-reactor fuel.
f. Includes plutonium in can-in-canister.
g. Assumes failure of 100 percent of stainless-steel-clad when placed into dry storage.

with gram quantities of uranium dioxide, uranium metal, and borosilicate glass were calculated for approximately 140 consecutive 70-year average human lifetimes to determine releases from the 10,000-year analysis period. Weighting criteria were used to ensure appropriate contributions by the different types of spent nuclear fuel and the high-level radioactive waste in each region, as appropriate. The result was a single release rate for each region that accounted for the different materials (uranium dioxide, uranium metal, and borosilicate glass).

The radionuclide distributions in the spent nuclear fuel and high-level radioactive waste (Appendix A) were used for these analyses. These were expressed as radionuclide-specific curies for storage packages (assembly or canister). The curies per storage package were converted to curies per gram of uranium dioxide, uranium metal, or borosilicate glass (as described above for each spent nuclear fuel and high-level radioactive waste material). This radionuclide distribution was multiplied by release flux (curies of spent nuclear fuel and high-level radioactive waste material per 70-year period) after being corrected for decay and the ingrowth of decay products for various times after disposal. These corrections were determined using the ORIGEN computer program (ORNL 1991, all) for each of the approximately 140 consecutive 70-year human lifetimes to determine the release over the 10,000-year period. The results of the ORIGEN runs were used as input to the environmental transport program.

In addition to the 53 isotopes important to the repository long-term impact analysis specified in Appendix A, the No-Action Scenario 2 analysis considered 167 other isotopes in the

DEFINITIONS

Fission products: Elements produced when uranium atoms split in a nuclear reactor, some of which are radioactive. Examples are cesium, iodine, and strontium.

Transuranics: Radioactive elements, heavier than uranium, that are produced in a nuclear reactor when uranium atoms absorb neutrons rather than splitting. Examples of transuranics include plutonium, americium, and neptunium.

Curie: The basic unit of radioactivity. It is equal to the quantity of any radionuclide in which 37 billion atoms are decaying per second.

Specific activity: An expression of the number of curies of activity per gram of a given radionuclide. It is dependent on the half life and molecular weight of the nuclide.

light-water reactor radiological database (DOE 1992, Page 1.1-1). Of the 220 isotopes evaluated, six would contribute more than 99.5 percent of the total dose. Table K-4 lists these six isotopes along with technetium-99, which individually would contribute less than 0.003 percent of the total dose. Plutonium-239 and -240 would contribute more than 96 percent of the radiological impacts during the 10,000-year analysis period because of their very large dose conversion factors. Americium-241 and -243 would be minor contributors to the dose. Neptunium-237 and technetium-99 were of tertiary importance (Table K-4).

Table K-4. Radionuclides and relative contributions over 10,000 years to Scenario 2 impacts.[a]

Isotope	Percent of total dose
Americium-241	3.2
Americium-243	0.86
Neptunium-237	0.29
Plutonium-238	0.2
Plutonium-239	49.0
Plutonium-240	47.0
Technetium-99	< 0.003

a. Source: Toblin (1998a, page 6).

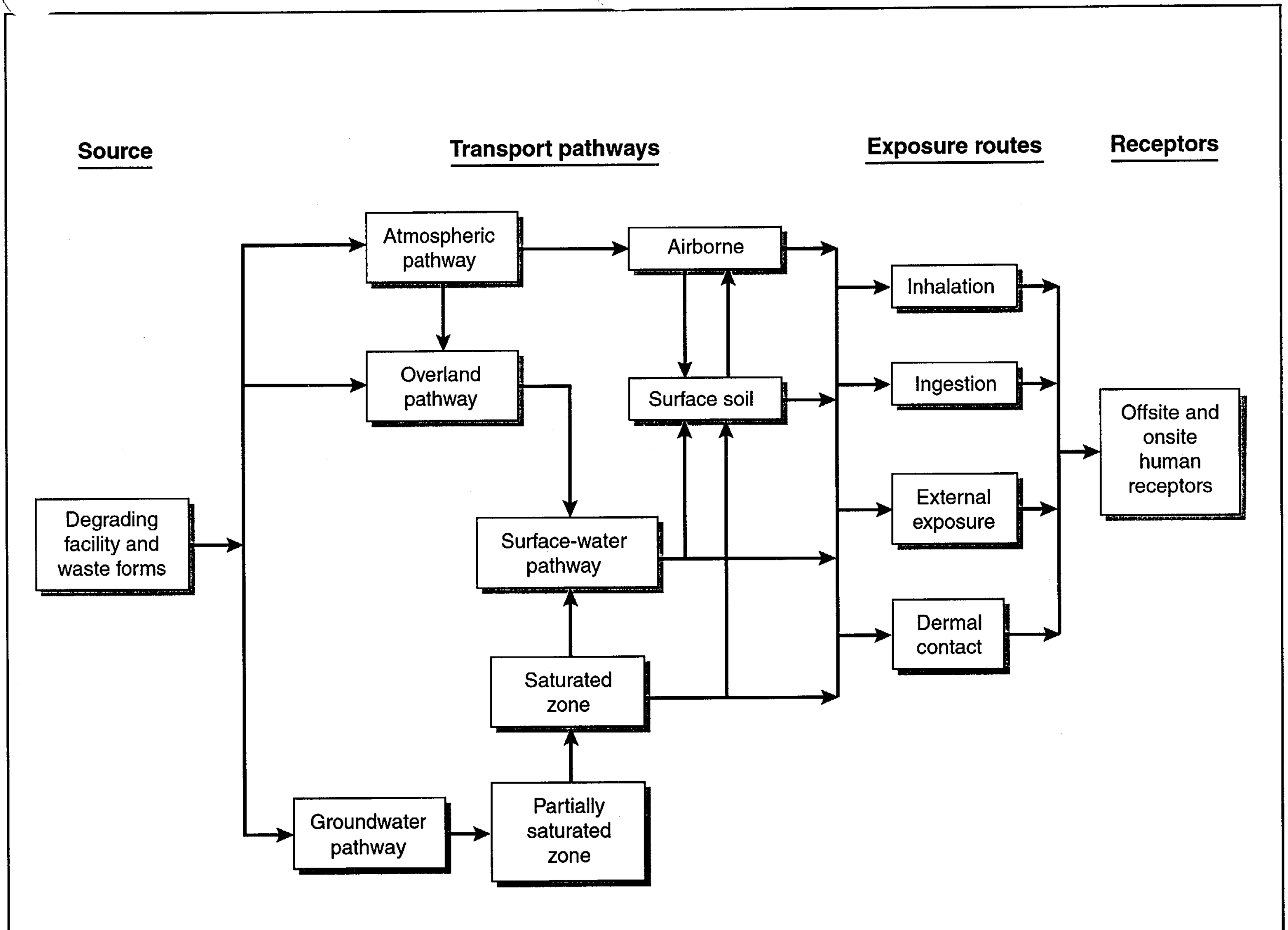
K.2.3 ENVIRONMENTAL TRANSPORT OF RADIOACTIVE MATERIALS

Radioactive materials in degraded spent nuclear fuel and high-level radioactive waste could be transported to the environment surrounding each storage facility by three pathways: groundwater, surface-water runoff, and atmosphere. Figure K-6 shows the potential exposure pathways. The analysis assumed that existing local climates would persist throughout the time of exposure of the spent nuclear fuel and high-level radioactive waste to the environment. The assumed configuration for the degraded storage facilities would have debris covering the radioactive material, which would remain inside the dry storage canisters. While the dry storage canisters could fail sufficiently to permit water to enter, they probably would retain their structural characteristics, thereby minimizing the dispersion of radioactive particulate material to the atmosphere (Mishima 1998, page 4). Based on this analysis, the airborne particulate pathway generally would not be an important source of human exposure. The assumption is that after radionuclides dissolved in the precipitation they would reach the environment either through groundwater or surface-water transport.

The analysis performed environmental fate and transport pathway modeling using the Multimedia Environmental Pollutant Assessment System program (Buck et al. 1995, all). The Multimedia Environmental Pollutant Assessment System is an integrated system of analytical, semianalytical, and empirically based mathematical models that simulate the transport and fate of radioactive materials through various environmental media and calculate concentrations, doses, and health effects at designated receptor locations.

The Multimedia Environmental Pollutant Assessment System was originally developed by Pacific Northwest National Laboratory to enable DOE to prioritize the investigation and remediation of the Department's hazardous, radioactive, and mixed waste sites in a scientific and objective manner based on readily available site information. The Multimedia Environmental Pollutant Assessment System has evolved into a widely accepted (by Federal and international agencies) computational tool for calculating the magnitude of environmental concentrations and public health impacts caused by releases of radioactive material from various sources.

The following sections discuss the assumptions and methods used to determine radioactive material transport for groundwater and surface-water pathways. Environmental parameters defined for input to the



Source: Adapted from Buck et al. (1995, all).

Figure K-6. Potential exposure pathways associated with degradation of spent nuclear fuel and high-level radioactive waste.

Multimedia Environmental Pollutant Assessment System program were collected from various sources for specific sites (Sinkowski 1998, page 2) and regionalized parameters were developed (Poe and Wise 1998, all). The analysis used long-term averages to represent environmental conditions, and assumed that these parameters would remain constant over the 10,000-year analysis period. The following sections discuss the method for each pathway.

K.2.3.1 Groundwater Transport

Precipitation falling on degrading spent nuclear fuel and high-level radioactive waste material would form a radioactive solution (leachate) that could migrate through the vadose zone (the unsaturated upper layer of soil) to the underlying water table, which would dilute, disperse, and transport the material downgradient through the local aquifer system. As a result, there is a potential for human exposure through the groundwater pathway to downgradient well users and to populations along surface-water bodies where groundwater feeds into surface water.

The groundwater component of the radioactive material fluxes (infiltration) averaged over 70-year (lifetime) increments was entered in the Multimedia Environmental Pollutant Assessment System program. The infiltration would carry the contaminated leachate down through the vadose zone to the saturated zone (aquifer). The contaminants would be diluted and dispersed as they traveled through the aquifer. Radioactive material retardation would occur in both the unsaturated (above the water table) and saturated (below the water table) zones. A distribution adsorption (that is, surface retention) coefficient, K_d, (the amount of material adsorbed to soil particles relative to that in the water) modeled this retardation (Toblin 1998a, page 2). This coefficient is radioactive material-specific and varies for each material based on such factors as soil pH and clay content.

Table K-5 lists the adsorption coefficients, K_d, for the elements explicitly modeled for groundwater transport. The coefficients are expressed as a function of the clay content of the soil through which the elements are being transported; the analyses assumed a soil pH between 5 and 9. Note that the K_d values of all isotopes of a given element (for example, plutonium-238, -239, and -240) are the same, because adsorption is a chemical rather than nuclear process.

The time required to traverse the groundwater was determined for each radionuclide and 70-year period (Toblin 1998a, page 4). Tables K-6 and K-7 list the range of nuclide groundwater transport times, from source to receptor, for each of the five regions. Times are listed for the important nuclides (see Table K-4). The analysis assumed that the vadose/aquifer flow fields were steady-state, so that the nuclide travel times at a particular site would be constant over the 10,000-year analysis period, although the nuclide release rates were not. Table K-6 lists parameters describing the total (over the analysis period) and maximum nuclide release rates for the same important nuclides. Region 5, dominated by two large DOE sites, is seen to result in the largest nuclide releases of all of the regions.

Table K-7 also lists the number of water systems and people that would obtain water from the affected waterways. Many of these people would be subject to impacts from more than one site because they would obtain their water from affected waterways downstream from multiple sites.

When the groundwater reached the point where it outcropped to surface water, radioactive material transport would be subject to further dilution and dispersion. For most of the regions analyzed, the distance between the storage location and the downgradient surface-water body would be inside the site boundary; therefore, offsite wells generally would not be affected. However, the analysis calculated groundwater concentrations for hypothetical onsite and offsite receptors. The Multimedia Environmental Pollutant Assessment System program calculated groundwater and surface-water concentrations at each receptor location for consecutive 70-year lifetimes in the 10,000-year analysis period.

Table K-5. Multimedia Environmental Pollutant Assessment System default elemental equilibrium adsorption coefficients (K_d) for soil pH between 5 and 9.[a]

Element	Clay content by weight < 10 percent	10 to 30 percent	≥ 30 percent
Actinium	228	538	4,600
Americium	82	200	1,000
Californium	0	0	0
Carbon	0	0	0
Cesium	51	249	270
Chlorine	0	0	0
Cobalt	2	9	200
Curium	82	200	1,000
Iodine	0	0	0
Krypton	0	0	0
Lead	234	597	1,830
Neptunium	3	3	3
Nickel	12	59	650
Niobium	50	100	100
Palladium	0	4	40
Plutonium	10	100	250
Protactinium	0	50	500
Radium	24	100	124
Ruthenium	274	351	690
Samarium	228	538	4,600
Selenium	6	15	15
Strontium	24	100	124
Technetium	3	20	20
Thorium	100	500	2,700
Tin	5	10	10
Tritium	0	0	0
Uranium	0	50	500
Zirconium	50	500	1,000

a. Source: Toblin (1998a, page 2).

The parameters necessary for the spent nuclear fuel and high-level radioactive waste storage sites for the Multimedia Environmental Pollutant Assessment System were defined. Pertinent hydrologic and hydrogeologic information was derived from the site-specific Updated Final Safety Analysis Reports for commercial nuclear sites and site-specific data provided by the various DOE sites (Jenkins 1998, page 1).

Table K-8 lists the range (over the individual sites) in each region of the important hydrogeologic parameters that would affect the transport of the radionuclides through the groundwater. These parameters form the basis for the nuclide transport times listed in Table K-7.

A simplifying analytical assumption was that radioactive material transport would occur only through the shallowest aquifer beneath the site. Because this assumption limits the interchange of groundwater with underlying aquifers, less radioactive material dilution would occur, and groundwater pathway impacts could be slightly overestimated. However, because impacts from the groundwater pathway would be minor in comparison to surface-water pathways, the total estimated impacts would not be affected by this assumption.

Table K-6. Regional source terms and environmental transport data for important isotopes used for collective drinking water radiological impact analysis.[a]

Parameter	Plutonium-239/240	Plutonium-238	Americium-241	Americium-243	Neptunium-237	Technetium-99
Nuclide released in 10,000 years (curies)						
Region 1	4,200	20	660	115	8.9	98
Region 2	17,000	97	1,500	240	32	1,200
Region 3	130,000	660	31,000	3,300	260	2,600
Region 4	4,300	17	450	110	9.0	89
Region 5	570,000	180	42,000	1,700	720	6,500
Maximum annual nuclide release (curies per year)						
Region 1	19	0.020	1.2	0.053	0.0031	0.034
Region 2	53	0.035	2.2	0.11	0.0083	0.19
Region 3	60	0.71	56	1.6	0.092	1.0
Region 4	0.20	0.016	0.78	0.054	0.0034	0.035
Region 5	140	0.22	66	0.47	0.14	1.4
Years (from 2016) of maximum annual nuclide release						
Region 1	1,435	1,435	1,435	1,435	1,435	1,435
Region 2	1,575	1,575	1,575	1,575	1,575	1,575
Region 3	1,155	1,155	1,155	1,155	1,155	1,155
Region 4	1,715	1,715	1,715	1,715	1,715	1,715
Region 5	875	875	875	875	875	875
Nuclide reaching receptors in 10,000 year (curies)						
Region 1	3,600	11	130	43	8.8	95
Region 2	13,000	10	1.4	39	31	1,100
Region 3	110,000	250	380	510	250	2,500
Region 4	2,000	3.6	0.66	24	6.0	59
Region 5	180,000	2.6	0.020	1.2	630	5,600
Nuclide transport time[b] (years)						
Region 1	10-5,500	10-5,500	10-45,000	10-45,000	10-1,700	10-1,700
Region 2	460-9,000	460-9,000	2,000-36,000	2,000-36,000	43-860	140-1,500
Region 3	65-45,000	65-45,000	410-260,000	410-260,000	31-9,800	31-9,800
Region 4	850-520,000	850-520,000	3,000-1,000,000	3,000-1,000,000	59-16,000	130-100,000
Region 5	1,400-26,000	1,400-26,000	2,700-220,000	2,700-220,000	44-8,000	280-8,000

a. Source: Toblin (1998a, page 4).
b. Time from source to receptor.

Table K-7. Transport and population data for drinking water pathway impact analysis.

Parameter	Region 1	Region 2	Region 3	Region 4	Region 5
Groundwater flow time (years)[a]	2.0 to 59	4.6 to 37	1.8 to 420	4.6 to 960	2.9 to 190
Number of people that would obtain domestic water supply from affected waterways (millions)[b]	6.7	5.3	13.1	5.3	0.16
Affected drinking water systems[c]	112	147	137	64	23

a. From source to outcrop; source: adapted from Jenkins (1998, Table 2).
b. Source: Poe (1998b, page 12).
c. Source: Adapted from Sinkowski (1998, all).

K.2.3.2 Surface-Water Transport

The amount of leachate from degraded spent nuclear fuel and high-level radioactive waste in the surface-water pathway would depend on soil characteristics and the local climate. The Multimedia Environmental Pollutant Assessment System considers precipitation rates (Table K-2), soil infiltration, evapotranspiration, and erosion management practices to determine the amount of leachate that would run

Table K-8. Multimedia Environmental Pollutant Assessment System regional groundwater input parameters.[a]

Parameter	Region 1	Region 2	Region 3	Region 4	Region 5
Vadose zone					
Contaminated liquid infiltration rate (vertical Darcy velocity) (feet per year)[b]	3.1 - 3.5	4.4	2.7 - 3.1	2.7 - 4.4	0.88 - 3.1
Clay content (percent)	0 - 15	1 - 47	1 - 47	3 - 15	1 - 15
pH of pore water	5 - 9	5 - 9	5 - 9	5 - 9	5 - 9
Thickness (feet)	6 - 50	10 - 50	4 - 160	2 - 80	23 - 250
Bulk density (grams per cubic centimeter)	1.4 - 1.9	1.4 - 1.6	1.4 - 1.6	1.4 - 1.6	1.4 - 1.7
Total porosity (percent)	5 - 46	38 - 49	38 - 49	38 - 46	38 - 49
Field capacity (percent)	2.5 - 28	9 - 42	9 - 42	9 - 28	9 - 28
Saturated hydraulic conductivity (feet per year)	210 - 6,800	27 - 6,800	27 - 6,800	210 - 6,800	72 - 6,800
Aquifer					
Clay content (percent)	0 - 3	0 - 47	0 - 15	0 - 15	0 - 10
pH of pore water	5 - 9	5 - 9	5 - 9	5 - 9	5 - 9
Thickness (feet)	7 - 100	10 - 85	7 - 160	20 - 150	25 - 250
Bulk density (grams per cubic centimeter)	1.6 - 2.1	1.4 - 2.0	1.5 - 1.7	1.4 - 1.7	1.5 - 1.9
Total porosity (percent)	5 - 38	5 - 49	5 - 44	5 - 46	23 - 44
Effective porosity (percent)	2.9 - 22	2.9 - 28	2.9 - 25	22 - 27	13 - 25
Saturated hydraulic conductivity (feet per year)	210 - 6,800	27 - 6,800	27 - 6,800	210 - 6,800	72 - 6,800
Darcy velocity (feet per year)	6.8 - 1,400	12 - 170	3.9 - 430	0.58 - 270	33 - 560
Travel distance (feet)	1,900 - 5,600	2,000 - 4,700	1,900 - 23,000	1,600 - 12,000	1,900 - 37,000

a. Source: Adapted from Jenkins (1998, Table 2).
b. Annual precipitation rate (through degraded structure).

off rather than percolate into the soil. The contaminated runoff would travel overland and eventually enter nearby rivers and streams that would dilute it further.

