Final

Supplemental Environmental Impact Statement

for a

Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada

Volume II
Appendices A through J

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

DOE/EIS-0250F-S1

June 2008

ACRONYMS AND ABBREVIATIONS

To ensure a more reader-friendly document, the U.S. Department of Energy (DOE) limited the use of acronyms and abbreviations in this Repository supplemental environmental impact statement. In addition, acronyms and abbreviations are defined the first time they are used in each chapter or appendix. The acronyms and abbreviations used in the text of this document are listed below. Acronyms and abbreviations used in tables and figures because of space limitations are listed in footnotes to the tables and figures.

°C	degrees Celsius
CFR	Code of Federal Regulations
dBA	A-weighted decibels
DOE	U.S. Department of Energy (also called *the Department*)
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
°F	degrees Fahrenheit
FEIS	final environmental impact statement
FR	*Federal Register*
GNEP	Global Nuclear Energy Partnership
MTHM	metric tons of heavy metal
NEPA	*National Environmental Policy Act*
NRC	U.S. Nuclear Regulatory Commission
NWPA	*Nuclear Waste Policy Act*, as amended
PM_{10}	particulate matter with an aerodynamic diameter of 10 micrometers or less
$PM_{2.5}$	particulate matter with an aerodynamic diameter of 2.5 micrometers or less
REMI	Regional Economic Models, Inc.
RMEI	reasonably maximally exposed individual
SEIS	supplemental environmental impact statement
Stat.	United States Statutes
TAD	transportation, aging, and disposal (canister)
TSPA	Total System Performance Assessment
U.S.C.	United States Code
VdB	vibration velocity in decibels with respect to 1 micro-inch per second

TERMS AND DEFINITIONS

In this Repository SEIS, DOE has italicized terms that appear in the Glossary (Chapter 12) the first time they appear in a chapter.

UNDERSTANDING SCIENTIFIC NOTATION

DOE has used scientific notation in this Repository SEIS to express numbers that are so large or so small that they can be difficult to read or write. Scientific notation is based on the use of positive and negative powers of 10. The number written in scientific notation is expressed as the product of a number between 1 and 10 and a positive or negative power of 10. Examples include the following:

Positive Powers of 10	Negative Powers of 10
$10^1 = 10 \times 1 = 10$	$10^{-1} = 1/10 = 0.1$
$10^2 = 10 \times 10 = 100$	$10^{-2} = 1/100 = 0.01$
and so on, therefore,	and so on, therefore,
10^6 = 1,000,000 (or 1 million)	10^{-6} = 0.000001 (or 1 in 1 million)

Probability is expressed as a number between 0 and 1 (0 to 100 percent likelihood of the occurrence of an event). The notation 3×10^{-6} can be read 0.000003, which means that there are 3 chances in 1 million that the associated result (for example, a fatal cancer) will occur in the period covered by the analysis.

Substantive changes in this document are indicated in the margins with change bars.

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COVER SHEET

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ABSTRACT: DOE's Proposed Action is to construct, operate, monitor, and eventually close a geologic repository at Yucca Mountain for the disposal of spent nuclear fuel and high-level radioactive waste. Under the Proposed Action, spent nuclear fuel and high-level radioactive waste in storage or projected to be generated at 72 commercial and 4 DOE sites would be shipped to the repository by rail (train), although some shipments would arrive at the repository by truck. The Repository SEIS evaluates (1) the potential environmental impacts from the construction, operations, monitoring, and eventual closure of the repository; (2) potential long-term impacts from the disposal of spent nuclear fuel and high-level radioactive waste; (3) potential impacts of transporting these materials nationally and in the State of Nevada; and (4) potential impacts of not proceeding with the Proposed Action (the No-Action Alternative).

COOPERATING AGENCIES: Nye County, Nevada, is a cooperating agency in the preparation of the Repository SEIS.

PUBLIC COMMENTS: In preparing this Repository SEIS, DOE considered written comments received by letter, electronic mail, and facsimile transmission, and oral and written comments given at public hearings at six locations in Nevada, one location in California, and in Washington, DC.

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A. OPTIONS TO ELEMENTS OF THE PROPOSED ACTION

The U.S. Department of Energy (DOE or the Department) has added this new appendix since it completed the *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F; DIRS 155970-DOE 2002, all) (Yucca Mountain FEIS). The appendix describes options to elements of the Proposed Action presented in Chapter 2, Section 2.1 of this *Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F-S1) (Repository SEIS). It evaluates these options in terms of how the potential environmental impacts would differ from what the DOE presents in Chapter 4 of this Repository SEIS.

The options discussed in this appendix include:

- Wastewater treatment at the repository;
- Reduced transportation, aging, and disposal (TAD) canister use;
- National rail routes;
- Workforce residency;
- Extended monitoring analytical period; and
- Highway routing.

This appendix provides insight to the extent potential impacts would be sensitive to modifications to the Proposed Action; for example, what is the situation if only 75 percent of commercial spent nuclear fuel could be placed in TAD canisters at commercial sites, with the remainder being loaded into TAD canisters at the repository.

A.1 Wastewater Treatment at the Repository Option

Chapter 2, Section 2.1.2.4.3, of this Repository SEIS acknowledges that under the Proposed Action, utility design does not specifically include a wastewater treatment facility; DOE could, however, develop one in the future to allow the reuse and disposal of treated waste water. The current repository design includes septic tanks and leach fields for the treatment of sanitary sewage. A wastewater treatment facility would provide more options for industrial and sanitary wastewater, which would include the potential for reuse and recycling of the treated water. The following sections address the potential benefits and environmental impacts from a wastewater treatment facility.

If DOE implemented this option, it would use a premanufactured wastewater treatment facility. Such facilities are readily available and are in common use in small municipalities and on individual properties. A typical premanufactured wastewater treatment facility includes equipment for screening grit and solids, a compartment or tank for flow equalization, equipment and a tank for aeration to facilitate biological treatment of the main flow, clarification equipment, tanks for digestion of sludge separated from the main flow, and effluent disinfection (generally chlorination) equipment. Systems typically arrive as ready-to-connect modular components.

Nevada permits premanufactured wastewater treatment facilities with a minimum design flow of 5,000 gallons per day (Nevada Revised Statutes 445A.540). The facility must meet secondary treatment standards (DIRS 182842-NDEP n.d., all). If wastewater reuse became the option for effluent disposal, a state groundwater discharge permit would be necessary for nonsurface-water discharges. DOE would dispose of wastewater discharge in excess of reuse needs to the surface by either a rapid infiltration pond or a leach field at the proposed repository.

A.1.1 POTENTIAL BENEFITS OF THE PREMANUFACTURED WASTEWATER TREATMENT FACILITY

A premanufactured wastewater treatment facility would enable wastewater reuse that the proposed septic systems would not offer. DOE could use the treated wastewater for dust suppression, landscaping, or other uses, thereby reducing the burden on the current once-through use of groundwater resources. For example, estimates of water demand for the Proposed Action (DIRS 181232-Fitzpatrick-Maul 2007, all) include a designation of up to about 25,000 cubic meters (20 acre-feet) of water per year for activities such as dust suppression. Treated wastewater could supplement a portion or possibly all of this demand. The flexible design of the facilities would enable the installation of additional modules to treat increases in wastewater volume. A treatment facility would offer the flexibility to accept industrial wastewater in addition to sanitary sewage.

A.1.2 POTENTIAL ENVIRONMENTAL IMPACTS OF THE PREMANUFACTURED WASTEWATER TREATMENT FACILITY

A premanufactured wastewater treatment facility would disturb no more land than the currently proposed septic tanks and leach fields. It would not affect air quality, biological resources, cultural resources, aesthetics, or noise. It would not affect surface- or groundwater resources differently than the currently proposed septic systems. There could be a positive impact through the treatment and reuse of water for activities such as dust suppression and landscaping. While there could be one or two additional employees involved with a wastewater treatment facility, there would be no additional socioeconomic impacts. There would be no additional environmental impacts from the selection of a wastewater treatment facility over the currently proposed septic systems.

A premanufactured facility would require an initial outlay of capital that could be greater than that for construction of a conventional, large-capacity septic system. In addition, a wastewater treatment facility would entail a higher level of regulatory compliance and monitoring than a conventional septic system, such as National Pollutant Discharge Elimination System permitting and monitoring, and increased monitoring of treated wastewater intended for reuse.