To determine the impacts of the contaminated discharge to surface water on the downstream populations using that water (affected populations), DOE calculated the surface water flow rate and the release rate of contaminants (as curies per year) contributed by each storage location draining to the surface water. Using these values, DOE determined surface-water radionuclide concentrations for each receptor location. DOE applied these concentrations to the respective affected populations to estimate impacts for each region.

K.2.3.3 Atmospheric Transport

If degraded spent nuclear fuel or high-level radioactive waste was exposed to the environment, small particles could become suspended in the air and transported by wind. The Multimedia Environmental Pollutant Assessment System methodology includes formulations for radioactive material (particulate) suspension by wind, vehicular traffic, and other physical disturbances of the ground surface. The impacts from the atmospheric pathways would be small in comparison to surface-water pathways because the cover provided by the degraded structures and the relatively large particle size and density of the materials (see Section K.2.3) would preclude suspension by wind. Therefore, impacts from the transport of radioactive particulate materials were not included in the analysis.

K.2.4 HUMAN EXPOSURE, DOSE, AND RISK CALCULATIONS

This section describes methods used in the No-Action Scenario 2 analysis to estimate dose rates and potential impacts (latent cancer fatalities) to individuals and population groups from exposures to radionuclide contaminants in groundwater and surface water and in the atmosphere. As discussed above, these contaminated environmental media would result from the degradation of storage facilities (Sections K.2.1.1), corroding dry storage canisters (Section K.2.1.2), cladding failure (Section K.2.1.4), spent nuclear fuel and high-level radioactive waste dissolution (Section K.2.1.5), leachate percolation and groundwater transport (Section K.2.3.1), surface-water runoff (Section K.2.3.2), and atmospheric suspension and transport (Section K.2.3.3).

For Scenario 1 and the first 100 years of Scenario 2, the presence of effective institutional control would ensure that radiological releases to the environment and radiation doses to workers and the public remained within Federal limits and DOE Order requirements and were maintained as low as reasonably achievable. As a result, impacts to members of the public would be very small. Potential radiological human health impacts that could occur would be due primarily to occupational radiation exposure of onsite workers. The analysts estimated these impacts based on actual operational data from commercial nuclear powerplant sites (NRC 1991a, pages 22 - 25) and projected these impacts for the 100- and 10,000-year analysis periods for Scenario 1.

For Scenario 2, impacts to onsite workers and the public during institutional control (approximately 100 years) would be the same as those for Scenario 1. However, because the assumption for Scenario 2 is that there would be no effective institutional control after approximately 100 years, engineered barriers would begin to degrade and eventually would not prevent radioactive materials from the spent nuclear fuel and high-level radioactive waste from entering the environment. During the period of no effective institutional control, there would be no workers at the site. Thus, impacts were calculated only for the public.

For Scenario 2, the potential highest exposures and dose rates over a 70-year lifetime period were evaluated for individuals and exposed populations. In addition, the total integrated dose to the exposed population for the 10,000-year analysis period was estimated. Human exposure parameters (exposure times, ingestion and inhalation rates, agricultural activities, food consumption rates, etc.) were developed based on recommendations from Federal agencies (EPA 1988, pages 113 to 131; EPA 1991, Attachment B; NRC 1977, pages 1.109-1 to 1.109-2; Shipers and Harlan 1989, all; NRC 1991b, Chapter 6) and are reflected as Multimedia Environmental Pollutant Assessment System default values (Buck et al. 1995, Section 1.0). Other parameters chosen for this analysis are summarized in supporting documentation (Sinkowski 1998, all; Toblin 1998a,b,c, all). Table K-9 lists the exposure and usage parameters for all of the pathways considered in the analysis (see Section K.3.1).

The Scenario 2 analysis evaluated long-term radiation doses and impacts to populations exposed through the surface-water and groundwater pathways. This analysis estimated population impacts only for the drinking water pathway using regionalized effective populations and surface-water dilution factors discussed in Section K.2.3.2. Other pathways were evaluated to determine their potential contribution in relation to drinking water doses. These analyses are discussed in Section K.3.1.

K.2.4.1 Gardener Impacts

To reasonably bound human health impacts resulting from human intrusion, two types of gardener were evaluated—the onsite gardener (10 meters [33 feet]) from the degrading storage facility) and the near-site gardener (5 kilometers [3 miles] from the degrading facility). The analysis had both of these hypothetical gardeners residing on the flow path for groundwater. The gardeners would obtain all their drinking water from contaminated groundwater, grow their subsistence gardens in contaminated soils, and irrigate them with the contaminated groundwater. The contaminated garden soils, suspended by the wind, would contaminate the surfaces of the vegetables consumed by the gardeners. The hypothetical onsite gardener would be the maximally exposed individual.

Table K-9. Multimedia Environmental Pollutant Assessment System human exposure input parameters for determination of all pathways radiological impacts sensitivity analysis (page 1 of 2).[a]

Water source[b]	Surface water
Domestic water supply treatment[c]	Yes
Fraction of plutonium removed by water treatment[d]	0.3
Drinking water rate (liters per day per person)[e]	2
Irrigation rate (liters per square meter per month)[f]	100
Leafy vegetable consumption rate (kilograms per day per person)[g]	0.021
Other vegetable consumption rate (kilograms per day per person)	0.13
Meat consumption rate (kilograms per day per person)	0.065
Milk consumption rate (kilograms per day per person)	0.075
Finfish consumption rate (kilograms per day per person)	0.0065
Shellfish consumption rate (kilograms per day per person)	0.0027
Shoreline contact (hours per day per person)	0.033
Americium ingestion dose conversion factor (rem per picocurie)[h]	3.6×10^{-6}
Americium finfish bioaccumulation factor	250
Americium shellfish bioaccumulation factor	1,000
Americium meat transfer factor (days per kilogram)	3.5×10^{-6}
Americium milk transfer factor (days per liter)	4.0×10^{-7}
Neptunium ingestion dose conversion factor (rem per picocurie)	4.4×10^{-6}
Neptunium finfish bioaccumulation factor	250
Neptunium shellfish bioaccumulation factor	400
Neptunium meat transfer factor (days per kilogram)	5.5×10^{-5}
Neptunium milk transfer factor (days per liter)	5.0×10^{-6}
Technetium ingestion dose conversion factor (rem per picocurie)	1.5×10^{-9}
Technetium finfish bioaccumulation factor	15
Technetium shellfish bioaccumulation factor	5
Technetium meat transfer factor (days per kilogram)	8.5×10^{-3}
Technetium milk transfer factor (days per liter)	1.2×10^{-2}
Plutonium ingestion dose conversion factor (rem per picocurie)[i]	3.5×10^{-6}
Plutonium finfish bioaccumulation factor	250
Plutonium shellfish bioaccumulation factor	100
Plutonium meat transfer factor (days per kilogram)	5.0×10^{-7}
Plutonium milk transfer factor (days per liter)	1×10^{-7}
Yield of leafy vegetables [kilograms (wet) per square meter]	2.0
Yield of vegetables [kilograms (wet) per square meter]	2.0
Yield of meat feed crops [kilograms (wet) per square meter]	0.7
Yield of milk animal feed crops [kilograms (wet) per square meter]	0.7
Meat animal intake rate for feed (liters per day)	68
Milk animal intake rate for feed (liters per day)	55
Meat animal intake rate for water (liters per day)	50
Milk animal intake rate for water (liters per day)	60
Agricultural areal soil density (kilograms per square meter)	240
Retention fraction of activity on plants	0.25
Translocation factor for leafy vegetables	1.0
Translocation factor for other vegetables	0.1
Translocation factor for meat animal	0.1
Translocation factor for milk animal	1.0
Fraction of meat feed contaminated	1.0
Fraction of milk feed contaminated	1.0
Fraction of meat water contaminated	1.0
Fraction of milk water contaminated	1.0
Meat animal soil intake rate (kilograms per day)	0.5

Table K-9. Multimedia Environmental Pollutant Assessment System human exposure input parameters for determination of all pathways radiological impacts sensitivity analysis (page 2 of 2).[a]

Water source[b]	Surface water
Milk animal soil intake rate (kilograms per day)	0.5
Leafy vegetable growing period (days)	60
Other vegetable growing period (days)	60
Beef animal feed growing period (days)	30
Milk animal feed growing period (days)	30
Water intake rate while showering (liters per hour)	0.06
Duration of shower exposure (hours per shower)	0.167
Shower frequency (per day)	1.0
Thickness of shoreline sediment (meters)	0.04
Density of shoreline sediments (grams per cubic meter)	1.5
Shore width factor for shoreline external exposure	0.2

a. Source: Buck et al. (1995, MEPAS default settings).
b. Groundwater for gardener.
c. No for gardener.
d. Zero for gardener.
e. To convert liters to gallons, multiply by 0.26418.
f. To convert liters per square meter to gallons per square foot, multiply by 0.00025.
g. To covert kilograms to pounds, multiply by 2.2046.
h. Sediment ingestion = 0.1 grams per hour (0.000022 pounds per hour) during contact.
i. For plutonium-239/240.

HUMAN INTRUSION

Spent nuclear fuel and high-level radioactive waste in surface or below-grade storage facilities would be readily accessible in the absence of institutional control. For this reason, DOE anticipates that both planned and inadvertent intrusions could occur. An example of the former would be the scavenger who searches through the area seeking articles of value; an example of the latter would be the farmer who settles on the site and grows agricultural crops with no knowledge of the storage structure beneath the soil. Intrusions into contaminated areas also could occur through activities such as building excavations, road construction, and pipeline or utility replacement.

Under the conditions of Scenario 2, intruders could receive external exposures from stored spent nuclear fuel and high-level radioactive waste that would grossly exceed current regulatory limits and, in some cases, could be sufficiently high to cause prompt fatalities. In addition, long-term and repeated intrusions, such as those caused by residential construction or agricultural activities near storage sites, could result in long-term chronic exposures that could produce increased numbers of latent cancer fatalities. These intrusions could also result in the spread of contamination to remote locations, which could increase the total number of individuals potentially exposed.

Calculations were performed using transport models described by Buck et al. (1995, all) for gardeners in each of the five analysis regions using regionalized source terms and environmental parameters. Therefore, calculated impacts to the regional gardener (maximally exposed individual) would not represent the highest impacts possible from a single site in a given region, but rather would reflect an average impact for the region. Details of the analysis are provided in Toblin (1998c, all). The regional hydrogeologic parameters listed in Table K-10, together with transient nuclide release rates (the maximum of which is indicated in the table), were used to determine the radiological impacts to the regional gardener as a result of groundwater transport. The regional parameters were based on a curie-weighting of the individual site parameters for plutonium and americium. The exposure parameters in

Table K-10. Multimedia Environmental Pollutant Assessment System groundwater transport input parameters for estimating radiological impacts to the onsite and near-site gardener.[a]

Parameter	Region 1	Region 2	Region 3	Region 4	Region 5
Vadose zone					
Contaminated liquid infiltration rate (vertical Darcy velocity) (feet per year)[b,c]	3.5	4.4	2.7	3.5	0.88
Clay content (percent)	1	10	12	11	2
pH of pore water	5 - 9	5 - 9	5 - 9	5 - 9	5-9
Thickness (feet)	11	44	7.1	43	180
Longitudinal dispersivity (feet)	0.11	0.44	0.071	0.43	1.8
Bulk density (grams per cubic meter)[d]	1.6	1.5	1.5	1.5	1.6
Total porosity (percent)	38	42	44	45	41
Field capacity (percent)	9.3	15	23	21	12
Saturated hydraulic conductivity (feet per year)	6,500	660	1,700	1,000	5,900
Aquifer					
Clay content (percent)	1.8	6.5	1.2	4.4	0.69
pH of pore water	5 - 9	5 - 9	5 - 9	5 - 9	5 - 9
Thickness (feet)	45	50	37	64	210
Bulk density (grams per cubic meter)	1.6	1.8	1.6	1.6	1.7
Total porosity (percent)	38	40	38	35	30
Effective porosity (percent)	22	23	22	20	17
Darcy velocity (feet per year)	340	62	69	51	300
Longitudinal dispersivity (feet)	f(x)[e]	f(x)	f(x)	f(x)	f(x)
Lateral dispersivity (feet)	f(x) ÷ 3	f(x) ÷ 3	f(x) ÷ 3	f(x) ÷ 3	f(x) ÷ 3
Vertical dispersivity (feet)	f(x) ÷ 400	f(x) ÷ 400	f(x) ÷ 400	f(x) ÷ 400	f(x) ÷ 400
Maximum annual plutonium-239 and -240 release (curies per year)	4.9	0.24	3.8	0.32	2.1
Years (from 2016) of maximum annual plutonium release	1,365	1,575	1,155	1,715	875

a. Source: Toblin (1998c, page 2-4).
b. Annual precipitation rate (through degraded structure).
c. To convert feet to meters, multiply by 0.3048.
d. To convert grams per cubic meter to pounds per cubic foot, multiply by 0.0000624.
e. $f(x) = 2.72 \times (\log_{10} 0.3048 \times x)^{2.414}$, where x = downgradient distance.

Table K-9 describe the radionuclide exposure to the gardener where applicable (for example, exposure parameters related to the fish are not applicable to the gardener).

K.2.4.2 Direct Exposure

The analysis evaluated potential external radiation dose rates to the maximally exposed individual for a commercial independent spent fuel storage installation because this type of facility would provide the highest external exposures of all the facilities analyzed in this appendix. Maximum dose rates over the 10,000-year analysis period were evaluated for each region. The maximally exposed individual was assumed to be 10 meters (about 33 feet) from an array of concrete storage modules containing 1,000 MTHM of commercial spent nuclear fuel. The maximum dose rate varied between regions depending on how long the concrete shielding would remain intact (Table K-1).

The direct gamma radiation levels were calculated (Davis 1998, page 1). To ensure consistency between this analysis and the Total System Performance Assessment, the same radionuclides were used for the design of the Yucca Mountain Repository surface facility shielding (TRW 1995, Attachment 9.5). Radionuclide decay and radioactive decay product ingrowth over the 10,000-year analysis period were calculated using the ORIGEN computer program (ORNL 1991, all).

Neutron emissions were not included because worst-case impacts (death within a short period of exposure) would be the same with or without the neutron component. Details of these calculations and analyses are provided in supporting documentation (Rollins 1998b, all).

K.2.5 ACCIDENT METHODOLOGY

Spent nuclear fuel and high-level radioactive waste stored in above-ground dry storage facilities would be protected initially by the robust surrounding structure (either metal or concrete) and by a steel storage container that contained the material. Normal storage facility operations would be primarily passive because the facilities would be designed for cooling via natural convection. DOE evaluated potential accident and criticality impacts for both Scenario 1 (institutional control for 10,000 years) and Scenario 2 (assumption of no effective institutional control after approximately 100 years with deterioration of the engineered barriers initially protecting the spent nuclear fuel or high-level radioactive waste).

For Scenario 1, human activities at each facility would include surveillance, inspection, maintenance, and equipment replacement when required. The facilities and the associated systems, which would be licensed by the Nuclear Regulatory Commission, would have certain required features. License requirements would include isolation of the stored material from the environment and its protection from severe accident conditions (10 CFR 50.34). The Nuclear Regulatory Commission requires an extensive safety analysis that considers the impacts of plausible accident-initiating events such as earthquakes, fires, high winds, and tornadoes. No plausible accident scenarios have been identified that result in the release of radioactive material from the storage facilities (PGE 1996, all; CP&L 1989, all). In addition, the license would specify that facility design requirements include features to provide protection from the impacts of severe natural events. This analysis assumed maintenance of these features indefinitely for the storage facilities.

DOE performed a scoping analysis to identify the kinds of events that could lead to releases of radioactive material to the environment prior to degradation of concrete storage modules and found none. The two events determined to be the most challenging to the integrity of the concrete storage modules would be the crash of an aircraft into the storage facility and a severe seismic event.

- Davis, Strenge, and Mishima (1998, all) concluded that the postulated aircraft crash would be potentially more severe than a postulated seismic event because storage facility damage from an aircraft crash probably would be accompanied by a fire that could heat the spent nuclear fuel or high-level radioactive waste and increase the quantity of material released to the environment. The analysis showed that hurtling aircraft components produced by such an event would not penetrate the storage facility and that a subsequent fire would not result in a release of radioactive materials.

- For the seismic event, meaningful damage would be unlikely because storage facilities would be designed to withstand severe earthquakes. Even if such an event caused damage, no immediate release would occur because no mechanism has been identified that would cause meaningful fuel pellet damage to create respirable airborne particles. If this damage did not occur, the source term would be limited to gaseous fission products, carbon-14, and a very small amount of preexisting fuel pellet dust. Subsequent repairs to damaged facilities or concrete storage modules would preclude the long-term release of radionuclides.

Criticality events are not plausible for Scenario 1 because water, which is required for criticality, could not enter the dry storage canister. The water would have to penetrate several independent barriers, all of which would be maintained and replaced as necessary under Scenario 1.

Under Scenario 2, facilities would degrade over time and the structures would gradually deteriorate and lose their integrity. The analysis determined that two events, an aircraft crash and inadvertent criticality, would be likely to dominate the impacts from accidents, as described in the following paragraphs.

K.2.5.1 Aircraft Crash

DOE determined that an aircraft crash into a degraded concrete storage module would be the largest plausible accident-initiating event that could occur at the storage sites. This event would provide the potential for the airborne dispersion of radioactive material to the environment and, as a result, the potential for exposure of individuals who lived in the vicinity of the site. The aircraft crash could result in mechanical damage to the storage casks and the fuel assemblies they contained, and a fire could result. The fire would provide an additional mechanism for dispersion of the radioactive material. The frequency and consequences of this event are described in detail in Davis, Strenge and Mishima (1998, all).

The aircraft assumed for the analysis is a midsize twin-engine commercial jet (Davis, Strenge, and Mishima 1998, page 2). The area affected by a crash was computed using the DOE standard formula (DOE 1996, Chapter 6) in which the aircraft could crash directly into the side or top of the concrete storage modules, or could strike the ground in the immediate vicinity of the facility and skid into the concrete storage modules. Using this formula, the dimensions of a typical storage facility as shown in Chapter 2, Figure 2-37, and the aircraft configuration would result in an estimated aircraft crash frequency of 0.0000032 (3 in 1 million) crashes per year (Davis, Strenge, and Mishima 1998, page 5). This frequency is within the range that DOE typically considers the design basis, which is defined by DOE as 0.000001 or greater per year (DOE 1993, page 28).