A.2 Reduced Transportation, Aging, and Disposal Canister Use Option

DOE's goal under the Proposed Action (Chapter 2, Section 2.1.1) is the packaging of 90 percent of commercial spent nuclear fuel in TAD canisters at commercial sites. However, the sensitivity analysis in this appendix considers the potential case that only 75 percent of commercial spent nuclear fuel could be placed in TAD canisters at commercial sites, with the remainder placed in TAD canisters at the repository.

This Repository SEIS evaluates the potential environmental impacts of shipping nominally 90 percent [56,700 metric tons of heavy metal (MTHM)] of commercial spent nuclear fuel in TAD canisters. During the SEIS public scoping process, DOE received comments from the nuclear industry and others about receipt of less than 90 percent of the commercial spent nuclear fuel in TAD canisters. The following sections evaluate the difference in potential impacts if only 75 percent (47,250 MTHM) of the commercial spent nuclear fuel was shipped in TAD canisters and the remainder either in dual-purpose canisters or as uncanistered fuel. DOE would load uncanistered fuel and fuel that arrived at the repository site in dual-purpose canisters into TAD canisters in the Wet Handling Facility.

This analysis evaluated the effects on transportation impacts and the estimated impacts at the repository. Differences in transportation impacts could result from differences in the number of transportation casks shipped. Consistent with the discussion in Chapter 6 of this Repository SEIS, the transportation impacts would be associated with occupational and public health and safety. Differences in the impacts at the repository could result from the replacement of the third Canister Receipt and Closure Facility with a second Wet Handling Facility.

A.2.1 TRANSPORTATION IMPACTS

Table A-1 lists the amount of commercial spent nuclear fuel and the estimated number of transportation casks that DOE would transport and receive at the proposed repository for the nominal 90-percent case and the 75-percent case. In the 90-percent case, 88 percent of the commercial spent nuclear fuel would be shipped in rail casks containing TAD canisters, 5 percent would be shipped in rail casks containing dual-purpose canisters, and 7 percent would be shipped uncanistered in truck casks. These percentages are based on MTHM, not on the number of casks.

Table A-1. Comparison of commercial spent nuclear fuel transportation using 90-percent and 75-percent implementation of TAD canisters.

Transportation mode	Metric tons of heavy metal		Number of casks	
	90-percent case	75-percent case	90-percent case	75-percent case
TAD canister in rail cask	88.2	75.0	6,499	5,526
Dual-purpose canister in rail cask	4.8	4.8	307	310
Uncanistered spent nuclear fuel in rail cask	0.0	13.1	0	1,123
Uncanistered spent nuclear fuel in truck cask	7.0	7.1	2,650	2,666

Source: DIRS 181377-BSC 2007, all.
TAD = Transportation, aging, and disposal (canister).

In the 75-percent case, the amount of commercial spent nuclear fuel shipped uncanistered in truck casks and dual-purpose canisters in rail casks was held constant. The amount of commercial spent nuclear fuel shipped in rail casks containing TAD canisters was reduced from 88 percent to 75 percent. DOE assumed that the remaining 13 percent of commercial spent nuclear fuel would be shipped uncanistered in rail casks. As with the 90-percent case, these percentages are based on MTHM, not on the number of casks. Table A-4 of *Calculation of Transportation Data for SEIS Analyses* (DIRS 181377-BSC 2007, all) lists transportation cask fleet assumptions.

For both the 90- and 75-percent cases, DOE estimated that there would be about 8 transportation-related fatalities. These fatalities would include latent cancer fatalities, fatalities from exposure to vehicle emissions, and traffic fatalities. Therefore, DOE concluded that a deviation in the percentage of implementation of TAD canisters at the reactor sites would not measurably affect the transportation impacts.

A.2.2 REPOSITORY IMPACTS

Under the 90-percent case, 10 percent (6,300 MTHM) of the commercial spent nuclear fuel would require handling in the Wet Handling Facility. Under the 75-percent case, 25 percent (15,750 MTHM) of the commercial spent nuclear fuel would require handling in the Wet Handling Facility. This is an increase of 2.5 times the baseline case evaluated in Chapter 4 of this Repository SEIS. The fuel would not be packaged in TAD canisters at the generator sites, but instead would be packaged at the repository. Long-term impacts and repository performance would not change.

To accommodate the increased handling of bare commercial spent nuclear fuel, the Department would construct an additional Wet Handling Facility rather than a third Canister Receipt and Closure Facility in the geologic repository operations area. Because this would not result in an overall addition of a facility, there would be no additional impacts to land use, air quality, biological and cultural resources, socioeconomics, noise, aesthetics, and utilities, energy, and materials.

Although the additional Wet Handling Facility would include a spent fuel pool for the underwater handling of fuel, the additional impacts to the estimated annual water demand would be minimal because DOE would closely monitor this pool, once filled, and would continually filter and maintain the water. The additional water demand from the new facility would be somewhat offset by the reduction in the number of Canister Receipt and Closure Facilities.

The additional spent fuel pool in the Wet Handling Facility would affect the management of repository-generated waste. DOE would treat the spent resins used to filter the pool, and the incremental increase in low-level radioactive waste from this source would be somewhat offset by the reduction in the number of Canister Receipt and Closure Facilities. Approximately 580 cubic meters (20,500 cubic feet) of low-level radioactive waste (including both solids and liquids before treatment) would be generated each year from a Wet Handling Facility in comparison with about 76 cubic meters (2,700 cubic feet) of low-level radioactive waste (including both solids and liquids before treatment) from a Canister Receipt and Closure Facility (DIRS 182319-Morton 2007, all).

Radiological impacts to workers would result primarily from external radiation from activities associated with the receipt, handling, aging, and emplacement of spent nuclear fuel and high-level radioactive waste. The reduction in the number of Canister Receipt and Closure Facilities would offset the external radiation impacts to workers from the additional Wet Handling Facility. The additional airborne release of manmade radionuclides would make virtually no contribution to the overall doses the repository workforce received.

Occupational and public health and safety would be the resource area most affected by the additional Wet Handling Facility. Airborne releases of manmade radionuclides during normal operations would occur only from the Wet Handling Facility. With two of these facilities to handle an increased (by 150 percent) inventory of commercial spent nuclear fuel, the releases of manmade radionuclides to the environment would also increase by 150 percent. Naturally occurring radon would account for more than 99.8 percent of the radiological impacts to the offsite public (Chapter 4, Section 4.1.7). The remainder (less than 1 percent) would be attributable to releases from the Wet Handling Facility. Therefore, an increase of 150 percent in these releases would have no measurable effect on the offsite public.

Consequences from accidents associated with the additional Wet Handling Facility would be the same as those identified in Chapter 4, Section 4.1.8, of this Repository SEIS for the original facility. The only effect of the additional facility would be an increase in the overall probability of the identified accidents because the number of activities (for example, crane lifts and fuel handling) would be greater. On the other hand, the number of associated activities that resulted in accidents in the Canister Receipt and Closure Facilities would decrease.

In summary, this analysis illustrated that the deviations in the percentage implementation of TAD canisters would have little effect on transportation or repository-related estimated impacts.

A.3 National Rail Route Option

DOE used the TRAGIS computer program to generate the representative rail routes it used to estimate the transportation impacts in Chapter 6 and Appendix G of this Repository SEIS. These rail routes are called unconstrained because minimal constraints, or blocks, were not placed in the rail network. DOE based its identification of the representative national rail routes on historic railroad industry routing practices. The Department identified these routes by giving priority to the use of rail lines that have the most rail traffic, which are the best maintained and have the highest quality track; giving priority to originating railroads; minimizing the number of interchanges between railroads; and reducing the distance traveled.

Because DOE has not determined the specific rail routes it would use for the transportation of spent nuclear fuel and high-level radioactive waste to the repository and the routes would probably not be the same as the representative routes identified by the TRAGIS program, this section provides a perspective on the sensitivity of the analysis to changes in the routing from the generator sites to the proposed repository. In addition, this analysis responds to the State of Nevada public scoping comment that “heavy traffic congestion along northern cross-country rail corridors will very likely make the southern routing option attractive.”

The purpose of this analysis was to evaluate the effects on the national transportation impacts if the TRAGIS computer program included constraints in the rail network that would lead to other ways the railroads might route shipments. Based on preliminary discussions DOE has had with representatives of the railroad industry, stakeholder groups, and other interested parties, the routing modifications that were represented by constraints in the rail network are:

- A constraint on routing of spent nuclear fuel and high-level radioactive waste through long tunnels, such as the Moffat Tunnel west of Denver and the Flathead Tunnel in Montana;

- A constraint on use of the high-traffic Union Pacific rail line between North Platte and Gibbon Junction, Nebraska. This rail line currently handles about 130 trains per day and the presence of trains that contained spent nuclear fuel and high-level radioactive waste traveling at a maximum speed of 80 kilometers (50 miles) per hour would have the potential to disrupt railroad operations;

- A constraint on avoidance of major rail traffic congestion areas, such as the Chicago rail yards.

This section contains national-level maps of the constrained routes and national-level impact estimates. As with the unconstrained routes, DOE used the TRAGIS program to generate these rail routes. Figures A-1 and A-2 show the constrained routes from each generator site to the repository using the

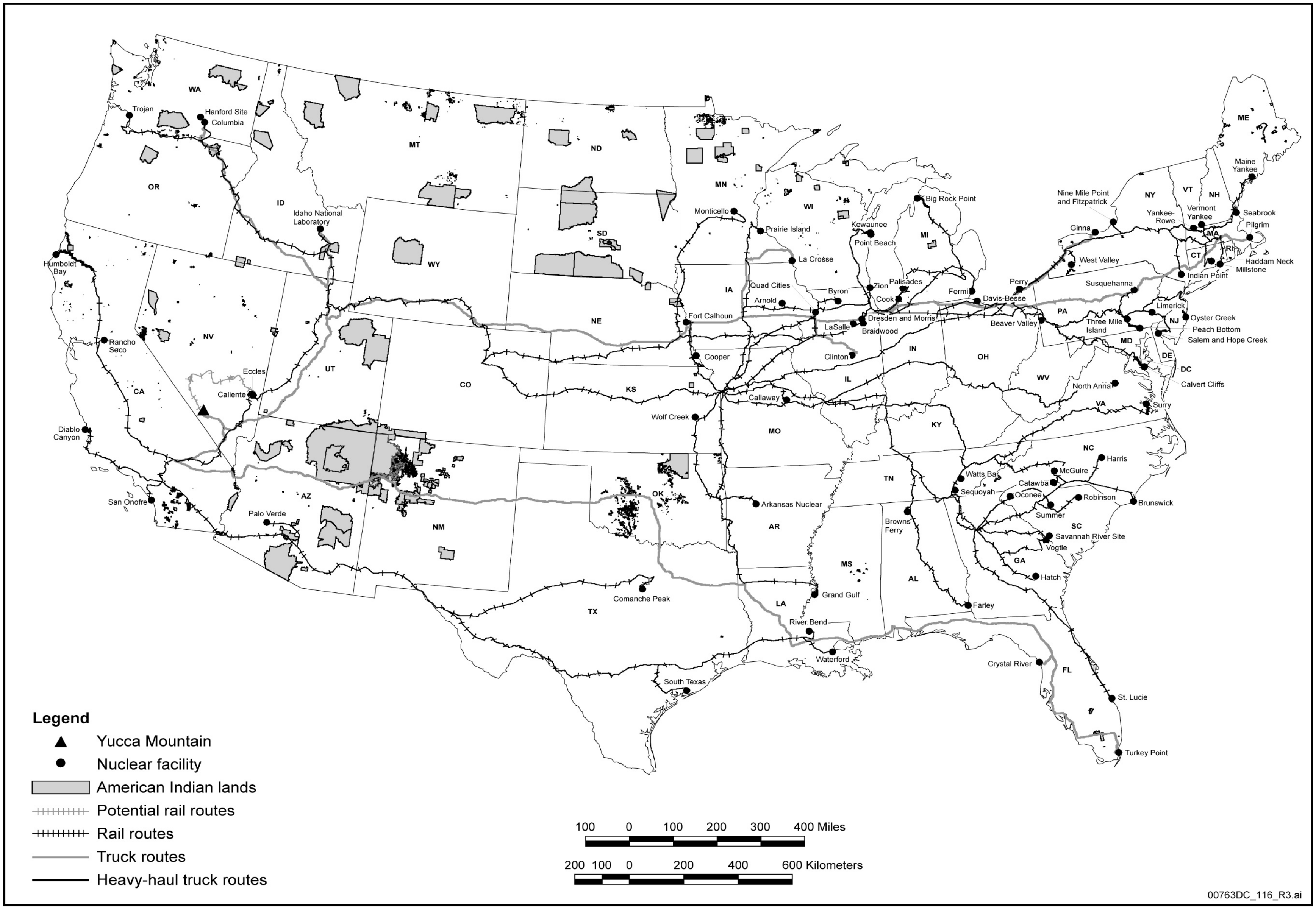

Figure A-1. Representative rail and truck transportation constrained routes if DOE used the Caliente rail corridor.

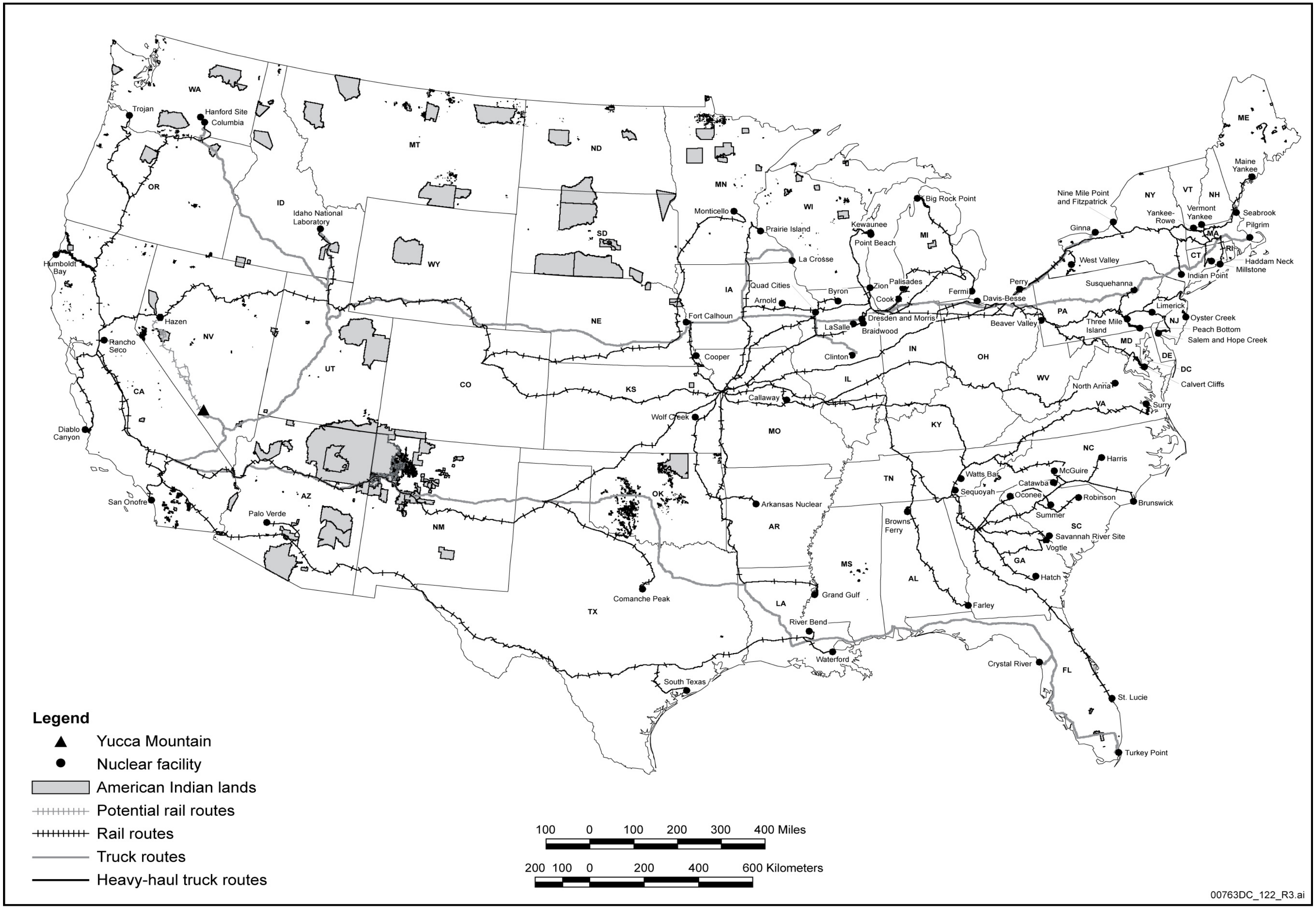

Figure A-2. Representative rail and truck transportation constrained routes if DOE used the Mina rail corridor.