The analysis estimated the consequences of the aircraft crash on degraded concrete storage modules. The twin-engine jet was assumed to crash into an independent spent fuel storage installation that contained 100 concrete storage modules, each containing 24 pressurized-water reactor fuel assemblies. Using the penetration methodology from DOE (1996, Chapter 6), an aircraft crash onto these concrete storage modules could penetrate 0.8 meter (2.6 feet). Because the concrete storage modules have 1.2-meter (3.9-foot) thick walls, the crash projectiles would not penetrate the reinforced concrete in the as-constructed form. Thus, DOE determined that the aircraft crash would not cause meaningful consequences until the concrete storage modules were considerably degraded, when an aircraft projectile could penetrate a concrete storage module and damage a storage cask (Davis, Strenge, and Mishima 1998, page 7). The degradation process is highly location-dependent, as noted in Section K.2.1.1. For sites in northern climates, the degradation would be relatively rapid due to the freeze/thaw cycling that would expedite concrete breakup; considerable degradation could occur in 200 to 300 years. For southern climates, the degradation would be much slower. Thus, an aircraft crash probably would not result in meaningful consequences for a few hundred to a few thousand years, depending on location. The timing is of some importance because the radioactive materials in the fuel would decay over time, and the potential for radiation exposure would decline with the decay.

The analysis assumed that the aircraft crash occurred 1,000 years after the termination of institutional control at a facility where the concrete had degraded sufficiently to allow breach of the dry storage canister. Computing public impacts from the air crash event requires estimating the population to a distance of 80 kilometers (50 miles) from a hypothetical site (the distance beyond which impacts from an airborne release would be very small). This analysis considered two such sites, one in an area of a high population site and one in an area of low population. The average population around all of the sites in each of the five regions defined in Figure K-2 was computed based on 1990 census data. The average ranged from a high of 330 persons per square mile in region 1 (high population) to a low of 77 persons per square mile in region 4 (low population). Both of these population densities (assumed to be uniform around the hypothetical sites) were used in the consequence calculation.

Estimating the amount of airborne respirable particles that would result from a crash requires assumptions about the impact and resulting fire. The impact of the jet engines probably would cause extensive damage to the fuel assemblies in the degraded concrete storage module, and would scatter fuel pins around the immediate area. The fuel tanks in the aircraft would rupture, and fuel would disperse around the site and collect in pools. These pools would ignite, and an intense fire [hotter than 500°C (approximately 930°F)] (Davis, Strenge, and Mishima 1998, page 8) would result. The fire would heat the fuel pins to the point of cladding rupture. The ruptured fuel pins would cause fuel pellets to be exposed to the fire. As the fire burned, the fuel pools would recede, exposing additional fuel pellets to the air. This would cause oxidation of the hot uranium dioxide fuel pellets, converting them to U_3O_8 (another form of uranium oxide), which would produce a large amount of fuel pellet dust, including small particles that could become airborne and inhaled into the lungs. The estimated fraction of the fuel converted to respirable airborne dust would be 0.12 percent (Davis, Strenge and Mishima 1998, page 9). The fire would cause a thermal updraft that could loft the fuel pellet dust into the atmosphere.

The consequences from the event were computed with the MACCS2 program (Rollstin, Chanin, and Jow 1990, all). This model has been used extensively by the Nuclear Regulatory Commission and DOE to estimate impacts from accident scenarios involving releases of radioactive materials. The model computes dose to the public from the direct radiation by the cloud of radioactive particles released during the accident, from inhaling particles, and from consuming food produced from crops and grazing land that could be contaminated as the particles are deposited on the ground from the passing cloud. The food production and consumption rates are based on generic U.S. values (Kennedy and Strenge 1992, pages 6.19 to 6.28; Chanin and Young 1998, all). The program computes the dispersion of the particles as the cloud moves downwind. The dispersion would depend on the weather conditions (primarily wind speed, stability, and direction) that existed at the time of the accident. This calculation assumed median weather conditions and used annual weather data from airports near the centers of the regions.

K.2.5.2 Criticality

DOE evaluated the potential for nuclear criticality accidents involving stored spent nuclear fuel. A criticality accident is not possible in high-level radioactive waste because most of the fissionable atoms were removed or the density of fissionable atoms was reduced by the addition of glass matrix. Nuclear criticality is the generation of energy by the fissioning (splitting) of atoms as a result of collisions with neutrons. The energy release rate from the criticality event can be very low or very high, depending on several factors, including the concentration of fissionable atoms, the availability of moderating materials to slow the neutrons to a speed that enables them to collide with the fissionable atoms, and the presence of materials that can absorb neutrons, thus reducing the number of fission events.

Criticality events are of concern because under some conditions they could result in an abrupt release of radioactive material to the environment. If the event were energetic enough, the dry storage canister could split open, fuel cladding failure could occur, and fragmentation of the uranium dioxide fuel pellets could occur.

The designs of existing dry storage systems for spent nuclear fuel, in accordance with Nuclear Regulatory Commission regulations (10 CFR Part 72) preclude criticality events by various measures, including primarily the prevention of water entering the dry storage canister. If water is excluded, a criticality cannot occur.

If institutional control was maintained at the dry storage facilities (Scenario 1), a criticality is not plausible because the casks would be monitored and maintained such that introduction of water into the canister would not be possible. However, under Scenario 2, eventual degradation (corrosion) of the dry storage canisters could lead to the entry of water from precipitation, at which point criticality could be possible if other conditions were met simultaneously.

The analysis considered three separate criticality events:

- A low-energy event that involved a criticality lasting over an intermediate period (minutes or more). This event would not produce high temperatures or generate large additional quantities of radionuclides. Thus, no fuel cladding failures and no meaningful increase in consequences would be likely.

- An event in which a system went critical but at a slow enough rate so the energy release would not be large enough to produce steam, which would terminate the event. This event could continue over a relatively long period (minutes to hours), and would differ from the low-energy event in that the total number of fissions could be very large, and a large increase in radionuclide inventory could result. This increase could double the fission product content of the spent nuclear fuel. No fuel cladding failures would be likely in this event, so no abrupt release of radionuclides would occur.

- An energetic event in which a system went critical and produced considerable fission energy. This event could occur if seriously degraded fuel elements collapsed abruptly to the bottom of the canister in the presence of water that had penetrated the canister. This event would produce high fuel temperatures that could lead to cladding rupture and fuel pellet oxidation. The radiotoxicity of the radionuclide inventory produced by the fission process would be comparable to the inventory in the fuel before the event.

The probability of a criticality occurring as described in these scenarios is highly uncertain. However, DOE expects the probability would be higher for the first two events, and much lower for the third (energetic energy release). Several conditions would have to be met for any of the three events to occur. The concrete storage module and dry storage canister must have degraded such that water could enter but not drain out. The fuel would have to contain sufficient fissionable atoms (uranium-235, plutonium 239) to allow criticality. This would depend on initial enrichment (initial concentration of uranium-235) and burnup of the fuel in the reactor before storage (which would reduce the uranium-235 concentration). Because a small amount of spent nuclear fuel would be likely to have appropriate enrichment burnup combinations that could enable criticality to occur, none of the criticality events can be completely ruled out. The energetic criticality event is the only one with the potential to produce large impacts. Such an event would be possible, but would be highly unlikely; its consequences would be uncertain. The event could cause a prompt release of radionuclides. However, the amount released would not be likely to exceed that released by the aircraft crash event evaluated above. Thus, this analysis did not evaluate specific consequences of a criticality event.

K.3 Results

K.3.1 RADIOLOGICAL IMPACTS

Impacts to human health from long-term environmental releases and human intrusion were estimated using the methods described in Section K.2 and in supporting technical documents (Sinkowski 1998, all; Jenkins 1998, all; Battelle 1998, all; Poe 1998a,b, all; Poe and Wise 1998, all; Toblin 1998a,b,c, all). The radiological impacts on human health would include internal exposures due to the intake of radioactive materials released to surface water and groundwater.

Six of the seven radionuclides listed in Table K-4 would contribute more than 99 percent of the total dose. Table K-11 lists the estimated radiological impacts by region during the last 9,900 years under Scenario 2 for the Proposed Action and Module 1 inventories of spent nuclear fuel and high-level radioactive waste. As noted above, these impacts would be to the public from drinking water from the major waterways contaminated by surface-water runoff of radioactive materials from degraded spent nuclear fuel and high-level radioactive waste storage facilities (Toblin 1998a,b, all). Figure K-7 shows the locations of all commercial nuclear and DOE waste storage sites in the United States and more than 20 potentially affected major waterways. At present, 30.5 million people are served by municipal water systems with intakes along the potentially affected portions of these waterways. Over the 9,900-year analysis period, about 140 generations would be potentially affected. However, because releases are not estimated to occur during about the first 1,000 years for most regions, the potential affected population could be as high as 3.9 billion.

SCENARIO 2 IMPACTS

The principal long-term human health consequences from the storage of spent nuclear fuel and high-level radioactive waste would result from rainwater flowing through degraded storage facilities where it would dissolve the material. The dissolved material would travel through groundwater and surface-water runoff to rivers and streams where people could use it for domestic purposes such as drinking water and crop irrigation. The Scenario 2 analysis estimated population impacts resulting only from the consumption of contaminated drinking water and exposures resulting from land contamination due to periodic flooding, although other pathways, such as eating contaminated fish, could contribute additional impacts larger than those from drinking water for selected individuals in the exposed population.

Table K-11. Estimated collective radiological impacts to the public from continued storage of Proposed Action and Module 1 inventories of spent nuclear fuel and high-level radioactive waste at commercial and DOE storage facilities – Scenario 2.[a]

	9,900-year population dose[b] (person-rem)		9,900-year LCFs[c]		Years until peak impact[d]	
Region	Proposed Action	Module 1	Proposed Action	Module 1	Proposed Action	Module 1
1	1,800,000	1,820,000	900	900	1,400	1,400
2	760,000	1,260,000	380	630	5,100	8,300
3	3,500,000	3,650,000	1,800	1,830	3,400[d]	3,400[d]
4	70,000	138,000	30	69	3,900	3,900
5	460,000	461,000	230	230	7,100	7,000
Totals	**6,590,000**	**7,330,000**	**3,340**	**3,700**		

a. Total population (collective) dose from drinking water pathway over 9,900 years.
b. LCF = latent cancer fatality; additional number of latent cancer fatalities for the exposed population group based on an assumed risk of 0.0005 latent cancer fatality per person-rem of collective dose (NCRP 1993a, page 112).
c. Years after 2116 when the maximum doses would occur.
d. Year of combined U.S. peak impact would be the same as for Region 3 peak impact, because the predominant impact would be in Region 3.

Table K-11 indicates the variability of individual doses and potential impacts in the five regions analyzed (see Section K.2.1.6). The variability among regions is due to differences in types and quantities of spent nuclear fuel and high-level radioactive waste, annual precipitation, size of affected populations, and surface-water bodies available to transport the radioactive material.

Table K-11 also indicates that the Proposed Action inventory would produce a collective drinking water dose of 6.6 million person-rem over 9,900 years, which could result in an additional 3,300 latent cancer

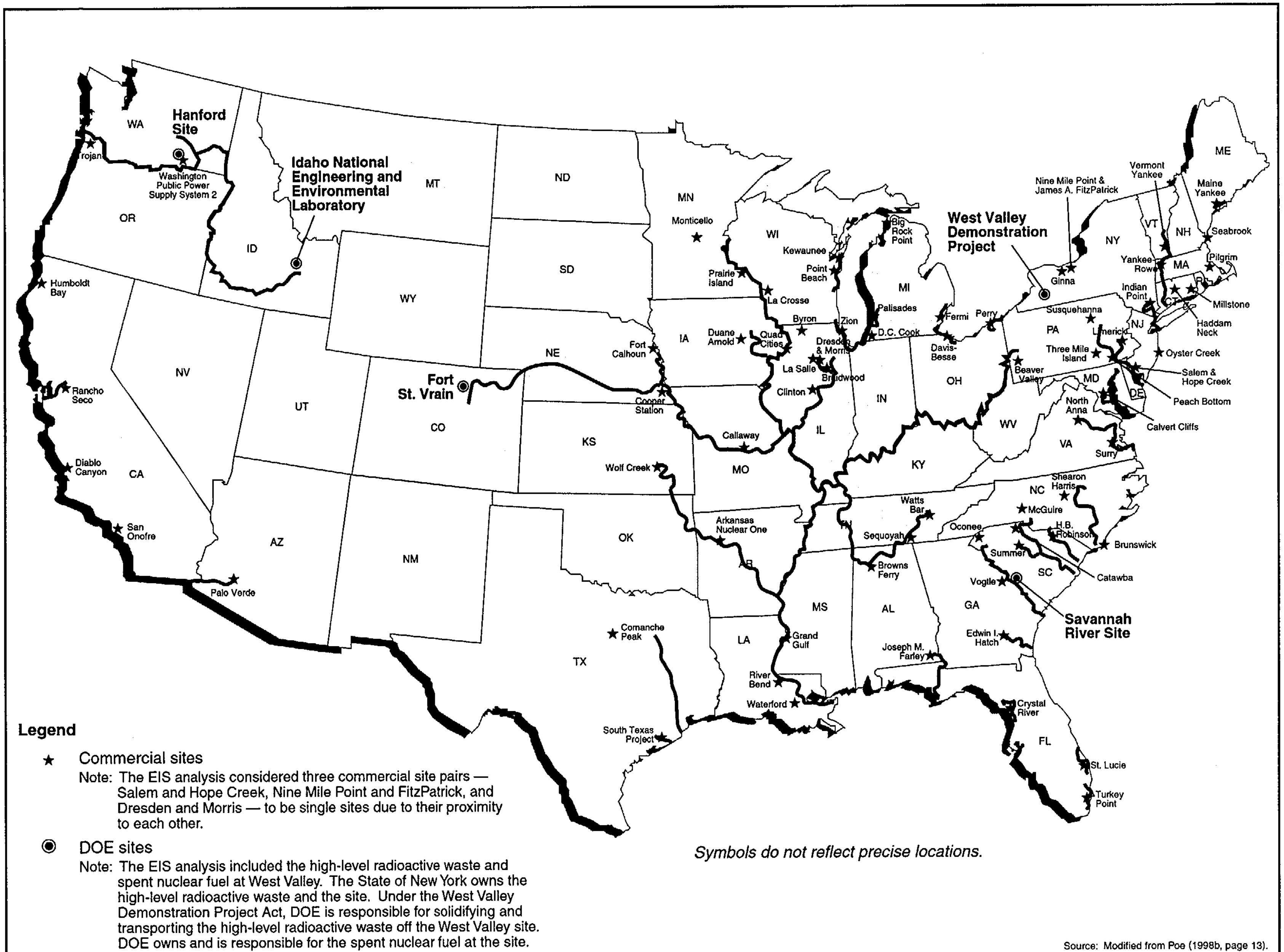


Figure K-7. Major waterways near commercial and DOE sites.

fatalities in the total potentially exposed population of 3.9 billion, in which about 900 million fatal cancers [using the lifetime fatal cancer risk of 24 percent (NCHS 1993, page 5)] would be likely to occur from all other causes. Figures K-8 and K-9 show the Proposed Action inventory regional collective doses and potential latent cancer fatalities, respectively, for approximately 140 consecutive 70-year lifetimes that would occur during the 9,900-year analysis period. The peaks shown in Figures K-8 and K-9 would result from the combination of the sites that drain to the Mississippi River and the relatively large populations potentially affected along these waterways. These values include impacts for the Proposed Action inventory only. Similar curves for the Module 1 inventory are not shown because of their similarity to those for the Proposed Action inventory. As listed in Table K-11, the impacts from the Module 1 inventory would be approximately 20 percent greater than for the Proposed Action inventory.

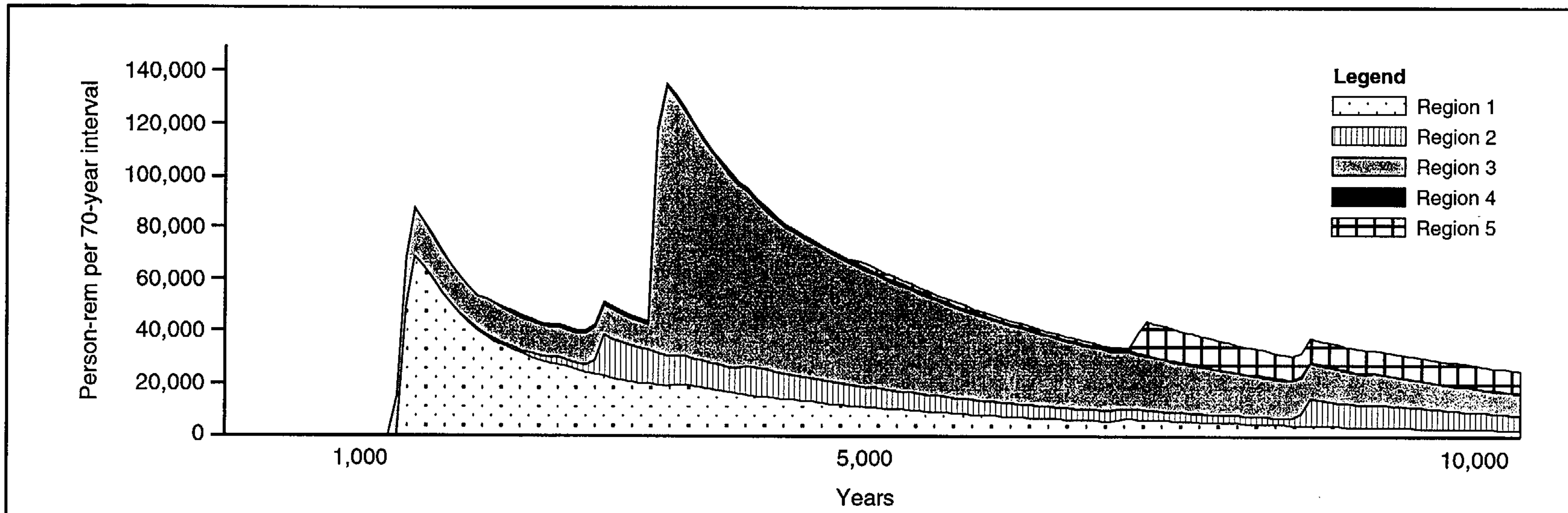
The additional 3,300 Proposed Action latent cancer fatalities (or 3,700 Module 1 latent cancer fatalities) over the 10,000-year analysis period would not be the only negative impact. Under Scenario 2, more than 20 major waterways of the United States (for example, the Great Lakes, the Mississippi, Ohio, and Columbia rivers, and many smaller rivers along the Eastern Seaboard) that currently supply domestic water to 30.5 million people would be contaminated with radioactive material. The shorelines of these waterways would be contaminated with long-lived radioactive materials (plutonium, uranium, americium, etc.) that would result in exposures to individuals who came into contact with the sediments, potentially increasing the number of latent cancer fatalities. Each of the 72 commercial and 5 DOE sites throughout the United States would have potentially hundreds of acres of land and underlying groundwater systems contaminated with radioactive materials at concentrations that would be potentially lethal to anyone who settled near the degraded storage facilities. The radioactive materials at the degraded facilities and in the floodplains and sediments would persist for hundreds of thousands of years.