Caliente and Mina rail corridors, respectively. For both the unconstrained and constrained cases on the national level, DOE estimated that there would be a total of about 8 transportation-related fatalities. These would include latent cancer fatalities, fatalities from exposure to vehicle emissions, and traffic fatalities. DOE estimated that there would be 1 to 2 fatalities in Nevada for both the unconstrained and constrained cases. Therefore, DOE concluded that the use of constrained routing would not measurably affect transportation impacts.

A.4 Workforce Residency Option

This Repository SEIS evaluates socioeconomic impacts in Chapter 4, Section 4.1.6 and assumes that 80 percent of the onsite Yucca Mountain Repository workers would reside in Clark County (Las Vegas). DOE based this assumption on historical data, which is consistent with the assumption it made for the analysis in the Yucca Mountain FEIS.

During the public scoping process for this Repository SEIS, DOE received comments from Nye County that requested evaluations of a higher percentage of the workforce residing in that county. For this analysis, this section provides an estimate of the potential socioeconomic impacts if 80 percent of the workforce assigned to the repository site, but none of the workforce assigned to offsite locations, resided in Nye County. While DOE did not base this percentage on historical precedent as it did in Chapter 4, Section 4.1.6, the analysis provides a perspective of the range of socioeconomic impacts that could occur. Uncertainties are becoming inherent in the historical patterns, given that certain factors that affect the current situation could affect future changes in ways different from those evaluated in the past. These factors include the increase in housing costs in Las Vegas due to large in-migration and the scarcity of land for development. In addition, in the future water issues could constrain development and further increase the cost of living in the Las Vegas Valley. These factors have already led to increased development in Nye County and outlying areas of Clark County. Because the majority of socioeconomic impacts would occur during the construction and operations analytical periods, this sensitivity analysis addresses those periods. Impacts during the monitoring or closure analytical periods would be smaller because the workforce would be smaller.

The maximum of about 1,900 repository workers per year would make a small difference in the Las Vegas metropolitan area population of about 2 million. However, if a higher percentage of the onsite workers resided in Nye County, with a population of about 40,000, the socioeconomic impacts could be greater.

The worker residency option could result in increased traffic at the intersection of Nevada State Route 373 and U.S. Highway 95 in Nye County, particularly during the repository construction analytical period. Chapter 6, Section 6.4.3 of this Repository SEIS discusses impacts to regional traffic. Impacts to traffic on U.S. Highway 95 at this intersection under the worker residency option would be similar to those in Section 6.4.3.

A.4.1 SOCIOECONOMIC IMPACTS

The evaluation in Chapter 4, Section 4.1.6 assumes that 80 percent of the proposed repository site workers would live in Clark County and includes impacts to the State of Nevada. For this perspective analysis, DOE evaluated the impacts to the socioeconomic environment in Nye County under the assumption that 80 percent of the proposed repository site workers would live in Nye County (the

80-percent assumption). All other modeling parameters remained the same. The evaluation considered changes to employment, population, three economic measures (real disposable personal income, spending by state and local governments, and Gross Regional Product), housing, and some public services in Nye County. This perspective analysis focused on the impacts in that county. Because DOE estimated that the percentage of onsite workers who would live in Nye County would range between 20 and 80 percent, this discussion and that in Section 4.1.6 present bounding parameters of impacts in the county. This evaluation used the Regional Economic Models, Inc. model, *Policy Insight*®, Version 9, to estimate and project baseline socioeconomic conditions from 2012 to 2067 and to estimate employment and population changes due to the Proposed Action. DOE prepared this alternative analysis of potential socioeconomic impacts as a result of scoping comments from Nye County. This analysis provides a perspective of the range of socioeconomic impacts that could occur. Because the majority of the socioeconomic impacts would occur during the construction and operations analytical periods, this analysis addresses just those periods.

A.4.1.1 Impacts to Employment

A.4.1.1.1 *Impacts to Employment During Construction*

Repository surface and subsurface construction would begin in 2012. In 2014, the peak year of direct project employment during the initial construction analytical period, the Proposed Action would directly employ about 2,590 workers. About 1,860 of these workers, which would include approximately 220 current employees, would work at the repository site in Nye County. Workers employed during construction would include skilled craft workers and professional and technical support staff (such as engineering, safety analysis, and safety and health). Onsite employment during construction would peak during the last year of the construction period in 2016, with about 1,920 workers, as DOE transferred offsite positions and responsibilities from Clark County to the repository site.

Table A-2 lists the estimated direct project employment during the construction analytical period. The direct onsite employment would increase by a factor of 4 from the current level of about 220 workers to about 1,000 at the beginning of the construction period and then to about 1,920 workers by the end of the construction period.

Table A-2. Direct project employment during construction, 2012 to 2016.

Employment	2012	2013	2014	2015	2016
Directly employed project workers[a] (onsite and offsite)	1,720	2,200	2,590	2,550	2,510
Directly employed repository site workers[a] (onsite only)	1,010	1,480	1,860	1,900	1,920

Source: DIRS 182205-Bland 2007, all.
Note: Numbers are rounded to three significant figures.
a. Includes current workers.

During the construction analytical period, the estimated employment baseline (number of jobs without the Proposed Action) in Nye County would grow from about 19,830 persons to about 20,820 persons. Because DOE believes the compensation packages for employment at the proposed repository would be very attractive, the analysis assumed some current Nye County workers would leave their current positions to join the repository workforce. Some of the vacated positions would not be filled because some jobs would be dissolved; others would remain unfilled. The *Policy Insight* model shows that, although the Yucca Mountain project would employ an additional 1,090 construction workers in 2014 (DIRS 182205-Bland 2007, all), this phenomenon could occur because, with construction of the repository, the average wage rate in the area would probably rise. Former sole proprietors and some county-based employers could elect to consolidate or eliminate abandoned positions rather than pay the higher wages necessary to attract replacement employees. Workers new to the labor force, the county, or the construction industry would fill some repository positions. Employment in the construction industry is constantly in flux and assignments begin and end in a relatively short period. Therefore, despite the new jobs at the repository, the number of composite jobs (direct and indirect) would be smaller than the number of direct repository jobs in Nye County during the construction period.

Figure A-3 shows changes in employment in Nye County during the construction analytical period. During construction, about 580 to 1,190 new jobs, or about 2.9 to 5.7 percent of the employment baseline in the county, would result from repository construction. These impacts to employment would be large because they would be at or over 5 percent in 3 of the 5 years of construction. Most of the new jobs in the county would occur in the construction, professional and technical services, retail trade, and food and beverage industries.

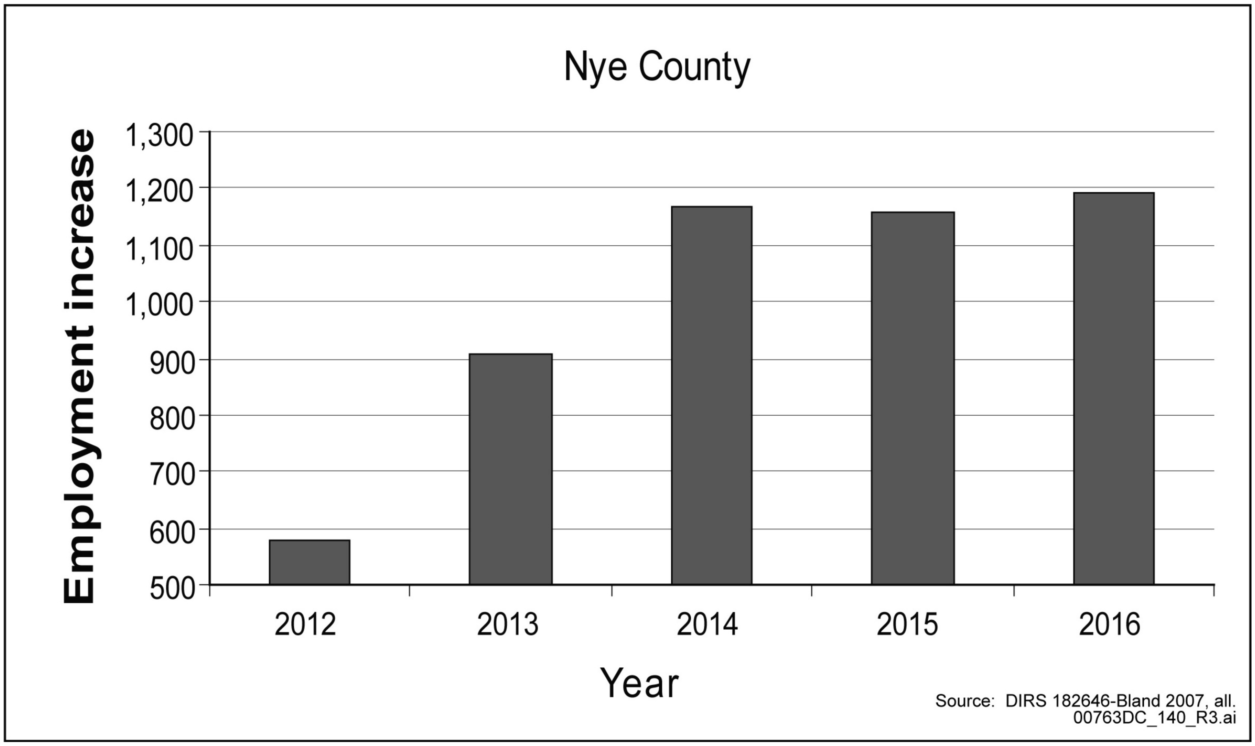

Figure A-3. Changes in Nye County employment from repository construction activities, 2012 to 2016.

A.4.1.1.2 *Impacts to Employment During Operations*

Although the operations analytical period would be from 2017 to 2067, most of the socioeconomic impacts would occur around 2020 in the early years of operations (in which subsurface construction would be concurrent with emplacement activities) and in 2040 when most subsurface construction activities would be complete. Because the years from 2020 to 2040 would be representative of the socioeconomic impacts from proposed activities during operations, the discussion focuses on these two decades.

Direct operations peak employment would occur near the beginning of the operations analytical period, when subsurface construction and emplacement activities occurred concurrently. In 2020, when repository operations would require about 2,590 workers, about 2,000 of these workers would work at the site in Nye County. Direct site employment would range from 2,000 to about 1,520 from 2020 to 2040, and then would be essentially stable with an average of about 560 workers until 2067. The Proposed Action would contribute jobs to the Nye County economy during the entire construction analytical period. The incremental increase in jobs would be about 1,700 jobs in 2020, 1,800 jobs in 2030 and 1,650 jobs in 2040. The number of jobs would decline as DOE completed emplacement activities. Figure A-4 shows the incremental increases over the county employment baseline during the operations period.

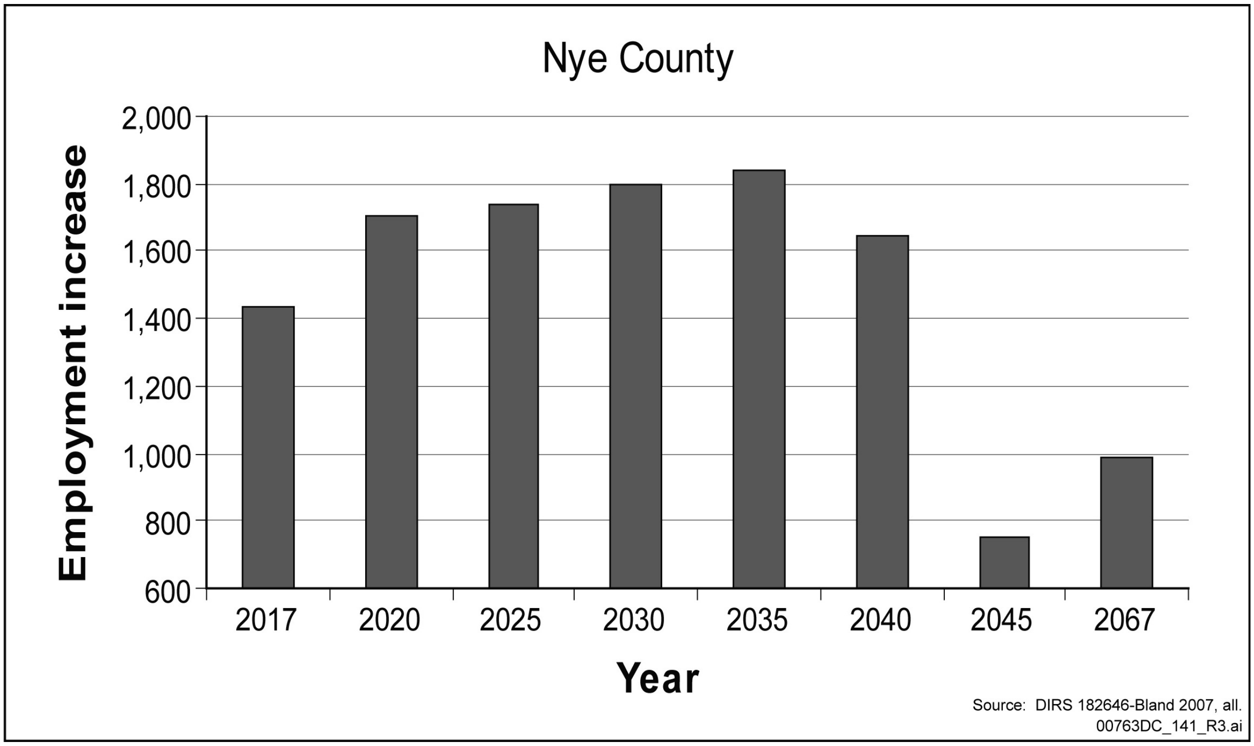

Figure A-4. Incremental changes in Nye County employment from repository operations, 2017 to 2067.

Direct employment would create many indirect jobs if 80 percent of the onsite workforce lived in Nye County because the county employment base is small and not able to provide the additional goods and services workers and their families would need without the creation of additional capacity; that is, more new capacity would be necessary. The Proposed Action would contribute jobs to the Nye County economy during the entire operations analytical period. Incremental changes in population would be smaller than changes in employment because current residents of the county or family members of the directly employed workers (rather than in-migrants) would fill many of the indirect jobs that resulted from the direct employment.

In 2020, Nye County would gain about 1,700 jobs. The change in the number of jobs would be substantial and represent an almost 8-percent acceleration of job growth over the baseline in the county for that year. From 2020 until 2040, job growth in Nye County without the repository would average about 1 percent each year; with the repository, the average annual growth rate would be 1.3 percent (almost a third more quickly). The Nye County estimated employment baseline for 2020 is 21,700 jobs. With the repository, the number of jobs would increase to 23,400 in 2020 (1,700 new jobs added to the

21,700 baseline jobs—jobs that would be in the county without the Proposed Action—for a total of 23,400 jobs). In 2040, the baseline number of jobs would be 26,300, and the number of additional repository jobs, 1,650, would mean a total of 27,950 jobs in the county. Generally, the number of baseline jobs in a county grows over time as it does in this analysis—from 21,700 in 2020 to 26,300 in 2040. Employment in 2040 and 2041 would be very similar, and repository employment after 2040 would be too small to affect the county. Table A-3 lists the baseline and the changes in employment for 2020 to 2040 in Nye County. Although the operations analytical period would extend beyond 2040, onsite employment and, therefore, impacts would decline after 2040. By 2042, the impacts to employment would decline to below 3 percent over the baseline.

Table A-3. Changes in Nye County employment from repository activities in the operations analytical period, representative years.