As mentioned above, DOE only estimated potential collective impacts resulting from the consumption of contaminated surface water. However, other pathways (food consumption, contaminated floodplains, etc.) that could contribute to collective dose were evaluated (Toblin 1998b, all; Rollins 1998c, all) to determine their relative importance to the drinking water pathway. These pathways included the following:

- Consumption of vegetables irrigated with contaminated water
- Consumption of meat and milk from animals that drank contaminated water or were fed with contaminated feed
- Consumption of contaminated finfish and shellfish
- Direct exposure to contaminated shoreline sediments
- Exposures resulting from contamination of floodplains during periods of high stream (river) flow

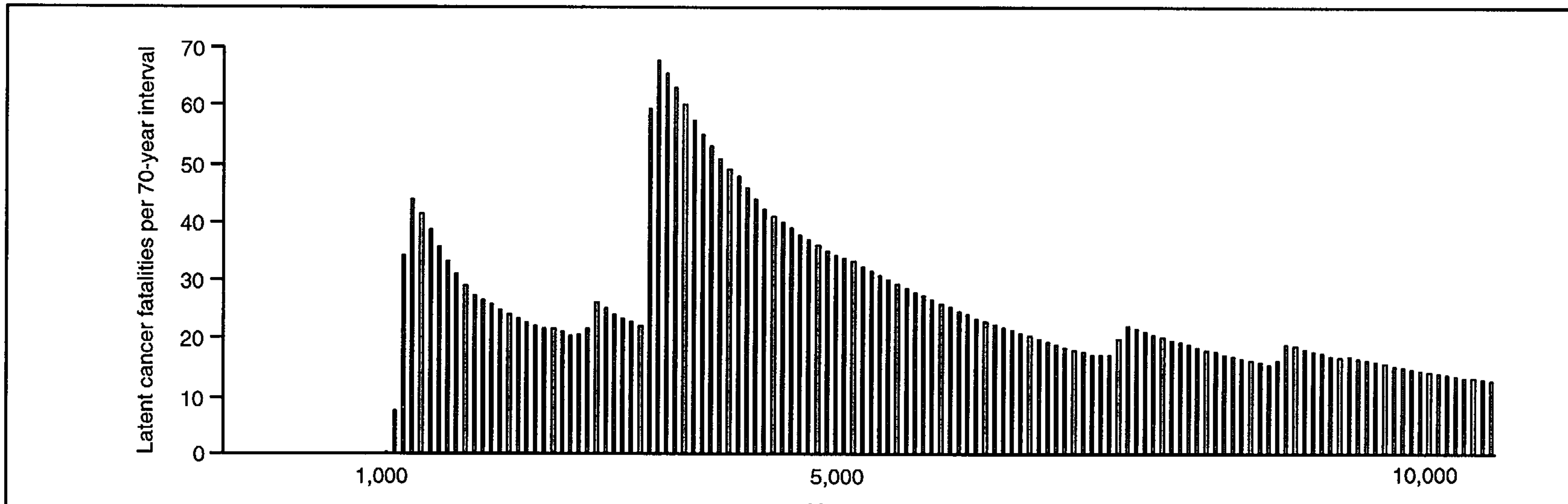
These analyses determined that an individual living in a contaminated floodplain and consuming vegetables irrigated with contaminated surface water could receive a radiation exposure dose three times higher than that from the consumption of contaminated surface water only (Toblin 1998b, page 3). In addition, the analysis determined that impacts to 30 million individuals potentially living in contaminated floodplains would be less than 10 percent of the collective impacts shown in Figure K-9 and, therefore, did not include them in the estimates because DOE did not want to overestimate the impacts from Scenario 2.

DOE evaluated airborne pathways (Mishima 1998, all) and judged that potential impacts from those pathways would be very small in comparison to impacts from liquid pathways because the degraded facility structures would protect the radioactive material from winds. To simplify the analysis, impacts to



Source: Adapted from Toblin (1998a, pages 5, 9, and 10).

Figure K-8. Regional collective dose from the Proposed Action inventory under No-Action Scenario 2.



Source: Adapted from Toblin (1998a, page 9).

Figure K-9. Total potential latent cancer fatalities throughout the United States from the Proposed Action inventory under No-Action Scenario 2.

the public from radiation emanating from the degraded storage facilities were not included. Those impacts were judged to represent a small fraction of the impacts calculated for the liquid pathways (Table K-11).

Estimates of localized impacts (Toblin 1998c, page 1) assumed that individuals (onsite and near-site gardeners) would take up residence near the degraded storage facilities and would consume vegetables from their gardens irrigated with groundwater withdrawn from the contaminated aquifer directly below their locations. In addition, the onsite gardener would be exposed to external radiation emanating from the exposed dry storage canisters; therefore, the onsite gardener would be the maximally exposed individual.

Table K-12 lists the internal estimated dose rates (see Section K.2.4.1 for details) and the times for peak exposure for each of the five regions.

Table K-12. Estimated internal dose rates (rem per year) and year of peak exposure[a] (in parentheses) for the onsite and near-site gardeners – Scenario 2.[b]

	Maximally exposed individual distances (meters)[c] from storage facilities			
Region	10[d]	150	1,000	5,000
1	3,100 (1,800)	670 (2,200)	51 (2,000)	12 (2,600)
2	100 (2,700)	96 (2,000)	12 (2,900)	2 (7,100)
3	3,100 (1,800)	1,800 (2,000)	150 (2,600)	31 (6,000)
4	140 (3,200)	130 (3,900)	14 (4,800)	2 (9,300)
5	3,300 (4,600)	180 (5,300)	59 (5,300)	2 (6,100)

a. Years after facility maintenance ended.
b. Source: Adapted from Toblin (1998c, Table 4, page 5).
c. To convert meters to feet, multiply by 3.2808.
d. The maximally exposed individual would be the onsite gardener.

The regional dose rates listed in Table K-12 would depend on the concentration of contaminants (primarily plutonium) in the underlying aquifer from which water was extracted and used by the gardener for consumption and crop irrigation. These aquifer concentrations, in turn, would be affected by the type and location of stored materials (spent nuclear fuel and high-level radioactive waste) in each region, the rate at which the contaminants were leached from the stored material, the amount of water (precipitation) available for dilution, and the thickness of the aquifer. For example, releases in Region 5 would probably be smaller and would occur later than those in other regions because of the region's lack of precipitation. This is indeed the case for commercial fuel, which is stored in above-grade concrete storage modules, stainless-steel dry storage canisters, and mostly intact corrosion-resistant zirconium alloy cladding. However, early releases would occur in Region 5 because most DOE spent nuclear fuel is stored in below-grade vaults (see Appendix A, page A-25) that would stop providing rain protection after 50 years (see Section K.2.1.1 for details). In addition, the analysis assumed no credit for the protectiveness of the DOE spent nuclear fuel cladding (see Section K.2.1.4.2 for details), which would result in releases that began early (about 800 years after weather protection was lost) and persist at a nearly constant rate for more that 6,000 years (Toblin 1998c, page 3).

The 10-meter (33-foot) doses listed in Table K-12 would be due to leachate concentrations from the storage area with no groundwater dilution. Downgradient doses decrease more rapidly in Regions 1 and 5 than in other regions because of greater groundwater dilution. The downgradient decrease in Region 5 would also be due to the relatively thick aquifer, which results in greater vertical plume spread and increases plume attenuation (Toblin 1998c, pages 4-6).

As shown in Table K-12, an onsite gardener in Region 5 could receive an internal committed dose as high as 3,300 rem for each year of ingestion of plutonium-239 and -240. However, the individual actually would receive only about 70 rem the first year, 140 rem the second year, 210 rem the third year, and so on until reaching an equilibrium annual dose (in approximately 50 years) of 3,300 rem per year. The individual would continue to receive this equilibrium dose as long as the radioactive material uptake remained constant.

If the annual doses are added, in less than 10 years the individual would have received more than 2,000 rem. If the International Commission on Radiological Protection risk conversion factor were applied to this dose, a probability of fatal cancer induction of 1 could be calculated. In other words, the use of this risk conversion would predict that the individual would contract a fatal cancer after 10 years of exposure. This calculated risk is approximately 4 times greater than the lifetime risk of contracting a fatal cancer from all other causes (24 percent).

Table K-13 shows that the direct radiation dose rate to the onsite gardener could be as high as 7,300 rem per year. Unlike internal dose, this dose would actually be delivered during the year of exposure. This maximum value assumes a complete loss of shielding normally provided by the concrete storage module at the same time as the loss of weather protection (see Table K-1). Assuming a dose of 7,300 rem per year, the individual probably would die from acute radiation exposure. This dose would probably cause extensive cell damage in the individual that would result in severe acute adverse health conditions and death within weeks or months (NRC 1996, page 8.29-5). However, these higher radiation dose rates are based on an early estimated time to structural failure of the concrete storage module. If these failure times were extended by as little as 100 years, the associated dose rates would decrease by a factor of 10 because the levels of radiation emanating from the degraded facilities would have decreased by about a factor of 10 due to radioactive decay (Rollins 1998c, page 12).

Table K-13. Estimated external peak dose rates (rem per year) for the onsite and near-site gardeners – Scenario 2.

		Maximally exposed individual distances (meters)[a] from storage facilities			
Region	Year of peak exposure[b]	10[c]	150	1,000	5,000
1	190	7,200	4	0.001	0.0
2	800	28	0.04	0.0	0.0
3	170	7,300	4	0.001	0.0
4	850	31	0.04	0.0	0.0
5	3,600	32	0.05	0.0	0.0

a. To convert meters to feet, multiply by 3.2808.
b. Years after 2116; source: adapted from Poe (1998a, all).
c. Source: Adapted from Davis (1998, all); the maximally exposed individual would be the onsite gardener.

The internal and external dose rates are presented separately because they would occur at different times and are therefore not additive.

K.3.2 UNUSUAL EVENTS

This section includes a quantitative assessment of potential accident impacts and a qualitative discussion of the impacts of sabotage.

K.3.2.1 Accident Scenarios

The analysis examined the impacts of accident scenarios that could occur during the above-ground storage of spent nuclear fuel and high-level radioactive waste and concluded that the most severe accident scenarios would be an aircraft crash into concrete storage modules or a severe seismic event. In Scenario 1, where storage would be in strong rigid concrete storage modules that had not degraded, the accident would not be expected to release radioactive material.

In Scenario 2, the concrete storage modules would deteriorate with time. DOE concluded that an aircraft crash into degraded concrete storage modules would dominate the consequences. The analysis evaluated the potential for criticality accidents and concluded that an event severe enough to produce meaningful consequences would be extremely unlikely, and that the consequences would be bounded by the aircraft crash consequences. Table K-14 lists the consequences of an aircraft crash on a degraded spent fuel concrete storage module.

Table K-14. Consequences of aircraft crash onto degraded spent nuclear fuel concrete storage module.[a]

Impact	High-population site[b]	Low-population site[c]
Collective population dose (person-rem)	26,000	6,000
Latent cancer fatalities	13	3

a. Source: Davis, Strenge, and Mishima (1998, page 11).
b. 330 persons per square mile.
c. 77 persons per square mile.

K.3.2.2 Sabotage

Storage of spent nuclear fuel and high-level radioactive waste over 10,000 years would entail a continued risk of intruder access at each of the 77 sites. Sabotage could result in a release of radionuclides to the environment around the facility. In addition, intruders could attempt to remove fissile material, which could result in releases of radioactive material to the environment. For Scenario 1, the analysis assumed that safeguards and security measures currently in place would remain in effect during the 10,000-year analysis period at the 77 sites. Therefore, the risk of sabotage would continue to be low. However, the difficulty of maintaining absolute control over 77 sites for 10,000 years would suggest that the cumulative risk of intruder attempts would increase.

For Scenario 2, the analysis assumed that safeguards and security measures would not be maintained at the 77 sites after approximately the first 100 years. For the remaining 9,900 years of the analysis period, the cumulative risk of intruder attempts would increase. Therefore, the risk of sabotage would increase substantially under this scenario.

K.4 Uncertainties

Section K.3 contains estimates of the radiological impacts of the No-Action Alternative, which assumes continued above-ground storage of spent nuclear fuel and high-level radioactive waste at sites across the United States. Associated with the impact estimates are uncertainties typical of predictions of the outcome of complex physical and biological phenomena and of the future state of society and societal institutions over long periods. DOE recognized this fact from the onset of the analysis; however, the predictions will be valuable in the decisionmaking process because they provide insight based on the best information and scientific judgments available.

This analysis considered five aspects of uncertainty:

- Uncertainties about the nature of changes in society and its institutions and values, in the physical environment, and of technology as technology progresses
- Uncertainties associated with future human activities and lifestyles
- Uncertainties associated with the mathematical representation of the physical processes and with the data in the computer models

- Uncertainties associated with the mathematical representation of the biological processes involving the uptake and metabolism of radionuclides and the data in the computer models
- Uncertainties associated with accident scenario analysis

The following sections discuss these uncertainties in the context of possible effects on the impact estimates reported in Chapter 7 and Section K.3.

K.4.1 SOCIETAL VALUES, NATURAL EVENTS, AND IMPROVEMENTS IN TECHNOLOGY

K.4.1.1 Societal Values

History is marked by periods of great social upheaval and anarchy followed by periods of relative political stability and peace. Throughout history, governments have ended abruptly, resulting in social instability, including some level of lawlessness and anarchy. The Scenario 1 assumption is that political stability would exist to the extent necessary to ensure adequate institutional control to monitor and maintain the spent nuclear fuel and high-level radioactive waste to protect the workers and the public for 10,000 years. The Scenario 2 assumption is that in the United States political stability would exist for 100 years into the future and that the spent nuclear fuel and high-level radioactive waste would be properly monitored and maintained and the public would be protected for this length of time. If a political upheaval, such as the one that recently occurred in the former Soviet Union, were to occur in the United States, the government could have difficulty protecting and maintaining the storage facilities, and the degradation processes could begin earlier than postulated in Scenario 2. If institutional control were not maintained for at least 100 years, radioactive materials from the spent nuclear fuel and high-level radioactive waste could enter the environment earlier, which would result in higher estimated impacts due to the higher radiotoxicity of the materials. However, this scenario would probably increase overall impacts by no more than a factor of 2.

K.4.1.2 Changes in Natural Events

Because of the difficulty of predicting impacts of climate change (glaciation, precipitation, global warming), DOE decided to evaluate facility degradation and environmental transport mechanisms based on current climate conditions. For example, glaciation, which many scientists agree will occur again within 10,000 years, probably would cover the northeastern United States with a sheet of ice. The ice would crush all structures including spent nuclear fuel and high-level radioactive waste storage facilities and could either disperse the radioactive materials in the accessible environment or trap the materials in the ice sheet. In addition, large populations would migrate from the northeastern United States to warmer climates, thus changing the population distribution and densities throughout the United States (the coastline could move 100 miles out from its current position due to the reduced water in the oceans). Other scientists predict that global warming could lead to extensive flooding of low-lying coastal areas throughout the world. Such changes would have to be known with some degree of certainty to make accurate estimates of potential impacts associated with the release of spent nuclear fuel and high-level radioactive waste materials to the environment. To simplify the analysis, DOE has chosen not to attempt to quantify the impacts resulting from the almost certain climate changes that will occur during the analysis period.

K.4.1.3 Improvements in Technology

We are living in a time of unparalleled technical advancement. It is possible that cures for many common cancers will be found in the coming decades. In this regard, the National Council on Radiation Protection and Measurements (NCRP 1995, page 51) states that:

> *One of the most important factors likely to affect the significance of radiation dose in the centuries and millennia to come is the effect of progress in medical technology. At some future time, it is possible that a greater proportion of somatic [cancer] diseases caused by radiation will be treated successfully. If, in fact, an increased proportion of the adverse health effects of radiation prove to be either preventable or curable by advances in medical science, the estimates of long-term detriments may need to be revised as the consequences (risks) of doses to future populations could be very different.*

Effective cures for cancer would affect the fundamental premise on which the No-Action Alternative impact analysis is based. However, this technology change was not included in the impact analyses.

Other advancements in technology could include advancements in water purification that could reduce the concentration of contaminants in drinking water supplies. Improved corrosion-resistant materials could reduce package degradation rates, which could reduce the release of contaminants and the resultant impacts. In addition, future technology could enable the detoxification of the spent nuclear fuel and high-level radioactive waste materials, thereby removing the risks associated with human exposure.

K.4.2 CHANGES IN HUMAN BEHAVIOR

General guidance for the prediction of the evolution of society has been provided by the National Research Council in *Technical Bases for Yucca Mountain Standards* (National Research Council 1995, pages 28 and 70), in which the Committee on Technical Bases for Yucca Mountain Standards concluded that there is no scientific basis for predicting future human behavior. The study recommends policy decisions that specify the use of default (or reference) scenarios to incorporate future human behaviors into compliance assessment calculations. This No-Action Alternative analysis followed this approach, based on societal conditions as they exist today. In doing so, the analysis assumed that populations would remain at their present locations and that population densities would remain at the current levels. This assumption is appropriate when estimating impacts for comparison with other proposed actions; however, it does not reflect reality. Populations are constantly moving and changing in size. If, for example, populations were to move closer to and increase in size in areas near the storage facilities, the radiation dose and resultant adverse impacts could increase substantially. However, DOE has no way to predict such changes accurately and, therefore, did not attempt to quantify the resultant effects on overall impacts.

Another lifestyle change that could affect the overall impacts would involve food consumption patterns. For example, people might curtail their use of public water supplies derived from rivers if they learned that the river water carried carcinogens. Widespread adoption of such practices could reduce the impacts associated with the drinking water pathway.

K.4.3 MATHEMATICAL REPRESENTATIONS OF PHYSICAL PROCESSES AND OF THE DATA INPUT

The DOE approach for the No-Action Alternative was to be as comparable as possible to the approach used for the predictions of impacts from the proposed Yucca Mountain Repository to enable direct comparisons of the impact estimates for the two cases. Therefore, the analysis either used the process models developed for the Total System Performance Assessment directly or adapted them for the

No-Action Alternative impact calculations. For processes that were different from those treated in the Total System Performance Assessment, DOE developed analytical approaches.

In a general sense, the Total System Performance Assessment calculations used a stochastic (random) approach to develop radiological impact estimates. Existing process models were used to generate a set of responses for a particular process. In the Total System Performance Assessment process, the impact calculations sample each set of process responses and calculate a particular impact result. A large number of calculations were performed. From the set of variable results, an expected value can be identified, as can a distribution of results that is an indication of the uncertainties in the calculated expected values.

For the No-Action Alternative analysis, the calculations were based on only a single set of best estimate parameters. No statistical distribution of results was generated as a basis for the quantification of uncertainties. This section describes the uncertainties associated with the input data and modeling used to evaluate the rates of degradation of the materials considered in this document and to estimate the impacts of the resulting releases. It describes the key assumptions, shows where the assumptions are consistent with Total System Performance Assessment assumptions, and qualitatively assesses the magnitude of the uncertainties caused by the assumptions.

Calculating the radiological impacts to human receptors required a mathematical representation of physical processes (for example, water movement) and data input (for example, material porosity). There are uncertainties in both the mathematical representations and in the values of data. The Total System Performance Assessment accommodates these uncertainties by using a probabilistic approach to incorporate the uncertainties, whereas the No-Action analysis uses a deterministic approach in combination with an uncertainty analysis. When done correctly, both approaches yield the same information, although, as in the case of the Total System Performance Assessment, the probabilistic approach provides quantitative information.

K.4.3.1 Waste Package and Material Degradation

The major approaches and assumptions used for the No-Action Scenario 2 analysis are listed in Table K-15. The table indicates where the continued storage calculations followed the basic methods developed for the Total System Performance Assessment. It also indicates the processes for which models other than those used in the Total System Performance Assessment were applied.