Change	2020	2025	2030	2035	2040
Incremental change[a]	1,700	1,740	1,800	1,840	1,650
Baseline employment[a]	21,700	22,600	23,700	24,900	26,300
Percent change over baseline[b]	7.9	7.7	7.6	7.4	6.3

Source: DIRS 182646-Bland 2007, all.
a. Numbers are rounded to three significant figures.
b. Percentages are rounded to two significant figures.

The change in the rate of job growth during operations would be pronounced. Most of the new jobs from the first 25 years of the operations analytical period would be professional and technical services positions, followed by federal civil service positions, retail trade positions, jobs in the food and beverage industry, and local government jobs. The construction industry would have a decreasing presence as the operations period advanced.

A.4.1.1.3 *Summary of Employment Impacts*

Under the 80-percent assumption, impacts on employment in Nye County would be large (greater than 5 percent over the baseline) for the first 30 years of construction and operations and then small (less than 3 percent over the applicable baselines). The repository would be Nye County’s largest employer.

A.4.1.2 Impacts to Population

Incremental changes in population due to repository employment would largely be the result of the choice of county of residence that workers and their families made. Changes in population would lag behind changes in employment by several years.

A.4.1.2.1 *Impacts to Population During Construction*

Without the Proposed Action, Nye County’s estimated baseline population would grow from 55,800 to 62,300 people during the construction analytical period. With the 80-percent assumption, the Proposed Action would result in an incremental increase in population in Nye County that grew steadily from about 81 persons in 2012 to 560 persons in 2016; these increases would be about 0.15 to 0.9 percent of the county’s population baseline, which would be small. In part, the increase in population would be small because many construction workers would live in temporary worker camps and, therefore, would not become part of the permanent census of the county.

A.4.1.2.2 *Impacts to Population During Operations*

In general, increases in population would lag behind increases in employment by several years because some workers would delay relocation. Because the labor force in Nye County is small, many operations workers who would live in Nye County would be new to the county. As a result of repository activities, in 2040 about 4,120 additional people, a change of 4.6 percent over the county's baseline population of 90,100 in that year, would live in Nye County, which would be a moderate impact. State and local government agencies would need to adjust levels of service to accommodate the increase in population. Unlike the temporary nature of increases during the construction analytical period, increases in population from repository activities during operations would be relatively permanent. The impact to population over the baseline would be moderate at first—3 to 5 percent from 2020 until 2040—and then it would decline to just below 3 percent. The repository would have a defining presence on the population in Nye County. Private-sector providers would need to consider the effects of the repository in their strategic plans. Figure A-5 shows the projected population increases from the repository in Nye County during the operations analytical period. Increases in population would result in impacts to housing and public services (Sections A.4.1.4 and A.4.1.5, respectively). Without the repository, Nye County's population would grow at an average annual rate of 1.4 percent; under the 80-percent assumption for this analysis, the county would grow at an average annual rate of 1.7 percent.

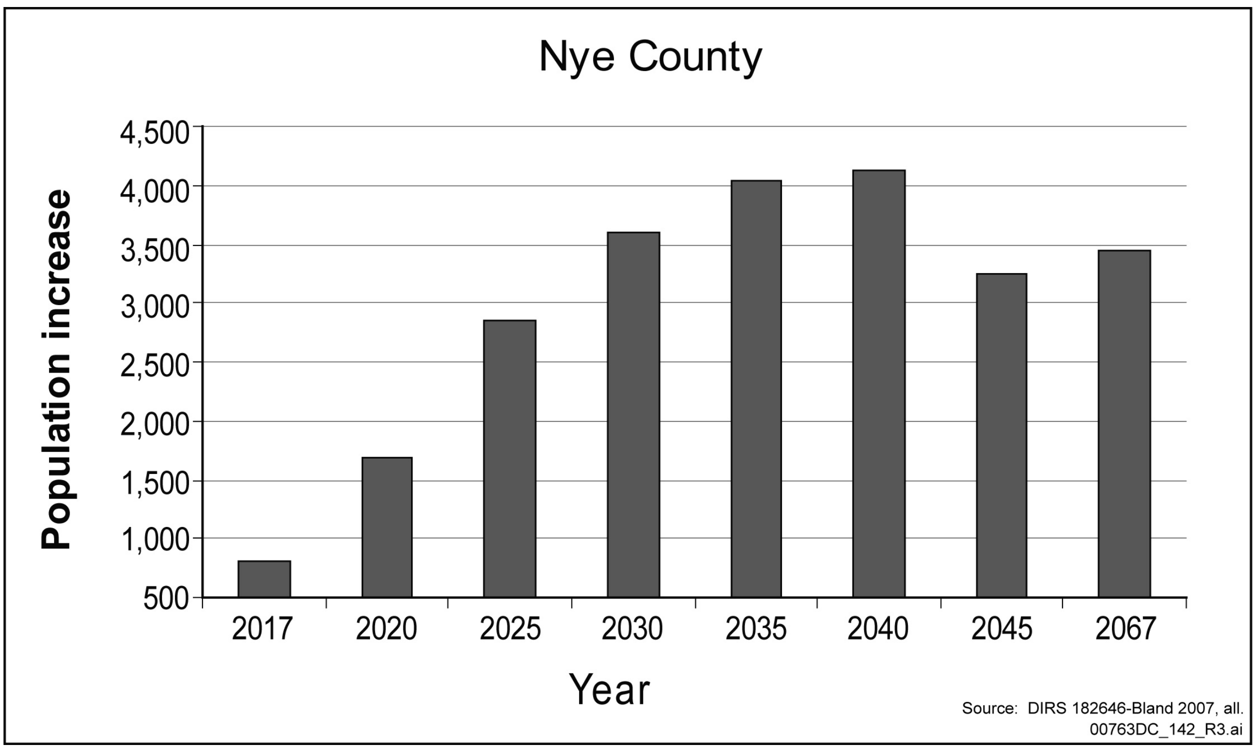

Figure A-5. Changes in Nye County population from repository operations, 2017 to 2067.

A.4.1.3 Impacts to Economic Measures

This section discusses changes in economic measures in Nye County that would result from repository activities during the construction and operations analytical periods. (Values are in 2006 dollars.)

A.4.1.3.1 *Impacts to Economic Measures During Construction*

Increases in real disposable personal income (after-tax income) in Nye County would peak in 2016 with an increase of about $65.7 million under the 80-percent assumption, which would be a moderate increase of 4.5 percent over the baseline of $1.47 billion. During the construction analytical period, the increase in real disposable personal income would result primarily from onsite worker wages. In 2016, per capita (per person) real disposable personal income would increase by about $800 to $24,600. Figure A-6 shows information about changes in real disposable personal income for the construction and operations periods.

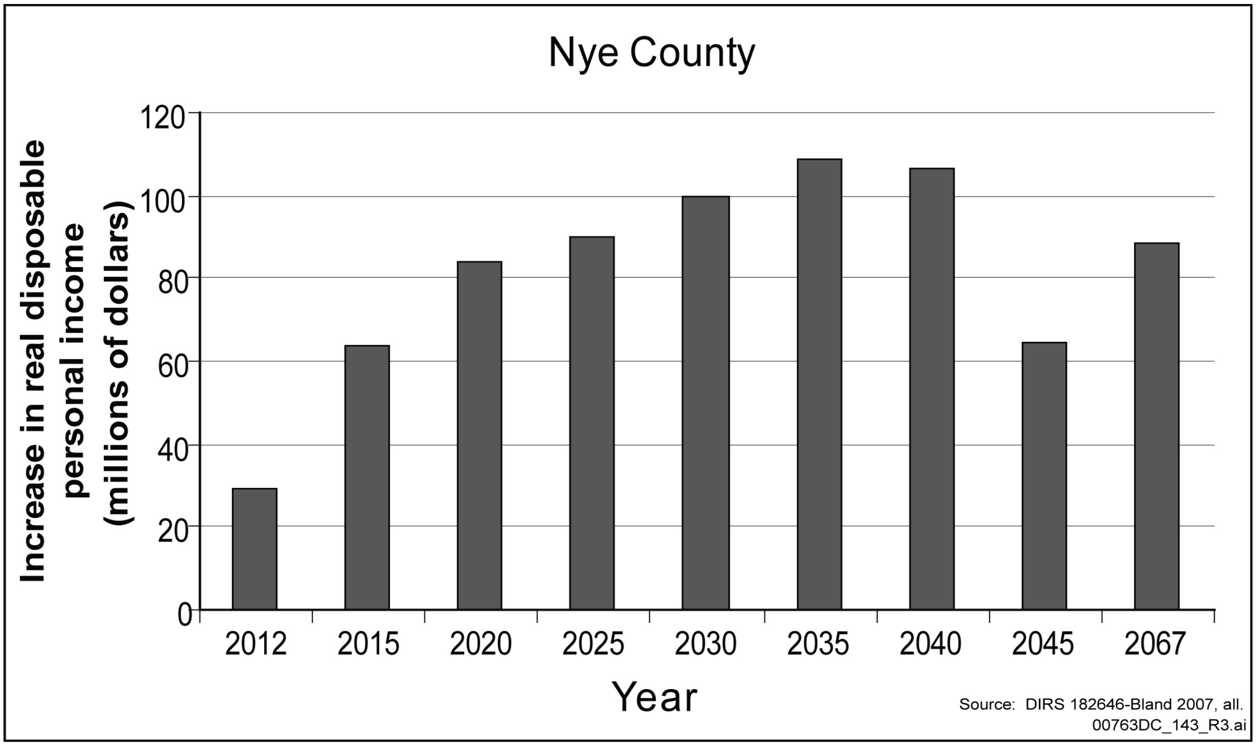

Figure A-6. Changes in real disposable personal income in Nye County during the construction and operations analytical periods, 2012 to 2067.

During the construction analytical period, increases in Gross Regional Product in Nye County would peak at the end of the construction period at about $86.9 million or about 5.4 percent of the baseline. The increase would occur as retailers and the service industry escalated efforts to produce goods and services for repository workers and other residents of Nye County. The county would produce some repository construction products (for example, concrete and tools), and those sales would be a part of the increases in Gross Regional Product. Per capita Gross Regional Product would grow by an addition $1,200. Figure A-7 shows estimated changes in Gross Regional Product for the construction and operations periods.

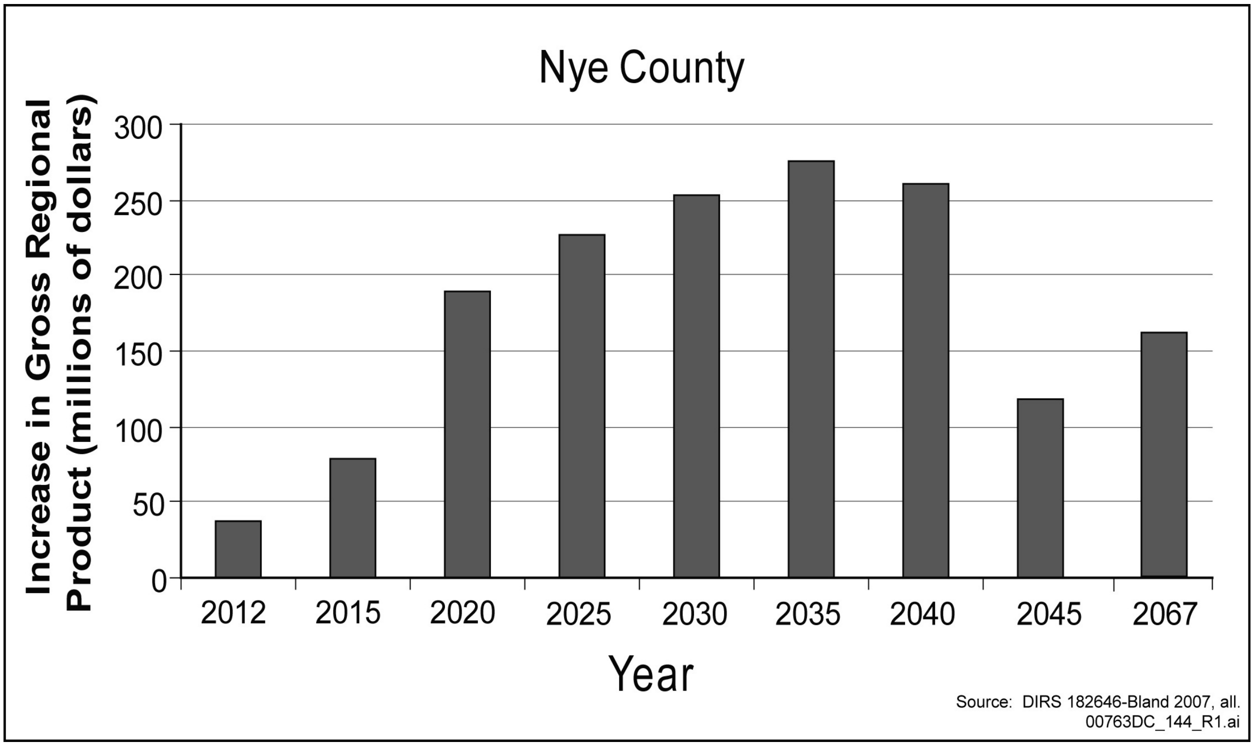

Figure A-7. Changes in Gross Regional Product in Nye County from repository activities during the construction and operations analytical periods, 2012 to 2067.

Changes in expenditures by the State of Nevada and local governments in Nye County during construction would peak at $2.4 million, a small change of less than 1 percent over the baseline. These changes would result from small incremental population increases during construction. Spending by state and local governments would be primarily from revenues from sales of goods and services. Per capita expenditures by state and local governments would increase very slightly, by about $10. Figure A-8 shows estimated changes in spending by state and local governments for the construction and operations analytical periods.

During construction, Nye County would experience moderate to large increases over the Gross Regional Product baseline and small to moderate changes in real disposable personal income over the baseline. Impacts to state and local government spending would be small—less than 1 percent.

A.4.1.3.2 *Impacts to Economic Measures During Operations*

As with employment and population, the years from 2020 to 2040 would be the most representative of socioeconomic impacts from repository operations. Nye County would experience a large impact from two economic measures during operations: Gross Regional Product and real disposable personal income. Figures A-6 to A-8 show the changes in economic measures in Nye County that would result from the repository project during the construction and operations analytical periods under the 80-percent assumption.

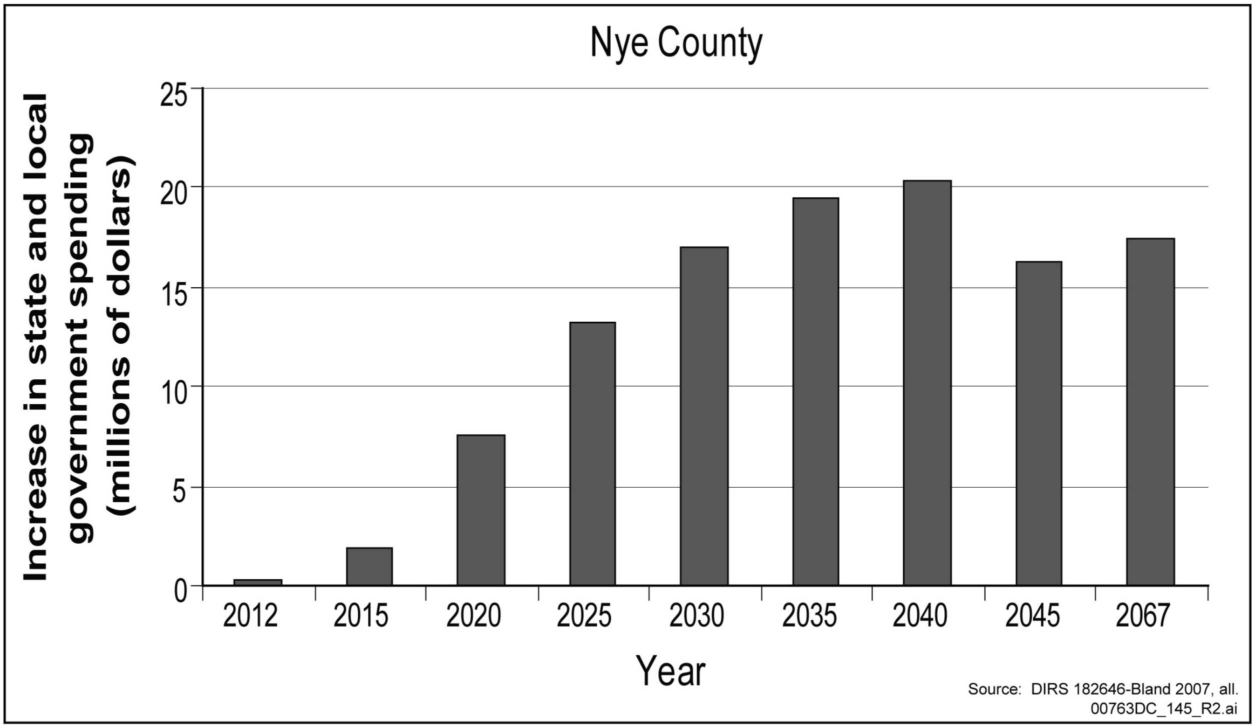

Figure A-8. Changes in spending by state and local governments in Nye County from repository activities during the construction and operations analytical periods, 2012 to 2067.