DOE analyzed surface storage of commercial spent nuclear fuel in horizontal stainless-steel canisters inside concrete storage modules. There are other probable forms of storage, including horizontal and vertical casks made of materials ranging from stainless steel to carbon steel. Degradation and releases from vertical carbon-steel casks were evaluated qualitatively. Such storage units would be likely to fail from corrosion earlier than concrete and stainless steel. The concrete and stainless-steel units were calculated to fail and begin releasing their contents at about 1,000 years after the assumed loss of institutional control. The less-resistant carbon-steel units could begin releasing their contents earlier and their use would result in a longer period of release and increased impacts. This difference is likely to be an increase of 10 to 30 percent in population dose commitment and resultant latent cancer fatalities.

K.4.3.2 Consequences of Radionuclide Release

The dose-to-risk conversion factors typically used to estimate adverse human health impacts resulting from radiation exposures contain considerable uncertainty. The risk conversion factor of 0.0005 latent cancer fatality per person-rem of collective dose for the general public typically used in DOE National Environmental Policy Act documents is based on recommendations of the International Commission on Radiological Protection (ICRP 1991, page 22) and the National Council on Radiation Protection and Measurements (NCRP 1993a, page 112). The factor is based on health effects observed in the high dose

Table K-15. Review of approaches, assumptions, and related uncertainties[a] (page 1 of 2).

Approach or assumption	Consistent with repository analysis assumptions	Sensitivity of impacts to approach or assumption[b]
Period of analysis – 10,000 years	Yes	None
Commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste quantities equivalent to NWPA specified 70,000 MTHM and Module 1	Yes	None
No credit for stainless-steel cladding on commercial spent nuclear fuel	Yes	If credit were taken for stainless-steel cladding, LCFs[a] could decrease by as much as a factor of 10.
0.1 percent of zirconium alloy cladding is initially failed	Yes	If energetic events (that is, concrete collapse) had been considered in the No-Action analysis, impacts could have been slightly smaller (additional protection from winds) to a factor of 100 higher.
Concrete storage module weather protection	This is a primary protective barrier for the No-Action analysis and is not applicable to TSPA	If weather protection from the concrete storage module had not been assumed in the No-Action analysis, LCFs could be higher by less than a factor of 10.
Concrete base pad degradation	Not applicable	Used NRC recommended values (probably overestimated degradation and reduced consequences in the No-Action analysis); increase in LCFs by several factors but less than a factor of 10
Credit for stainless-steel canister on high-level radioactive waste	No; TSPA does not take credit for stainless-steel container	If the No-Action analysis had not taken credit for the stainless-steel canister, LCFs would change very little (slight increase) because of the intrinsic stability of the borosilicate glass.
DOE spent nuclear fuel evaluated by a representative surrogate that is based mostly on DOE N-Reactor spent nuclear fuel (other spent nuclear fuel types not evaluated)	Yes	If actual fuel types were evaluated, LCFs could either increase or decrease by less than a factor of 2.
No credit given for zirconium alloy cladding on N-Reactor spent nuclear fuel	Yes	If credit was given for the N-Reactor zirconium alloy cladding, the LCFs would decrease by less than a factor of 2.
Stainless steel deterioration	Model paralleled TSPA approach for Alloy-22	Model based on best information; if incorrect and corrosion proceeds more rapidly and stainless steel offers no protection, LCFs could increase by as much as a factor of 100
Zirconium alloy cladding deterioration	Yes, very slow corrosion rate.	If the No-Action analysis had assumed larger or smaller deterioration rates, LCFs could have increased by several orders of magnitude or decreased by less than a factor of 2.
Zirconium alloy cladding credit	Yes	If the No-Action analysis had not taken credit for zirconium alloy cladding, LCFs could have increased by as much as 2 orders of magnitude.
Deterioration of spent nuclear fuel and high-level radioactive waste core materials	Yes	None

Table K-15. Review of approaches, assumptions, and related uncertainties[a] (page 2 of 2).

Approach or assumption	Consistent with repository analysis assumptions	Sensitivity of impacts to approach or assumption[b]
Use of recent regional climate conditions to determine deterioration (temperature, precipitation, etc.)	No; No-Action analysis used constant "effective" regional weather parameters weighted for material inventories and potentially affected downstream populations; TSPA used actual weather patterns measured at Yucca Mountain. The TSPA also assumed long-term climate changes would occur in the form of increased precipitation.	If actual site climate data and projected future potential climate changes had been considered in the No-Action analysis, LCFs could have increased or decreased by as much as a factor of 10. Climate change assumptions such as a glacier covering most of the northeastern seaboard of the United States would have made estimating impacts from continued storage virtually impossible.
Surface transport by precipitation	Not applicable; TSPA only considered groundwater transport because there is no surface-water transport pathway possible for the repository.	If the No-Action analysis had not considered the groundwater transport pathway, LCFs could have been as much as a factor of 10 higher.
Regional binning of sites – not specific site parameters	Not applicable; TSPA considered only a single site; the No-Action analysis evaluated potential impacts from 77 sites on a regional basis.	None, the No-Action analysis binned sites into categories and developed "effective" regional climate conditions such that calculated impacts would be comparable to those which could be calculated by a site-specific analysis.
Atmospheric dose consequences judged to be small when compared to liquid pathways.	Yes	Small impact on LCFs
Drinking water doses	Yes; primary pathway evaluated	Use of drinking-water-only pathway underestimates total collective LCFs by less than a factor of 3.
Used the Multimedia Environmental Pollutant Assessment System[c] (Buck et al. 1995, all (Leigh et al. 1993, all) modeling approach for calculating population uptake/ingestion	No; TSPA uses GENII-S.[d] GENII-S uses local survey data; the Multimedia Environmental Pollutant Assessment System uses EPA/NRC exposure/uptake default and actual population data	No impact. The two programs yield comparable results as used in these analyses.
ICRP[e] approach to calculate dose commitment from ingested radionuclides	Yes	No impact.
Human health impacts calculated as LCFs with NCRP[f] conversion factors	NA; TSPA does not estimate LCFs.	Use of other than the linear no-threshold model could result in a change in estimated LCFs from 0.25 to 2 times the nominal value.[g]

a. Abbreviations: NWPA = Nuclear Waste Policy Act; MTHM = metric tons of heavy metal; LCF = latent cancer fatality; TSPA = Total System Performance Assessment; NRC = Nuclear Regulatory Commission; ICRP = International Commission on Radiological Protection; EPA = Environmental Protection Agency.
b. Sensitivity of impacts to approach/assumption is based on professional judgement and, if applicable, the effects of the approaches/assumptions on calculations.
c. Buck et al. (1995, all).
d. Leigh et al. (1993, al).
e. ICRP (1979, all).
f. NCRP (1993a, page 112).
g. NCRP (1997, page 75).

and high dose rate region (20 to 50 rem per year). Health effects were extrapolated to the low-dose region (less than 10 rem per year) using the linear no-threshold model. This model is generally recommended by the International Commission on Radiological Protection and the National Council of Radiation Protection and Measurements, and most radiation protection professionals believe this model produces a conservative estimate (that is, an overestimate) of health effects in the low-dose region, which is the exposure region associated with continued storage of spent nuclear fuel and high-level radioactive waste. This report summarizes estimates of the impacts associated with very small chronic population doses to enable comparison of alternatives in this EIS. These impact estimates should be viewed as conservatively high; in fact, the uncertainties are such that the actual level of impact could be zero.

According to the National Council on Radiation Protection and Measurements, the results of an analysis of the uncertainties in the risk coefficients "show a range (90 percent confidence intervals) of uncertainty values for the lifetime risk for both a population of all ages and an adult worker population from about a factor of 2.5 to 3 below and above the 50th percentile value" (NCRP 1997, page 74).

The National Council on Radiation Protection and Measurements states, "This work indicates that given the sources of uncertainties considered here, together with an allowance for unspecified uncertainties, the values of the lifetime risk can range from about one-fourth or so to about twice the nominal values" (NCRP 1997, page 75).

Because of the large uncertainties that exist in the dose/effect relationship, the Health Physics Society has recommended "...against quantitative estimation of health risks due to radiation exposure below a lifetime dose of 10 rem ..." (HPS 1996, page 1). In essence, the Society has recommended against the quantification of risks due to individual radiation exposures comparable to those estimated in the No-Action analysis. These uncertainties are due, in part, to the fact that epidemiological studies have been unable to demonstrate that adverse health effects have occurred in individuals exposed to small doses (less than 10 rem per year) over a period of many years (chronic exposures) and to the fact that the extent to which cellular repair mechanisms reduce the likelihood of cancers is unknown.

Other areas of uncertainty in estimation of dose and risk include the following:

- *Uncertainties Related to Plant and Human Uptake of Radionuclides.* There are large uncertainties related to the uptake (absorption) of radionuclides by agricultural plants, particularly in the case where "regionalized," versus "site-specific" data are used. Also of importance are variations in the absorption of specific radionuclides through the human gastrointestinal tract. Factors that influence the absorption of radionuclides include their chemical or physical form, their concentrations, and the presence of stable elements having similar chemical properties. In the case of agricultural crops, many of these factors are site-specific.

- *Uncertainties in Dose and Risk Conversion Factors.* The magnitudes and sources of the uncertainties in the various input parameters for the analytical models need to be recognized. In addition to the factors cited above, these include those required for converting absorbed doses into equivalent doses, for calculating committed doses, and for converting organ doses into effective (whole body) doses. Although these various factors are commonly assigned point values for purposes of dose and risk estimates, each of these factors has associated uncertainties.

- *Conservatisms in Various Models and Parameters.* In addition to recognizing uncertainties, one must take into account the magnitudes and sources of the conservatisms in the parameters and models being used. These include the fact that the values of the tissue weighting factors and the methods for calculating committed and collective doses are based on the assumption of a linear no-threshold relationship between dose and effect. As the International Commission on Radiological Protection and the National Council on Radiation Protection and Measurements have stated, the use of the linear no-threshold hypothesis provides an upper bound on the associated risk (ICRP 1966, page 56). Also to be considered is that the concept of committed dose could overestimate the actual dose by a factor of 2 or more (NCRP 1993b, page 25).

K.4.3.3 Accidents and Their Uncertainty

The accident methodology used in this analysis is described in Section K.2.5 for Scenarios 1 and 2. It states that for Scenario 1 an aircraft crash into the storage array would provide the most severe accident scenario and its consequences would not cause a release from the rugged concrete storage module. The analysis placed considerable weight on the quality and strength of the concrete storage module and dry storage canister. For an analysis extending 10,000 years, more severe natural events can be postulated than those used as the design basis for the dry storage canister, and they could cause failure of the canister. This could exceed the consequences estimated for Scenario 1, but it would be unlikely to exceed the consequences for the aircraft accident scenario evaluated for Scenario 2.

Section K.2.5.1 concludes that the aircraft crash on the degraded concrete storage modules would be the largest credible event that could occur. The best estimate impacts from this event ranged from 3 latent cancer fatalities for a low-population site to 13 for a high-population site. The uncertainties in these estimates are very large. As discussed above, the aircraft crash could cause a minimum of no latent cancer fatalities given the uncertainty in the model that converts doses to cancers. The maximum impact could be 50 times greater than the estimated values if an aircraft crash involving the largest commercial jet occurred at the time of initial concrete storage module degradation at a northern site under adverse weather conditions (conditions that would maximize the offsite doses) involving spent fuel with the maximum expected inventory of radionuclides.

K.4.4 UNCERTAINTY SUMMARY

The sections above discuss qualitatively and semiquantitatively the uncertainties associated with impact estimates resulting from the long-term storage of spent nuclear fuel and high-level radioactive waste at multiple sites across the United States. As stated above, DOE has not attempted to quantify the variability of estimated impacts related to possible changes in climate, societal values, technology, or future lifestyles. Although uncertainties with these changes could undoubtedly affect the total consequences reported in Section K.3 by several orders of magnitude, DOE did not attempt to quantify these uncertainties to simplify the analysis.

DOE attempted to quantify a range of uncertainties associated with mathematical models and input data, and estimated the potential effect these uncertainties could have on collective human health impacts. By summing the uncertainties discussed in Sections K.4.1, K.4.2, and K.4.3 where appropriate, DOE estimates that total collective impacts over 10,000 years could have been underestimated by as much as 3 or 4 orders of magnitude. However, because there are large uncertainties in the models used for quantifying the relationship between low doses (that is, less than 10 rem) and the accompanying health impacts, especially under conditions in which the majority of the populations would be exposed at a very low dose rate, the actual collective impact could be zero.

On the other hand, impacts to individuals (human intruders) who could move to the storage sites and live close to the degraded facilities could be severe. During the early period (200 to 400 years after the assumed loss of institutional control), acute exposures to external radiation from the spent nuclear fuel and high-level radioactive waste material could result in prompt fatalities. In addition, after a few thousand years onsite shallow aquifers could be contaminated to such a degree that consumption of water from these aquifers could result in severe adverse health effects, including premature death. Uncertainties related to these localized impacts are related primarily to the inability to predict accurately how many individuals could be affected at each of the 77 sites over the 10,000-year analysis period. In addition, the uncertainties associated with localized impacts would exist for potential consequences resulting from unusual events, both manmade and natural.

Therefore, as listed in Table K-15, uncertainties resulting from future changes in natural phenomena and human behavior that cannot be predicted, process model uncertainties, and dose-effect relationships, taken together, could produce the results presented in Section K.3, overestimating or underestimating the impacts by as much as several orders of magnitude. Uncertainties of this magnitude are typical of predictions of the outcome of complex physical and biological phenomena over long periods. However, these predictions (with their uncertainties) are valuable to the decisionmaking process because they provide insight based on the best information available.

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Appendix L

Floodplain/Wetlands Assessment for the Proposed Yucca Mountain Geologic Repository

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APPENDIX L. FLOODPLAIN/WETLANDS ASSESSMENT FOR THE PROPOSED YUCCA MOUNTAIN GEOLOGIC REPOSITORY

L.1 Introduction

Pursuant to Executive Order 11988, *Floodplain Management*, each Federal agency is required, when conducting activities in a floodplain, to take actions to reduce the risk of flood damage; minimize the impact of floods on human safety, health, and welfare; and restore and preserve the natural and beneficial values served by floodplains. Pursuant to Executive Order 11990, *Protection of Wetlands*, each Federal agency is to avoid, to the extent practicable, the destruction or modification of wetlands, and to avoid direct or indirect support of new construction in wetlands if a practicable alternative exists. Regulations issued by the U.S. Department of Energy (DOE) that implement these Executive Orders are contained in Title 10 of the Code of Federal Regulations (CFR) Part 1022, *Compliance with Floodplain/Wetlands Environmental Review Requirements*.

In 1982, Congress enacted the *Nuclear Waste Policy Act* in recognition of the national problem created by the accumulation of spent nuclear fuel and high-level radioactive waste at many commercial and DOE sites throughout the country. The Act recognized the Federal government's responsibility to permanently dispose of the Nation's spent nuclear fuel and high-level radioactive waste. By 1986, DOE narrowed the number of potentially acceptable geologic repository sites to three. Then in 1987, Congress amended the Act by redirecting DOE to determine the suitability of only Yucca Mountain in southern Nevada.

If, after a possible recommendation by the Secretary of Energy, the President considers the site qualified for an application to the U.S. Nuclear Regulatory Commission for a construction authorization, the President will submit a recommendation of the site to Congress. If the site designation becomes effective, the Secretary of Energy will submit to the Nuclear Regulatory Commission a License Application for a construction authorization. DOE could then select a rail corridor or a site for an intermodal transfer station, along with its associated route for heavy-haul trucks, among those considered for Nevada in the EIS. Following such a decision, additional field surveys, environmental and engineering analyses, and National Environmental Policy Act reviews would likely be needed regarding a specific rail alignment for the selected corridor or the site for the intermodal transfer station and its associated route. When more specific information becomes available about activities proposed to take place within floodplains and wetlands, DOE will conduct further environmental review in accordance with 10 CFR 1022.

In 1989, DOE published a Notice of Floodplain/Wetlands Involvement (54 *FR* 6318, February 9, 1989) for site characterization studies at Yucca Mountain. These studies are designed to determine the suitability of Yucca Mountain to isolate nuclear waste. A floodplain assessment was prepared (DOE 1991, all) and a Statement of Findings was issued by DOE (56 *FR* 49765, October 1, 1991). In 1992, DOE prepared a second floodplain assessment on locating part of the entry point to the subsurface Exploratory Studies Facility in the 100-year floodplain of a wash at Yucca Mountain (DOE 1992, all). The Statement of Findings for this assessment was published in the Federal Register (57 *FR* 48363, October 23, 1992). Both Statements of Findings concluded that the benefits of locating activities and structures in the floodplains outweigh the potential adverse impacts to the floodplains and that alternatives to these actions were not reasonable.

The Nuclear Waste Policy Act, as amended, requires that a recommendation by the Secretary to the President to construct a repository must be accompanied by a Final EIS. As part of the EIS process, and following the requirements of 10 CFR Part 1022, DOE issued a *Notice of Floodplain and Wetlands Involvement* in the *Federal Register* (64 *FR* 31554, June 11, 1999). The Notice requested comments from the public regarding potential impacts on floodplains and wetlands associated with construction of a potential rail line or a potential intermodal transfer station with its associated route for heavy-haul trucks to and in the vicinity of Yucca Mountain, depending on the rail or intermodal alternative selected (Figure L-1). As of July 2,1999, DOE had received no comments from the public. This floodplain/wetlands assessment has been prepared in conjunction with the *Notice of Floodplain and Wetlands Involvement*, and in accordance with 10 CFR Part 1022.

This assessment examines the effects of proposed repository construction and operation and potential construction of a rail line or intermodal transfer station on:

1. Floodplains near the Yucca Mountain site (Fortymile Wash, Busted Butte Wash, Drillhole Wash, and Midway Valley Wash; there are no delineated wetlands near the Yucca Mountain site), and

2. Floodplains and areas that may have wetlands (for example, springs and riparian areas) along potential rail corridors in Nevada and at intermodal transfer station locations associated with routes for heavy-haul trucks. If DOE selects rail as the mode of spent nuclear fuel and high-level radioactive waste transport in Nevada to the Yucca Mountain site, one of five rail corridors would be selected (Figure L-2). If DOE selects heavy-haul as the mode of transport for spent nuclear fuel and high-level radioactive waste to the Yucca Mountain site, one of five corridors and one of three intermodal transfer station locations would be selected (Figure L-3). A more detailed floodplain/wetlands assessment of the selected rail corridor or route for heavy-haul trucks would then be prepared. This assessment compares what is known about the floodplains, springs, and riparian areas along the five possible rail corridors and at the three intermodal transfer station locations. This assessment does not evaluate potential floodplain or wetlands effects along routes because these existing roads should already be designed to meet 100-year floodplain design specifications. If upgrades to existing roads are deemed necessary, a more detailed floodplain/wetlands assessment would be prepared at that time.

Title 10 CFR Part 1022.4 defines a flood or flooding as "*...a temporary condition of partial or complete inundation of normally dry land areas from....the unusual and rapid accumulation of runoff of surface waters...*" Title 10 CFR Part 1022.4 identifies floodplains that must be considered in a floodplain assessment as the *base floodplain* and the *critical-action floodplain*. The base floodplain is the area inundated by a flood having a 1.0 percent chance of occurrence in any given year (referred to as the 100-year floodplain). The critical-action floodplain is the area inundated by a flood having a 0.2 percent chance of occurrence in any given year (referred to as the 500-year floodplain). *Critical action* is defined as any activity for which even a slight chance of flooding would be too great. Such actions could include the storage of highly volatile, toxic, or water-reactive materials. The critical-action floodplain was considered because petroleum, oil, lubricants, and other hazardous materials could be used during the construction of a rail line or road upgrades and because spent nuclear fuel and high-level radioactive waste would be transported across the washes.

Title 10 CFR Part 1022.11 requires DOE to use Flood Insurance Rate Maps or Flood Hazard Boundary Maps to determine if a proposed action would be located in the base or critical-action floodplain. On Federal or state lands where Flood Insurance Rate Maps or Flood Hazard Boundary Maps are not available, DOE is required to seek flood information from the appropriate land-management agency or from agencies with expertise in floodplain analysis. The U.S. Geological Survey was therefore asked by DOE to complete a flood study of Fortymile Wash and its principal tributaries (which include Busted Butte, Drillhole, and Midway Valley washes) and outline areas of inundation from 100-year and 500-year floods (Squires and Young 1984, Plate 1).

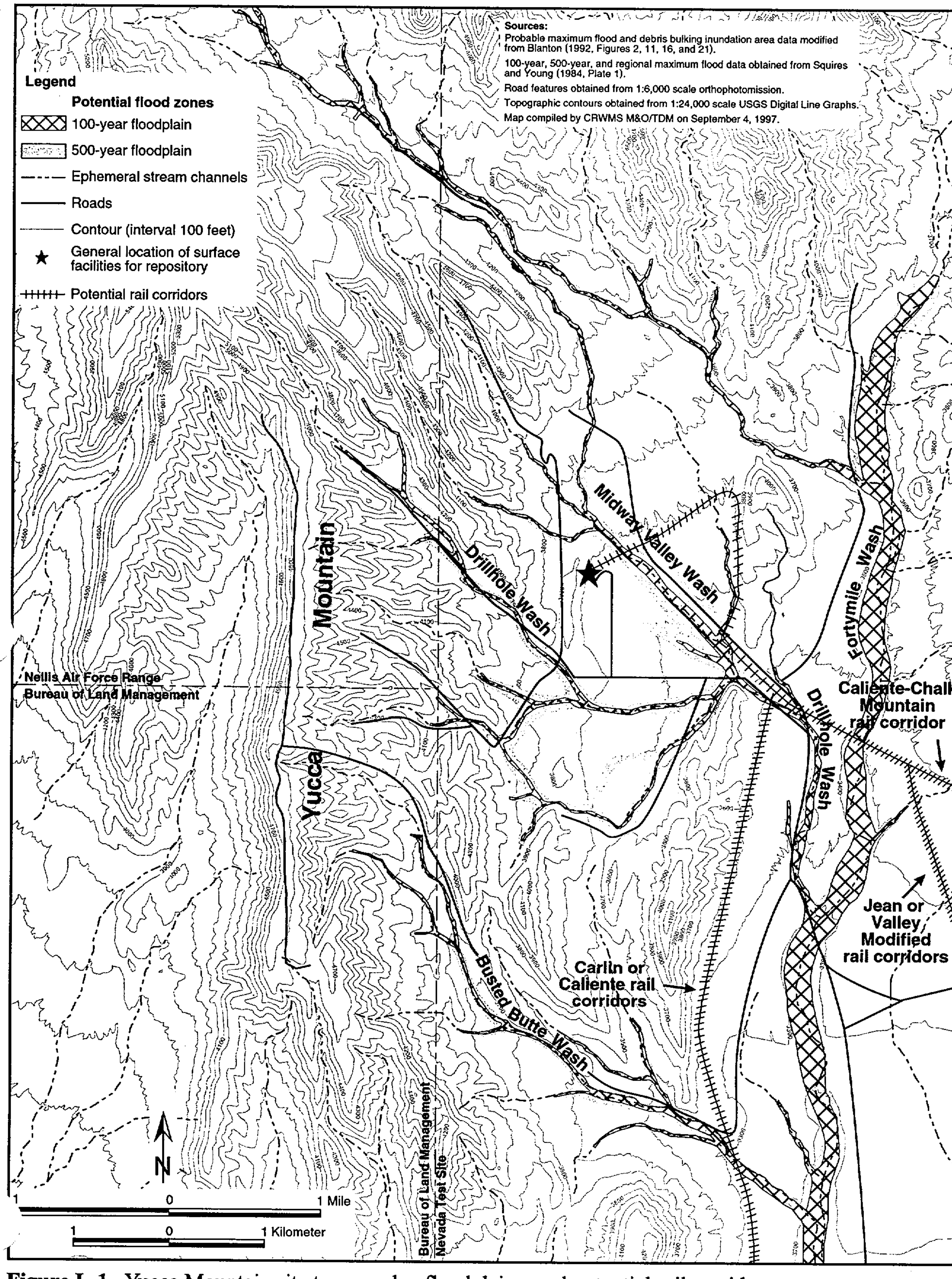

Figure L-1. Yucca Mountain site topography, floodplains, and potential rail corridors.

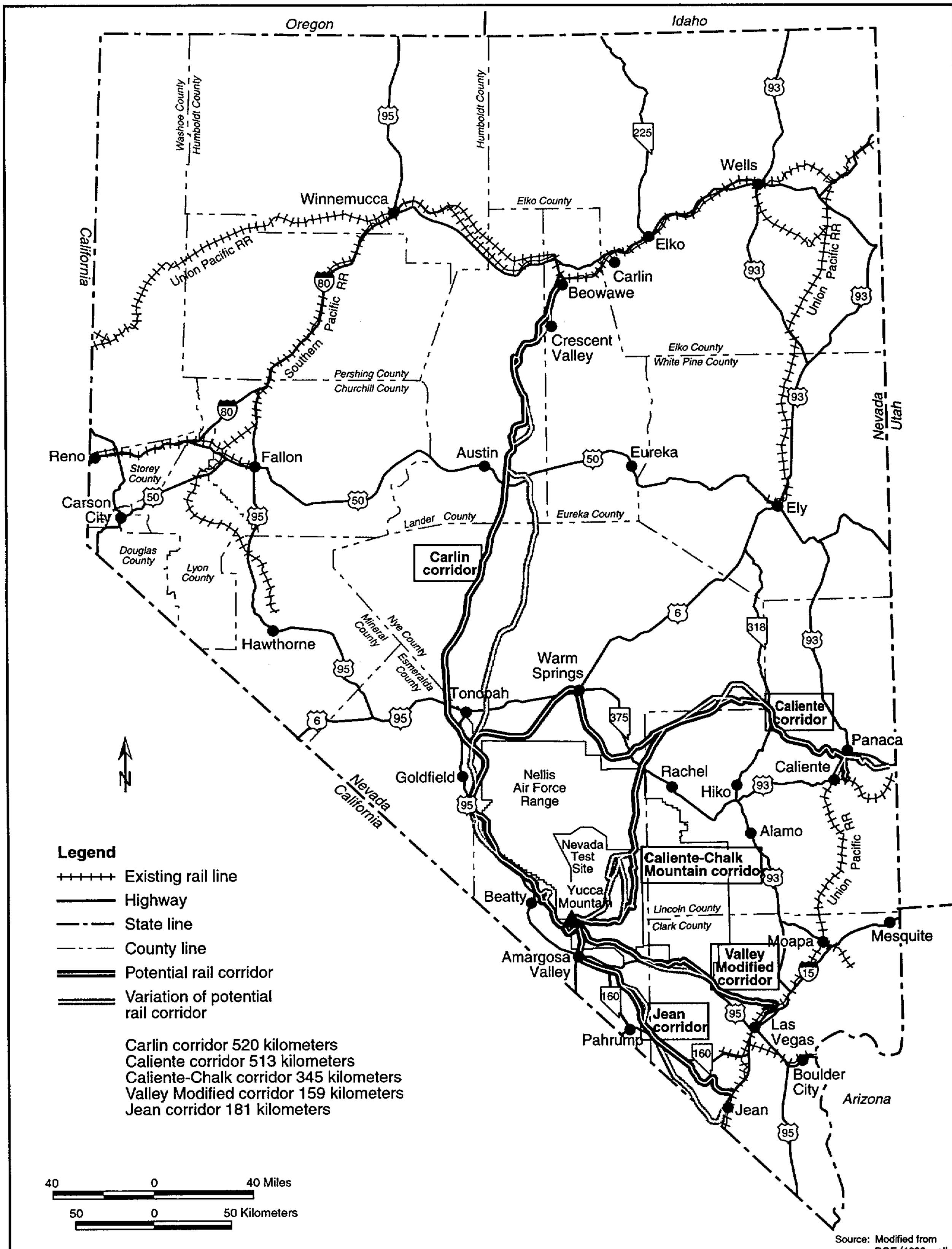

Figure L-2. Potential Nevada rail corridors to Yucca Mountain.

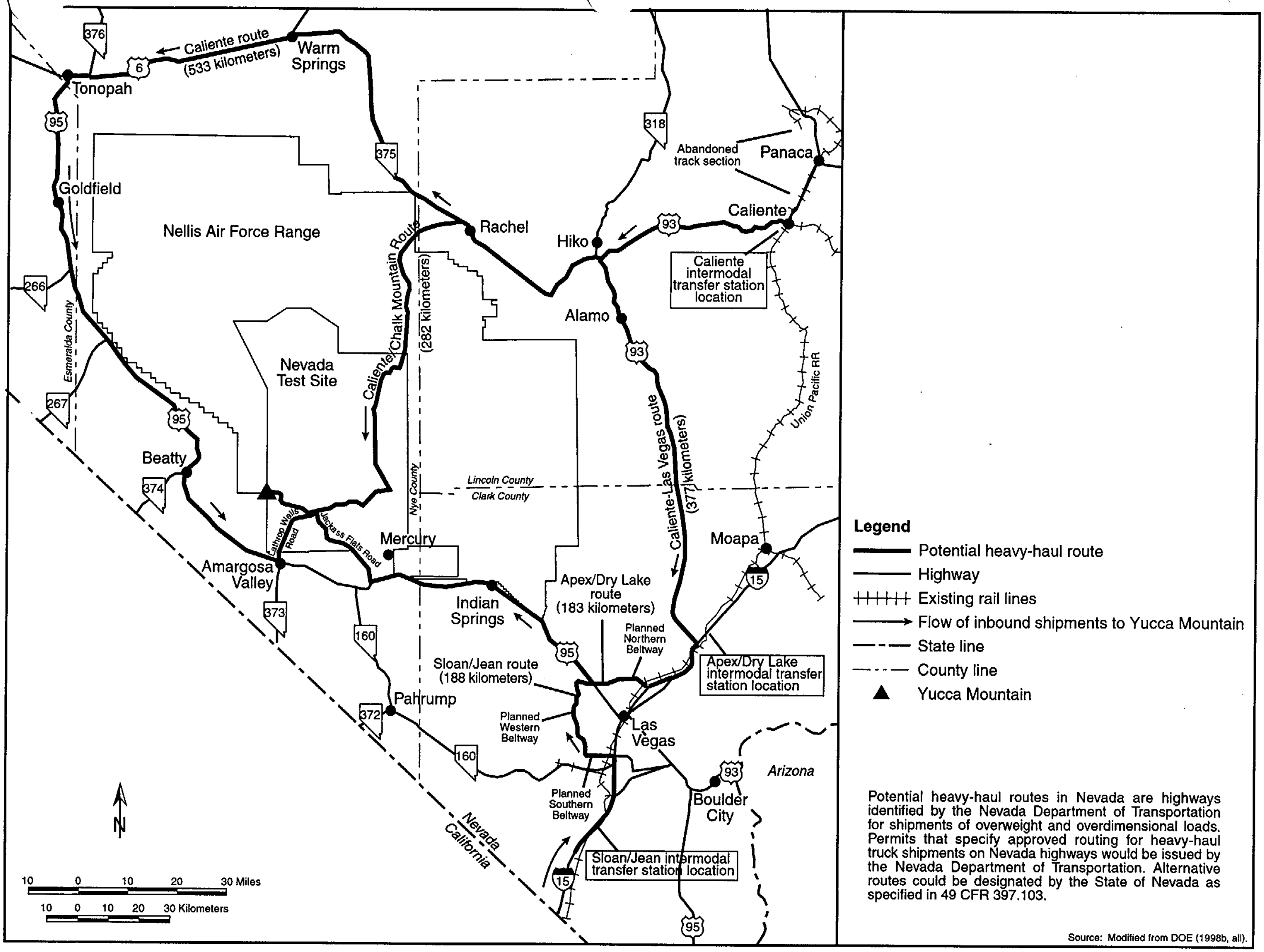

Figure L-3. Potential routes in Nevada for heavy-haul trucks.

Title 10 CFR Part 1022 also requires DOE to determine whether wetlands would be affected by the proposed action and, if necessary, to conduct a wetlands assessment. As required by 10 CFR Part 1022.11(c), DOE examined the following information with regard to possible wetlands in the vicinity of the Yucca Mountain site:

- *U.S. Fish and Wildlife Service National Wetlands Inventory.* Maps from the National Wetlands Inventory do not identify any naturally occurring wetlands in the vicinity of the Yucca Mountain site (FWS 1995, all).
- *U.S. Department of Agriculture, Soil Conservation Service Local Identification Maps.* The Soils Conservation Service (now called Natural Resource Conservation Service) has not conducted a soil survey of the Yucca Mountain site. However, DOE and other agencies have conducted comprehensive surveys and studies of soils at the Yucca Mountain site and in the surrounding area. These surveys are summarized in TRW (1999a, pages 2 to 6). The surveys indicate that there are no naturally-occurring hydric soils at Yucca Mountain.
- *U.S. Geological Survey Topographic Maps.* Topographic maps of the vicinity (for example, USGS 1983, all) do not show springs, permanent streams, or other indications of wetlands.
- *State Wetlands Inventories.* There are no State of Nevada wetlands inventories in the vicinity of Yucca Mountain.
- *Regional or Local Government-Sponsored Wetlands or Land-Use Inventories.* DOE has conducted a wetlands inventory of the Nevada Test Site (Hansen et al. 1997, page 1-161). The closest naturally occurring wetlands to Yucca Mountain is on the upper west slope of Fortymile Canyon, 6 kilometers (3.7 miles) north of the North Portal, outside of the proposed repository construction area. In addition, riparian vegetation occurs adjacent to four man-made well ponds east of Yucca Mountain (TRW 1999b, page 2-14), but these are outside of areas where construction or other proposed actions would occur.

Based on this information, DOE concluded that a wetlands assessment is not required to comply with 10 CFR Part 1022.

L.2 Project Description

If Yucca Mountain is selected as a site to construct a repository, DOE would ship spent nuclear fuel and high-level radioactive waste to the site for a period of about 24 years. Under the current schedule spent nuclear fuel and high-level radioactive waste emplacement would begin in 2010. One of five possible rail corridors leading to the site could be selected in Nevada (Figure L-2). In the vicinity of the Yucca Mountain site the five rail corridors converge to two possible routes. Alternatively, if heavy-haul transport were selected, one intermodal transfer station and one associated route would be identified from the three potential intermodal transfer station locations and five potential routes for heavy-haul trucks (Figure L-3). In the vicinity of the Yucca Mountain site, the potential routes converge to two possible routes that may require upgrades. At greater distances, routes would utilize public roads and existing Nevada Test Site roads to the extent possible.

Some transportation-related actions associated with the DOE proposal would occur in floodplains on the proposed repository site on land the Federal government would manage. Route construction and operation could affect the 100-year and 500-year floodplains of Fortymile Wash, Busted Butte Wash, Drillhole Wash, and Midway Valley Wash in the vicinity of the Yucca Mountain site. This assessment examines the potential floodplain impacts to all four washes although all four might not be affected. The effects on floodplains and areas that may contain wetlands elsewhere in Nevada along the five rail corridors and at the three intermodal station locations associated with heavy-haul transport are examined using available information. When DOE makes a decision whether to use rail or heavy-haul transport, more information would be obtained to support further environmental review.

This section is divided into two parts. Section L.2.1 discusses the proposed action in the vicinity of the Yucca Mountain site including rail access; heavy-haul truck access; and potential construction of an associated rail line, bridge, and roads. Section L.2.2 discusses possible actions elsewhere in Nevada including rail access and intermodal transfer station locations.

L.2.1 PROPOSED ACTIONS AT YUCCA MOUNTAIN

The preliminary layout of surface facilities at the repository is shown on Figure L-1. Except for a possible rail line and roads, no facilities are generally anticipated to be located within either the 100-year or 500-year floodplains of Fortymile Wash, Busted Butte Wash, Drillhole Wash, or Midway Valley Wash. The paragraphs below describe the rail line and roads that could affect the floodplains of these washes in the vicinity of the Yucca Mountain site.

L.2.1.1 Rail Access

At this time, there is no rail access to the Yucca Mountain site. DOE has identified five potential rail corridors in Nevada for transporting spent nuclear fuel and high-level radioactive waste to Yucca Mountain.

If DOE selected a rail corridor leading to the Yucca Mountain site from the west and south (either the Carlin or Caliente corridors), the rail line could cross Busted Butte Wash, Drillhole Wash just west of its confluence with Fortymile Wash, and Midway Valley Wash (Figure L-1). Cut, fill, drainage culverts or bridges could be used to cross Busted Butte, Drillhole, and Midway Valley washes. The widths of Busted Butte Wash and Drillhole Wash (including their floodplains) are about 150 meters (500 feet) each where they would be crossed by the rail line. The width of Midway Valley Wash (including its floodplain) is about 300 meters (1,000 feet) where it could be crossed by the rail line.

If DOE selected a rail corridor leading to the Yucca Mountain site from the east (Caliente-Chalk Mountain, Jean, or Valley-Modified corridors) the rail line could cross approximately 400 meters (1,300 feet) of Fortymile Wash and its associated floodplains. In this case, the rail line could cross the wash on either a bridge (with supports located in the wash) or on a raised rail line that could be constructed in the wash (with appropriately-sized drainage culverts). After crossing Fortymile Wash, the rail line could continue along the east side of Yucca Mountain and cross about 300 meters (1,000 feet) of Midway Valley Wash before arriving at the repository.

L.2.1.2 Heavy-Haul Truck Access

DOE has identified five potential routes for heavy-haul trucks in Nevada for transporting spent nuclear fuel and high-level radioactive waste to the Yucca Mountain site.

If DOE selected a route leading to the Yucca Mountain site from the west and south, the route could cross Busted Butte Wash, Drillhole Wash, and Midway Valley Wash (Figure L-1). Cut, fill, drainage culverts or bridges could be used to cross Busted Butte, Drillhole, and Midway Valley washes.

If DOE selected a route leading to the Yucca Mountain site from the east, the route could cross Fortymile Wash. The route could either cross through the wash or a bridge could be constructed over it. After crossing Fortymile Wash, the route could continue along the east side of Yucca Mountain and could cross Midway Valley Wash before arriving at the repository.

During potential repository operation, some spent nuclear fuel and high-level radioactive waste would be transported to the Yucca Mountain site by legal-weight trucks. These trucks could access Yucca Mountain from the east by crossing Fortymile Wash along the existing road or access Yucca Mountain along the route used by heavy-haul trucks. The legal-weight trucks could then proceed along the east side of Yucca Mountain and cross Midway Valley Wash along the route.

L.2.1.3 Construction

Construction of a potential rail line near Yucca Mountain as well as upgrading the existing roads for heavy-haul and legal-weight trucks in the vicinity would take about one year to complete. Standard construction practices would be used, including the use of explosives and heavy earth-moving equipment. Standard measures would also be used to minimize erosion. Petroleum fuels, oils, lubricants and other hazardous materials would be used during construction, although these materials would be stored outside the 500-year floodplain.

Construction aggregate could be obtained from local borrow pits, but rail-bed ballast would need to be obtained from outside sources. Concrete would be obtained from a nearby concrete batch plant or from a new batch plant that may be built closer to the repository site. Neither the borrow pits nor the concrete batch plant would be located in a floodplain or wetlands.

If a bridge were constructed across Fortymile Wash, it would be about 30 meters (100 feet) wide. Supports for the bridge would be constructed in the floodplain of the wash. If a rail line were constructed across the bottom of Fortymile Wash, extensive earthwork (cut and fill) would be required to maintain the less than two percent grade required for the rail alignment.

L.2.2 POSSIBLE ACTIONS ELSEWHERE IN NEVADA

At this time there is no rail access to Yucca Mountain. This means that material traveling by rail would have to continue to the repository on a new branch rail line or transfer to heavy-haul trucks at an intermodal transfer station in Nevada and then travel on existing highways. DOE is considering construction of *either* a new branch rail line *or* an intermodal transfer station and associated highway improvements. The DOE has identified five possible rail corridors, each of which has alignment variations (Figure L-2), and three possible locations for an intermodal transfer station associated with heavy-haul trucks (Figure L-3).

For analytical purposes, it is assumed that construction of a rail line in Nevada would take approximately two and one half years. If a decision were made to proceed with development of a repository, it is likely that the DOE would decide at that time whether to build a rail line or to develop an intermodal transfer station site for heavy-haul waste transport. Should the DOE decide to construct a rail line, standard practices for construction of rail lines would be used, including minimizing steep grades, utilizing cut and full earthwork techniques, and crossing flood prone areas using culverts or bridges. Should the DOE decide to use a route for heavy-haul trucks, portions of the existing roads used for heavy-haul transport may require upgrades to accommodate the heavy loads.

L.3 Existing Environment

L.3.1 EXISTING ENVIRONMENT AT YUCCA MOUNTAIN

Fortymile Wash is about 150 kilometers (93 miles) long and drains an area of about 810 square kilometers (310 square miles) to the east and north of Yucca Mountain (Figure L-1). The wash continues southward and connects to the Amargosa River. The Amargosa River drains an area of about 8,000 square kilometers (3,100 square miles) by the time it reaches Tecopa, California. The mostly-dry river bed extends another 90 kilometers (56 miles) before ending in Death Valley.

Busted Butte and Drillhole washes drain the east side of Yucca Mountain and flow into Fortymile Wash (Figure L-1; Midway Valley Wash is a tributary to Drillhole Wash). Busted Butte Wash drains an area of 17 square kilometers (6.6 square miles) and Drillhole Wash drains an area of 40 square kilometers (15 square miles).

The existing environment at and near Yucca Mountain, including Fortymile Wash, Busted Butte Wash, Drillhole Wash, and Midway Valley Wash is described in Chapter 3 of the EIS. The information below summarizes several of the more important aspects of the environment that pertain to this floodplain assessment.

L.3.1.1 Flooding

Water flow in the four washes is rare. The arid climate and meager precipitation [about 10 to 25 centimeters (4 to 10 inches) per year at Yucca Mountain] result in quick percolation of surface water into the ground and rapid evaporation. Flash floods, however, can occur after unusually strong summer thunderstorms or during sustained winter precipitation. During these times, runoff from ridges, pediments, and alluvial fans flows into the normally dry washes that are tributary to Fortymile Wash. Estimated peak discharges in Fortymile Wash are 340 cubic meters per second (720,000 cubic feet per second) for the 100-year flood and 1,600 cubic meters per second (3,390,000 cubic feet per second) for the 500-year flood. Estimated peak discharges in Busted Butte Wash are 40 cubic meters per second (85,000 cubic feet per second) for the 100-year flood and 180 cubic meters per second (380,000 cubic feet per second) for the 500-year flood. Estimated peak discharges in Drillhole Wash are 65 cubic meters per second (140,000 cubic feet per second) for the 100-year flood and 280 cubic meters per second (590,000 cubic feet per second) for the 500-year flood.

The nearest man-made structure within Fortymile Wash is U.S. Highway 95 more than 19 kilometers (12 miles) south of the confluence of Drillhole and Fortymile washes. Lathrop Wells, the nearest population center to Yucca Mountain, is also about 19 kilometers to the south along U.S. 95 and 3.2 kilometers (2 miles) east of Fortymile Wash.

L.3.1.2 Wetlands

There are no springs, perennial streams, hydric soils, or naturally occurring wetlands at Yucca Mountain. There are two man-made well ponds within Fortymile Wash, and two east of that wash, that have riparian vegetation (TRW 1999a, pages 5 to 6; TRW 1999b, page 2-14).

L. 3.1.3 Biology

Vegetation at and near Fortymile Wash is typical of the Mojave Desert. The mix or association of vegetation in Fortymile Wash, which is dominated by the shrubs white bursage (*Ambrosia dumosa*), creosotebush (*Larrea tridentata*), white burrobush (*Hymenoclea salsola*), and heathgoldenrod (*Ericameria paniculata*), differs somewhat from other vegetation association at Yucca Mountain (TRW 1998a, pages 5 to 7). No plant species are known to be restricted to the floodplains. In addition, none of the more than 180 plant species known to occur at Yucca Mountain is endemic to the area.

None of the 36 mammal, 27 reptile, or 120 bird species that have been documented at Yucca Mountain are restricted to or dependent on the floodplain. These species all are widespread throughout the region. No amphibians have been found at Yucca Mountain.

The only plant or animal species that has been found at Yucca Mountain that is classified as threatened, endangered, or proposed under the Endangered Species Act is the desert tortoise (*Gopherus agassizii*) which is classified as threatened. Yucca Mountain is at the northern edge of the range of the desert tortoise (Rautenstrauch, Brown, and Goodwin 1994, page 11). Desert tortoises are known to occur within the floodplain of Fortymile Wash, but their abundance there and elsewhere at Yucca Mountain is low compared to other parts of its range farther south and east (TRW 1997, pages 6 to 11). Information on the ecology of the desert tortoise population at Yucca Mountain is summarized in TRW (1999b, page 2-8).

Four species classified as sensitive by the Bureau of Land Management occur at Yucca Mountain: two species of bats [the long-legged myotis (*Myotis volans*) and the fringed myotis (*Myotis thysanodes*)] (TRW 1998b, page 11), the western chuckwalla (*Sauromalus obesus obesus*) (TRW 1998c, pages 22 to 23), and the western burrowing owl (*Speotyto cunicularia hypugaea*) (Steen et al. 1997, pages 19 to 29). These species may occur within the floodplain of Fortymile Wash, but they are not dependent upon habitat there (TRW 1998b, page 8; TRW 1998c, pages 22 to 23; Steen et al. 1997, pages 19 to 29).

L.3.1.4 Archaeology

Archaeological surveys have been conducted in Fortymile Wash east of Yucca Mountain. Fortymile Wash was an important crossroad where several trails converged from such distant places as Owens Valley, Death Valley, and the Avawtz Mountains.

L.3.2 EXISTING ENVIRONMENT ELSEWHERE IN NEVADA

The following sections describe the environment along each of the five possible rail corridors (Figure L-2) and at the three intermodal transfer station locations (Figure L-3). Table L-1 lists surface-water-related resources along each of the five rail corridors. The corridors are about 0.4 kilometer (0.25 mile) wide, and the length of each corridor varies (Table L-2). Details of each of the corridors and surface-water-related resources are found in TRW (1999b, Appendixes E, F, G, H, and I).

More detail on each of the rail corridors is provided in Chapter 2, Section 2.1.3.3.2, and Chapter 3, Section 3.2.2. Chapter 6, Section 6.3.2, describes the potential impacts of rail implementing alternatives and Chapter 6, Section 6.3.3 describes the potential impacts of the construction and use of intermodal transfer stations under the heavy-haul truck implementing alternatives.

L.3.2.1 Caliente Rail Corridor

Flooding: The Caliente rail corridor crosses 352 washes en route to the Yucca Mountain site (TRW 1999c, pages 3 to 4). Approximately 12 washes along this route are large enough that bridges would be required to cross them. Floodplains associated with these washes have not been defined at this time.

Wetlands: At least four springs or groups of springs and three streams or riparian areas that may have associated wetlands are within 0.4 kilometer (0.25 mile) of the Caliente rail corridor. However, no field

Table L-1. Surface-water-related resources along candidate rail corridors.[a]

Rail corridor	Distance from corridor (kilometers)[b]	Feature
Caliente		
Caliente to Meadow Valley	0.5	Springs – two unnamed springs, in Meadow Valley north of Caliente
	Within	Riparian area/stream – corridor crosses and is adjacent to stream and riparian area in Meadow Valley Wash
Meadow Valley to Sand Spring Valley	1.0	Spring – Bennett Spring, 3.2 kilometers southeast of Bennett Pass
	0.05 - 2.6	Springs – group of five springs (Deadman, Coal, Black Rock, Hamilton, and one unnamed) east of White River
	Within	Riparian/river – corridor parallels (and crosses) the White River for about 25 kilometers. August 1997 survey found river to be mostly underground with ephemeral washes above ground.
	0.8	Spring – McCutchen Spring, north of Worthington Mountains
Sand Spring Valley to Mud Lake	0.02	Spring – Black Spring, south of Warm Springs
Mud Lake to Yucca Mountain	Within - 2.5	Springs – numerous springs and seeps along Amargosa River in Oasis Valley
	Within	Riparian Area – designated area east of Oasis Valley, flowing into Amargosa Valley
	0.3 - 1.3	Springs – group of 13 unnamed springs in Oasis Valley north of Beatty
	Within - 0.3	Riparian area/stream – Amargosa River, with persistent water and extensive wet meadows near springs and seeps
Carlin		
Beowawe to Austin	0.5	Spring – Tub Spring, northeast of Red Mountain
	0.8	Spring – Red Mountain Spring, east of Red Mountain
	0.9	Spring – Summit Spring, west of corridor and south of Red Mountain
	0.4	Spring – Dry Canyon Spring, west of Hot Springs Point
	0.8	Spring – unnamed spring on eastern slope of Toiyabe Range, southwest of Hot Springs Point
	1.0	Riparian area – intermittent riparian area associated with Rosebush Creek, in western Grass Valley, north of Mount Callaghan
	Within	Riparian/creek – corridor crosses Skull Creek, portions of which have been designated riparian areas
	Within	Riparian/creek – corridor crosses intermittent Ox Corral Creek; portions designated as riparian habitat. August, 1997 survey found creek dry with no riparian vegetation present
	0.1	Spring – Rye Patch Spring, at north entrance of Rye Patch Canyon, west of Bates Mountain
	Within	Riparian area – corridor crosses and parallels riparian area in Rye Patch Canyon
	0.7	Spring – Bullrush Spring, east of Rye Patch Canyon
Austin to Mud Lake	0.8	Springs – group of 35 unnamed springs, about 25 kilometers north of Round Mountain on east side of Big Smokey Valley
	0.6	Riparian area – marsh area formed from group of 35 springs
	0.6	Spring – Mustang Spring, south of Seyler Reservoir
	0.3	Riparian/reservoir – Seyler Reservoir, west of Manhattan
Mud Lake to Yucca Mountain		See Caliente corridor
Caliente-Chalk Mountain		
Caliente to Meadow Valley		See Caliente corridor
Meadow Valley to Sand Spring Valley		See Caliente corrid
Sand Spring Valley to Yucca Mountain	1.0	Spring – Reitman's Seep, in eastern Yucca Flat, east of BJ Wye
	0.8	Spring – Cane Spring, on north side of Skull Mountain on Nevada Test Site
Jean		None identified
Valley Modified		None identified

a. Source: TRW (1999b, Appendixes E, F, G, H, and I).
b. To convert kilometers to miles, multiply by 0.62137.

Table L-2. Length of each rail corridor implementing alternative.

Rail corridor	Length
Caliente	513 kilometers (319 miles)
Carlin	520 kilometers (323 miles)
Caliente-Chalk Mountain	345 kilometers (214 miles)
Jean	181 kilometers (112 miles)
Valley Modified	159 kilometers (99 miles)

searches or formal delineations of wetlands have been conducted along this route. Black Spring is near the corridor at the north end of the Kawich Range and an unnamed spring is near the corridor at the north end of the North Pahroc Range. An unnamed spring is 0.3 kilometer (0.2 mile) east of the corridor between Mud Lake and the Yucca Mountain site. A group of springs is in the corridor near the Amargosa River in Oasis Valley. The corridor crosses the Meadow Valley Wash south of Panaca. The corridor also crosses the White River between U.S. Highway 93 and Sand Spring Valley and parallels the river for approximately 26 kilometers (16 miles). That portion of the White River normally is dry. The corridor crosses the Amargosa River in the north end of the Oasis Valley, in an area designated as riparian area by the Bureau of Land Management (TRW 1999b, page 3-23).

Biology: The desert tortoise is the only threatened or endangered species found along the Caliente rail corridor. The southern 50 kilometers (30 miles) of this corridor is within desert tortoise habitat. This area is not designated as critical habitat and the abundance of tortoises in the area is low (TRW 1999b, page 3-23). Three other species (Meadow Valley Wash speckled dace [*Rhinichthys osculus* ssp.], Meadow Valley Wash desert sucker [*Catostomus clarki* ssp.], and Nevada sanddune beardtongue) classified as sensitive by the Bureau of Land Management or as protected by Nevada have been found along the Caliente rail corridor. This rail corridor crosses approximately 14 areas designated as game habitat and one area classified as waterfowl habitat (TRW 1999b, page 3-23). Two of these species, the speckled dace and desert sucker, are restricted to the floodplain of the Meadow Valley Wash. The designated waterfowl habitat also is generally restricted to the floodplain of Meadow Valley Wash and adjacent wetlands.

Archaeology: There are 97 archaeological sites that have been recorded along the Caliente route.

L.3.2.2 Carlin Rail Corridor

Flooding: The Carlin rail corridor crosses 273 washes en route to the Yucca Mountain site (TRW 1999c, pages 3 to 4). Approximately 10 washes along this route are large enough that bridges would be required to cross them. Floodplains associated with these washes have not been defined at this time.

Wetlands: There are at least three springs or groups of springs, six streams designated as riparian areas by the Bureau of Land Management, and one reservoir that may have associated wetlands within 0.4 kilometer (0.25 mile) of the Carlin rail corridor. However, no field searches or formal delineations of wetlands have been conducted along this route. Rye Patch Spring is on the edge of the corridor at the south end of the Simpson Park Mountains, an unnamed spring is 0.3 kilometer (0.2 mile) east of the corridor between Mud Lake and Yucca Mountain, and a group of springs is in the corridor near the Amargosa River in Oasis Valley. Seyler Reservoir is 0.16 kilometer (0.1 mile) from the corridor in the south end of Big Smoky Valley. There are five riparian areas (Skull, Steiner, and Ox Corral creeks, and Water and Rye Patch canyons) along the section of the route between Beowawe and Austin at the south end of Grass Valley. Two of these (Steiner and Ox Corral creeks, both at the south end of Grass Valley) are ephemeral and have little or no riparian vegetation where the route crosses them. The corridor crosses the Amargosa River in the northern Oasis Valley, in an area designated as a riparian area by the Bureau of Land Management (TRW 1999b, pages 3-25 to 3-26).

Biology: The desert tortoise is the only threatened or endangered species found along the Carlin rail corridor. The southern 50 kilometers (30 miles) of this corridor is within desert tortoise habitat. This area is not designated as critical habitat and the abundance of tortoises in the area is low (TRW 1999b, page 3-25). Three other species (ferruginous hawk [*Buteo regalis*], San Antonio pocket gopher [*Thomomys umbrinus curtatus*], and Nevada sand dune beardtongue [*Penstemom arenarius*]) classified as sensitive by the Bureau of Land Management or as protected by the State of Nevada have been found along the Carlin rail corridor. Additionally, the rail corridor crosses approximately 11 areas designated as game habitat by the Bureau of Land Management (TRW 1999b, page 3-25). None of these species or game habitats are restricted to floodplains or areas that may have wetlands.

Archaeology: There are 110 archaeological sites that have been recorded along the Carlin route.

L.3.2.3 Caliente-Chalk Mountain Rail Corridor

Flooding: The Caliente-Chalk Mountain rail corridor crosses 281 washes en route to the Yucca Mountain site (TRW 1999c, pages 3 to 4). Approximately five washes along this route are large enough that bridges would be required to cross them. Floodplains associated with these washes have not been defined at this time.

Wetlands: One spring and two streams that may have associated wetlands occur within 0.4 kilometer (0.25 mile) of the Caliente-Chalk Mountain rail corridor. However, no field searches or formal delineations of wetlands have been conducted along this route. An unnamed spring is near the corridor at the north end of the North Pahroc Range. The corridor crosses Meadow Valley Wash south of Panaca. The corridor crosses the White River between U.S. 93 and Sand Spring Valley and parallels the river for approximately 26 kilometers (16 miles). That portion of the White River normally is dry.

Biology: The desert tortoise is the only threatened or endangered species found along the Caliente-Chalk Mountain rail corridor. The southern 40 kilometers (25 miles) of this corridor is within desert tortoise habitat. This area is not designated as critical habitat and the abundance of tortoises in the area is low (TRW 1999b, page 3-27). Six species (Meadow Valley Wash speckled dace, Meadow Valley Wash desert sucker, Ripley's springparsley [*Cymopterus ripleyi* var. *saniculoides*], largeflower suncup [*Camissonia megalantha*], Beatley's scorpionweed [*Phacelia beatleyae*], and long-legged myotis [*Myotis volans*]) classified as sensitive by the Bureau of Land Management or protected by Nevada have been found in the Caliente-Chalk Mountain rail corridor. This rail corridor crosses approximately eight areas designated as game habitat and one area of waterfowl habitat (TRW 1999b, page 3-27). Two of these sensitive species, the speckled dace and desert sucker, are restricted to the floodplain of the Meadow Valley Wash. The designated waterfowl habitat also is generally restricted to the floodplain of Meadow Valley Wash and adjacent wetlands.

Archaeology: There are 100 archaeological sites that have been recorded along the Caliente-Chalk Mountain route.

L.3.2.4 Jean Rail Corridor

Flooding: The Jean rail corridor crosses 89 washes en route to the Yucca Mountain site (TRW 1999c, pages 3 to 4). Approximately five washes along this route are large enough that bridges would be required to cross them. Floodplains associated with these washes have not been defined at this time.

Wetlands: No springs, perennial streams, or riparian areas that may have associated wetlands have been identified within 0.4 kilometer (0.25 mile) of the Jean rail corridor (TRW 1999b, page 3-29). However, no field searches or formal delineations of wetlands have been conducted along this route.

Biology: The desert tortoise is the only threatened or endangered species found along the Jean rail corridor. This entire corridor is within desert tortoise habitat, but does not cross any areas designated as critical habitat. The abundance of desert tortoises is low along most of the rail corridor, although there is a higher abundance along some portions in Ivanpah, Goodsprings, Mesquite, and Pahrump valleys (TRW 1999b, page 3-28). One species, the pinto beardtongue (*Penstemon bicolor* spp.) that is classified as sensitive by the Bureau of Land Management has been found within the corridor. This rail corridor crosses approximately 12 areas designated as game habitat by the Bureau of Land Management (TRW 1999b, page 3-28). None of these species or game habitats are restricted to floodplains or areas that may have wetlands.

Archaeology: Six archaeological sites have been recorded along the Jean rail corridor.

L.3.2.5 Valley-Modified Rail Corridor

Flooding: The Valley-Modified rail corridor crosses 95 washes en route to the Yucca Mountain site (TRW 1999c, pages 3 to 4). Approximately three washes along this route are large enough that bridges would be required to cross them. Floodplains associated with these washes have not been defined at this time.

Wetlands: No springs, perennial streams, or riparian areas that may have associated wetlands have been identified within 0.4 kilometer (0.25 mile) of the Valley-Modified rail corridor (TRW 1999b, pages 3-29 to 3-30). However, no field searches or formal delineations have been conducted along this route.

Biology: The desert tortoise is the only threatened or endangered species found along the Valley-Modified rail corridor. This entire corridor is within desert tortoise habitat, but does not cross any areas designated as critical habitat. The abundance of desert tortoises is low along this rail corridor (TRW 1999b, page 3-29). Two plant species (Parish's scorpionweed [*Phacelia parishii*] and Ripley's springparsley) classified as sensitive by the Bureau of Land Management have been found in the rail corridor. None of these species are restricted to floodplains or areas that may have wetlands. The Valley-Modified rail corridor does not cross any Bureau of Land Management-designated game habitat (TRW 1999b, page 3-29).

Archaeology: Nineteen archaeological sites have been recorded along the Valley-Modified rail corridor.

L.3.2.6 Caliente Intermodal Transfer Station

Flooding: The two proposed sites for the Caliente intermodal transfer station are located in the Meadow Valley Wash south of Caliente. Both areas are outside the inundation boundary of the 100-year floodplain, but within the boundary of the 500-year floodplain.

Wetlands: Part of the proposed station location is moist during at least some portions of the year and may be classified as wetlands. The adjacent perennial stream and riparian habitat along Meadow Valley Wash also might be classified as wetlands, although no formal delineation of wetlands has been conducted for this proposed activity (TRW 1999b, page 3-35).

Biology: No game habitat, threatened or endangered species, or species classified as sensitive by the Bureau of Land Management or protected by Nevada occur within the proposed station location (TRW 1999b, page 3-35).

Archaeology: Four archaeological sites have been recorded at the Caliente intermodal transfer station site.

L.3.2.7 Apex/Dry Lake Intermodal Transfer Station

Flooding: The two proposed sites for the Apex/Dry Lake intermodal transfer station are located outside of the 100-year and 500-year floodplain.

Wetlands: There are no springs or riparian areas within the proposed station location (TRW 1999b, page 3-36).

Biology: The only resident threatened or endangered species at this site is the desert tortoise. The abundance of desert tortoises in Dry Lake Valley generally is low, although some areas there have a higher abundance. One plant species, Geyer's milkvetch (*Astragalus geyeri triquetrus*), classified as sensitive by the Bureau of Land Management has been found in the proposed location. Neither of these species are restricted to floodplains or wetlands. No game habitat has been designated there (TRW 1999b, page 3-36).

Archaeology: Two archaeological sites have been recorded at the Apex/Dry Lake intermodal transfer station site.

L.3.2.8 Sloan/Jean Intermodal Transfer Station

Flooding: The southernmost proposed site for the Jean intermodal transfer station is located in the same general area as a 100-year flood inundation zone. The northern site proposed for the Jean intermodal transfer station is not in an inundation zone and is outside the 500-year floodplain. The northernmost proposed site for the Sloan intermodal transfer station is in an area with no printed Federal Emergency Management Agency map and it is outside the 500-year floodplain.

Wetlands: There are no springs or riparian areas within the proposed station location (TRW 1999b, page 3-36).

Biology: The only resident threatened or endangered species at this site is the desert tortoise. The abundance of desert tortoises in Ivanpah Valley generally is moderate to high, relative to other areas within the range of this species in Nevada. One plant species, pinto beardtongue, classified as sensitive by the Bureau of Land Management has been found in the proposed location. Neither of these species are restricted to floodplains or wetlands. No game habitat has been designated there (TRW 1999b, pages 3-36 to 3-37).

Archaeology: Seven archaeological sites have been recorded at the Sloan/Jean intermodal transfer station site.

L.4 Floodplain/Wetlands Effects

According to 10 CFR 1022.12(a)(2), a floodplain assessment is required to discuss the positive and negative, direct and indirect, and long- and short-term effects of the proposed action on the floodplain and/or wetlands. In addition, the effects on lives and property, and on natural and beneficial values of floodplains must be evaluated. For actions taken in wetlands, the assessment should evaluate the effects of the proposed action on the survival, quality, and natural and beneficial values of the wetlands. If DOE finds no practicable alternative to locating activities in floodplains or wetlands, DOE will design or modify its actions to minimize potential harm to or in the floodplains and wetlands. The floodplains that are assessed herein are those areas of normally dry washes that are temporarily and infrequently inundated from runoff during 100-year or 500-year floods.

L.4.1 FLOODPLAIN/WETLANDS EFFECTS NEAR YUCCA MOUNTAIN

DOE has not determined if rail casks will be transported in Nevada by heavy-haul trucks on existing highways or whether to construct a branch rail line to bring the spent nuclear fuel and high-level radioactive waste to the Yucca Mountain site. Near Yucca Mountain, however, it is possible that each of the four washes could be affected if a rail line and a road were to access the Yucca Mountain site from different directions. Because of this uncertainty, this assessment examines the configurations that would cause the most disturbances to the four washes and their floodplains, as follows:

- Potential construction of a heavy-haul-capable road west of Fortymile Wash that crosses Busted Butte Wash, Drillhole Wash, and Midway Valley Wash. Cut, fill, and drainage culverts could be used to cross Busted Butte and Drillhole washes. A bridge could be constructed over Midway Valley Wash. Heavy-haul trucks carrying spent nuclear fuel and high-level radioactive waste could travel along this road to the repository.

- Potential construction of a raised rail line through Fortymile Wash with appropriately-sized drainage culverts. The rail line could join the route for heavy-haul trucks north of Drillhole Wash and cross Midway Valley Wash on a separate rail-bridge before entering the repository. Trains carrying spent nuclear fuel and high-level radioactive waste could travel along the rail line to the repository.

- Potential upgrading of the existing road that crosses Fortymile Wash with appropriately-sized drainage culverts. The road could be used by legal-weight trucks to transport spent nuclear fuel and high-level radioactive waste to the repository, as well as transporting various types of hazardous and non-hazardous materials to and from the repository.

Construction in the washes would reduce the area through which floodwaters naturally flow. During large floods, bodies of water could develop on the upstream side of each of the crossings and slowly drain through culverts. Such floods, however, would not increase the risk of future flood damage, increase the impact of floods on human health and safety, or harm the natural and beneficial values of the floodplains because there are no human activities or facilities upstream or downstream that could be affected. A sufficiently large flood in Fortymile Wash could create a temporary large lake up-stream of the raised rail line and the legal-weight road. The water would slowly drain through culverts. If the flood occurred quickly and was sufficiently large, water would flow over the rail line and roads and continue downstream. Some damage to the rail line and the roads would be expected, but neither structure would increase the risk of future flood damage, increase the impact of floods on human health and safety, or harm the natural and beneficial values of the floodplains because there are no human activities or facilities downstream that could be affected.

During and after each flood, a large amount of sediment would accumulate on the up-stream side of each crossing. Periodically, this material would have to be removed so that future floods would have sufficient space to accumulate, rather than overflow the structures during successively smaller floods. This material would, when deemed necessary, be removed by truck and disposed of appropriately. Under natural conditions this sediment would have continued downstream and been deposited as the floodwaters receded. Compared to the total amount of sediment that is moved by the flood water along the entire length of the washes, the amount trapped behind the crossings would be small.

During a 100-year or 500-year flood, there would be no preferred channels; all channels across the entire width of each wash would be filled with water (Figure L-1). Therefore, the manmade crossings would not cause preferential flow in a particular channel or alter the velocity or direction of flow on the floodplains.

Potential construction of a route for heavy-haul trucks or rail line would require the removal of desert vegetation in the washes and the disturbance of soil and alluvium. These actions could adversely impact wildlife habitat and individuals, especially the desert tortoise, which is designated as threatened by the Fish and Wildlife Service. Prior to any construction, a biological survey would be conducted to locate and remove tortoises that are in the path of construction and other mitigation measures would be conducted as identified by the Fish and Wildlife Service during consultations under the Endangered Species Act for this action.

Construction in the floodplains could also affect unidentified cultural resources that may be present. Prior to any construction, archaeologists would survey the area following the procedure in DOE's Programmatic Agreement with the Advisory Council on Historic Preservation (DOE 1988, page 5).

Potential indirect impacts on flora and fauna include increased emissions of fugitive dust, elevated noise levels, and increased human activities. Emissions of fugitive dust would be short-term and would not be expected to significantly affect vegetation or wildlife. Likewise, no significant long-term impacts to wildlife are expected from the temporary increase in noise during construction. Wildlife displaced during construction would probably return after construction was completed.

There are no perennial sources of surface water at or downstream from the Yucca Mountain site that would be affected by the use of a route for heavy-haul trucks or the construction of a rail line. Two small well ponds with some riparian vegetation occur in Fortymile Wash downstream of the point where Drillhole Wash enters Fortymile Wash. During a 100- or 500-year flood, both riparian areas would likely be damaged or destroyed by floodwaters regardless of the existence of the crossings.

Neither the quality nor the quantity of groundwater that normally recharges through Fortymile Wash would be substantially affected due to the crossings. Water infiltration could increase somewhat after large floods as standing water slowly enters the ground behind the crossings. The total volume of these water bodies would be a few acre-feet at most, and much of the water would gradually drain through culverts or evaporate before reaching the groundwater table at 274 meters (900 feet) below the surface.

The use of petroleum, oil, lubricants, and other hazardous materials during construction would be strictly controlled and spills would be promptly cleaned up and, if needed, the soil and alluvium would be remediated. The small amount of these materials that might enter the ground would not affect the groundwater, which is 274 meters (900 feet) below the surface.

The nearest population center is about 19 kilometers (12 miles) to the south, along U.S. 95 at Lathrop Wells a few miles east of Fortymile Wash. If floodwaters from a 100- or 500-year flood reached this far downstream, there would be no measurable increase in flood velocity or sediment load attributable to the use of a route for heavy-haul trucks or construction of a rail line compared to natural conditions. Hence, disturbances to the floodplains of Fortymile Wash, Busted Butte Wash, Drillhole Wash, or Midway Valley Wash would have no adverse impacts on lives and property downstream. Moreover, impacts to these floodplains would be insignificant in both the short- and long-term compared to the erosion and deposition that occur naturally and erratically in these desert washes and floodplains.

During operation of the repository it would be extremely unlikely that a truck carrying spent nuclear fuel and high-level radioactive waste would fall into Busted Butte, Drillhole, or Midway Valley washes or that a train would derail in Fortymile Wash. However, even if this occurred, the shipping casks, which are designed to prevent the release of radioactive materials during an accident, would remain intact. The casks would then be recovered and transported to the repository. No adverse impacts to surface water or groundwater quality from such accidents would occur.

Hazardous materials needed during construction and operation of the repository would be transported along the legal-weight access road. If these materials were released during an accident, they would be cleaned-up quickly and the affected soil and alluvium would be remediated. No adverse impacts to groundwater quality from such accidents would occur because cleanup could be completed before contaminants reached the groundwater [the groundwater table is 274 meters (900 feet) below the surface].

There are no positive or beneficial impacts to the floodplains of Busted Butte, Drillhole, Midway Valley, or Fortymile washes that have been identified from the proposed action.

L.4.2 FLOODPLAIN/WETLANDS EFFECTS ELSEWHERE IN NEVADA

L.4.2.1 Effects along Rail Corridors

Potential rail routes would cross many small, and some large, washes. In general, the impacts caused by rail construction in any of these washes and their floodplains would be similar in magnitude to those described for Fortymile, Busted Butte, Drillhole, and Midway Valley washes. Regardless of the route selected, standard mitigation practices used throughout Nevada for highway construction would be used to minimize the impacts to floodplains. Most washes and their floodplains along the five potential rail corridors are in remote areas. Impacts to these floodplains from rail construction and operation would be insignificant in both the short- and long-term compared to erosion and deposition that occurs naturally and erratically in these desert washes and floodplains.

Based on current information, springs and riparian areas that may have associated wetlands occur within three of the rail corridors (Caliente, Carlin, and Caliente-Chalk Mountain). If the rail mode of spent nuclear fuel and high-level radioactive waste transport is selected by DOE, wetlands delineations along the selected route would be conducted and the effects would be described in a more detailed floodplain/wetlands assessment for public review.

L.4.2.2 Effects at Intermodal Transfer Stations

Neither the Dry Lake intermodal transfer station nor the Sloan/Jean intermodal transfer station would have any impacts on floodplains because these station locations are not in a floodplain. The Caliente intermodal transfer station, however, is located in Meadow Valley Wash, separated by the Union Pacific Railroad. If this site were selected, DOE would conduct a more detailed floodplain/wetlands assessment for public review to address the floodplain/wetlands effects at the Caliente intermodal transfer station location. The more detailed floodplain/wetlands assessment would also include potential upgrades to existing roads for heavy-haul use.

L.5 Mitigation Measures

According to 10 CFR 1022.12(a) (3), agencies must address measures to mitigate the adverse impacts of actions in a floodplain or wetlands, including but not limited to minimum grading requirements, runoff controls, design and construction constraints, and protection of ecologically-sensitive areas. Whenever possible, DOE would avoid disturbing wetlands and floodplains and would minimize impacts to the extent practicable, if avoidance was not possible. This section discusses the floodplain mitigation measures that would be considered in the vicinity of Yucca Mountain and elsewhere in Nevada and, where necessary and feasible, implemented during construction and maintenance in the washes.

Adverse impacts to the affected floodplains would be small. Even during 100- and 500-year floods, it is unlikely that differences in the rate and distribution of erosion and sedimentation caused by the use of a route for heavy-haul trucks or construction of a rail line near Yucca Mountain would be measurably different compared to existing conditions. Nevertheless, DOE would follow their reclamation guidelines (DOE 1995, pages 2-1 to 2-14) for site clearance, topsoil salvage, erosion and runoff control, recontouring, revegetation, siting of roads, construction practices, and site maintenance. Disturbance of surface areas and vegetation would be minimized, and natural contours would be maintained to the maximum extent feasible. Slopes would be stabilized to minimize erosion. Unnecessary off-road vehicle travel would be avoided. Storage of hazardous materials during construction would be outside the floodplains.

Before any potential construction could begin, DOE would require pre-construction surveys to make sure that the work would not impact important biological or archaeological resources. In addition, the site's reclamation potential would be determined during these surveys. In the event that construction could threaten important biological or archaeological resources, and modification or relocation of the roads and rail line is not reasonable, mitigation measures would be developed. Mitigation measures developed during the pre-construction surveys would be incorporated into the design of the work. These measures could include relocation of sensitive species, avoidance of archaeological sites, or data recovery if avoidance is not feasible.

If hazardous materials are spilled during construction of the crossings or during transport to the repository, the spill would be quickly cleaned-up and the soil and alluvium would be remediated. Hazardous materials would be stored away from all floodplains to decrease the probability of an inadvertent spill in these areas.

L.6 Alternatives

According to 1022.12(a)(3), DOE must consider alternatives to the proposed action. Alternative ways to access the Yucca Mountain site are considered in the following paragraphs, along with the no action alternative.

L.6.1 ALTERNATIVES NEAR YUCCA MOUNTAIN

To operate a potential repository at Yucca Mountain, heavy-haul-capable and legal-weight roads and a rail line to the facility would be considered so the spent nuclear fuel and high-level radioactive waste could be unloaded and emplaced underground. It is unreasonable to consider a railroad or heavy-haul-capable and legal-weight roads that access the repository directly from the west over Yucca Mountain because of engineering constraints, environmental damage, and cost associated with construction in such rugged terrain. Because of these concerns, this alternative was eliminated from detailed consideration.

Access to Yucca Mountain from the east side requires that Fortymile Wash be crossed. Alternative sites for these crossings were considered, but the impacts at any alternative site would be virtually identical to the proposed site. Moreover, the proposed sites provide the most direct routes to the repository and would cost less to build and/or upgrade than alternative sites that cross Fortymile Wash at wider locations.

L.6.2 ALTERNATIVE RAIL CORRIDORS AND ALTERNATIVE SITES FOR AN INTERMODAL TRANSFER STATION

Five potential rail corridors were identified by DOE through a winnowing process that considered a host of environmental constraints (see Chapter 2, Section 2.3.3). Other possible rail corridors in Nevada were examined but rejected because of such things as land use, private land, and engineering constraints. Identification of the three intermodal transfer station locations was limited to reasonable sites next to an existing rail line in Nevada. Other sites were considered by DOE, but rejected because of ownership and environmental concerns.

L.6.3 NO-ACTION ALTERNATIVE

Selection of the No-Action Alternative would avoid impacts to floodplains and wetlands. If Yucca Mountain was selected as a site to construct a repository, transport of spent nuclear fuel and high-level radioactive waste to the Yucca Mountain site would be required. In that case there would be no other practicable alternative to taking action in floodplains and wetlands because there would be no way to transport spent nuclear fuel and high-level radioactive waste to the Yucca Mountain site during repository operation without passing through some wetlands areas and floodplains.

L.7 Conclusions

DOE prepared this assessment in compliance with 10 CFR Part 1022. The assessment evaluates the effects to the floodplains near Yucca Mountain (Fortymile Wash, Busted Butte Wash, Drillhole Wash, and Midway Valley Wash) and generically to floodplains and wetlands elsewhere in Nevada from construction of a rail line or an intermodal transfer station and associated upgrades to existing highways for heavy-haul trucks.

Near Yucca Mountain, the closest man-made structure within Fortymile Wash is U.S. 95 more than 19 kilometers (12 miles) south of the confluence of Drillhole and Fortymile washes. Lathrop Wells, the nearest population center to Yucca Mountain, is also about 19 kilometers to the south along U.S. 95 and two miles east of Fortymile Wash. Construction- and operations-related impacts to the 100-year and 500-year floodplains of Fortymile Wash, Busted Butte Wash, Drillhole Wash, and Midway Valley Wash would be small. None of these impacts would increase the risk of future flood damage, or increase the impact of floods on human health and safety, or harm the natural and beneficial values of the floodplains. There are no positive or beneficial impacts to the floodplains of Busted Butte, Drillhole, Midway Valley, or Fortymile washes from the proposed actions that have been identified.

Elsewhere in Nevada, effects to floodplains and wetlands would probably be small, although a detailed floodplain/wetlands assessment would be conducted by DOE when more information is available upon selection of a rail corridor or route for heavy-haul trucks.

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