During the operations analytical period, the impact of changes in real disposable personal income would be proportionally greater than during construction because this economic measure more fully captures wages earned by directly and indirectly employed workers. Most operations workers would make Nye County their permanent home and spend the majority of their earnings in that county. Increases in real disposable personal income would be large from 2020 to 2040. Impacts over the baseline would range from 5.2 percent in 2020 to 4.3 percent in 2040. The impact after that would be small, less than 3 percent. Increases in real disposable personal income would range from $83.9 million in 2020 to about $106.5 million in 2040. Repository workers who lived in Nye County would spend most of their wages in that county and, in turn, create income for the providers of goods and services. Economic activity, which would include incidental spending by workers who lived in Clark County but worked in Nye County, would be responsible for this phenomenon. In addition, many indirect jobs and the income from those jobs would remain in Nye County. In 2020, repository activity would result in per capita real disposable personal income growing from the baseline $23,720 to $24,360. Figure A-6 shows information about changes in real disposable personal income for the construction and operations analytical periods.

Nye County would experience an increase from $189.5 million in 2020 to $260.4 million in 2040 in Gross Regional Product, an increase of 10.5 to 8.6 percent, respectively, over the baseline. These would be large impacts. The Gross Regional Project would increase as repository workers and their families demanded and consumed goods and services and area businesses met the demand by providing the desired products. Gross Regional Product is an important variable used to determine an area's economic health. The repository-related increase in Gross Regional Product coupled with the large impact to real disposable personal income would confirm the county's economic viability. Impacts to Gross Regional

Product would remain moderate from about 2040 to 2067. Figure A-7 shows changes in Gross Regional Product for the construction and operations analytical periods.

Spending by the State of Nevada and local governments in Nye County would increase by $7.5 million or 2.6 percent of the baseline in 2020 and by $20.4 million or 4.8 percent in 2040. Nye County could spend tax and marginal revenues (revenue sources that originate outside the county such as the Payments-Equal-to-Taxes provisions) from increased economic activity associated with the repository. Figure A-8 shows changes in spending by state and local government for the construction and operations analytical periods. Much of the spending could be due to the incremental increase in population from the repository. Throughout the operations period, the Proposed Action would have almost no impact on per capita spending by state and local governments. In 2020, per capita baseline spending by state and local government would be $4,305. Construction and operation of the repository would increase per capita spending by state and local governments by $15.

During operations, impacts to real disposable personal income and Gross Regional Product would generally be large. Impacts to spending by state and local governments would generally be moderate.

A.4.1.3.3 *Summary of Impacts to Economic Measures*

Under the 80-percent assumption, impacts from repository-related activities in Nye County would be more pronounced during the operations analytical period as workers and families established residency and spent earnings. Business activity would increase due to the production of goods and services to meet resident demands. Other businesses would produce increased goods and services to provide products for repository operations. As a result, the largest affected economic measure would be Gross Regional Product.

A.4.1.4 Impacts to Housing

Nye County and more specifically Pahrump have recently experienced rapid and largely unanticipated growth, and the county has a limited housing inventory for absorption of new workers and worker families. However, because the estimated incremental increases in population during construction would be small, the increased demand for housing would also be small. Many construction workers would live in temporary construction camps and not need additional housing.

Nye County would experience small to moderate increases in population when operation activities began. As a result of repository activities under the 80-percent assumption, as many as 4,120 additional people would live in Nye County in 2040. This would be an increase of 4.6 percent over the population baseline of 90,100 residents in that year. Because of its proximity to the proposed repository site, much of the additional demand for housing could concentrate in Pahrump. Demands on the county's specific housing inventory available at that time should be small to moderate because housing stock generally increases at approximately the same rate the population increases. Nye County would experience a rate of population growth of approximately 1.4 percent annually even without the Proposed Action. However, the impact to housing could be moderate, rather than small, because (1) the demand should be concentrated in Pahrump, which is currently managing very rapid growth (more rapid than the county as a whole), and (2) although there are no local or state growth control measures that limit housing development, water rights are increasingly scarce.

Nye County has an adequate supply of undeveloped land to meet expected future demands. The incremental increase in population from repository-related activities would occur over a long period and be predictable, so the private sector housing market could readily adapt. In addition, the county has demonstrated concern about future growth and has taken action to acquire land and prepared plans for a comprehensive live-work community to facilitate and accommodate the orderly development of land use that repository activities could trigger.

Nye County has also acquired land to facilitate and accommodate the orderly development of land uses that repository activities could trigger. The county's infrastructure system, particularly in Pahrump, is currently strained and at capacity. In addition, the desert setting of the county means developers are dependent on water rights, which are crucial to development. With a very limited supply of water and a rapidly growing population, the ability of the private or public sector to meet housing demands remains speculative. Unless infrastructure systems, including water rights, can expand, adequate housing supply for anticipated growth could be compromised.

Although the need for additional housing in Nye County can readily be predicted, the resolution of water rights issues and infrastructure funding issues could be much more protracted.

DOE analyzed potential impacts to housing at the county rather than the community level. The Department did not attempt to predict incremental housing demand at the community level because housing preferences (mobile home, modular assembly, stick-built), density or cluster choices (single family, multifamily), and desired lot sizes are difficult to predict.

A.4.1.5 Impacts to Public Services

The moderate repository-related increases in population in Nye County could cause impacts to public services, particularly in southern Nye County and Pahrump. Public services are currently at capacity and, because of their geographic separation from one another, Nye County communities cannot readily share public services. Although the current tax structure would not be able to support fully the increased demand on public services, because the changes in population in the county would occur steadily over a long period, the expected long-term increases in government revenues would enable the County to plan for the increased demands. However, since expansion of some public services could be necessary before the county could levy and collect taxes through the expansion of the tax base, additional nontax revenues could be sought. Sources for additional revenues could include mitigation funding or loans secured through the issuance of bonds. Cooperative mutual aid agreements could supplement the level of services to communities. DOE implementation of such agreements in conjunction with the Proposed Action would reduce strains on county services.

If the incremental population increased reflect the current patterns in Nevada (rather than Nye County, which has a large retirement-age population), at any given time about 21 percent of new residents would be school-age children. Schools in Nye County are presently at capacity, and the county is widely reliant on portable units. The county and the communities in the county would continue to provide services as the government revenue base grew. The recently opened hospital in Pahrump and the ample medical services in metropolitan Las Vegas help to alleviate the scarcity of medical services in Nye County.

Gross Regional Product would increase with repository activities. Under the 80-percent assumption, the increase in Nye County would be very large, approximately 10 percent when repository operations began.

The large impact to Gross Regional Product would result in tax revenue for local and state sources. Nevada collects sales tax of 6.75 percent (except on groceries). There is no corporate, personal, unitary, inventory, or franchise tax in the state or in Nye County, so wages and business profits would not directly benefit the coffers of state and local governments. Pahrump has the lowest property tax assessment of the county's local jurisdictions. As increased earnings drove the increases in real disposable personal income, businesses would rally to provide more goods and services to meet the increased demand. The purchase of some goods and services due to repository construction and operations would occur from county-based vendors. Under the 80-percent assumption, these increases would be noticeable because the impacts would represent a large percentage increase rather than a large absolute increase. DOE facilities have historically had cooperative agreements with local governments for mutual aid and support of emergency services. DOE implementation of such an agreement in conjunction with the Proposed Action would reduce strains on regional emergency services infrastructure. Repository-related impacts to public services could require mitigation because the impacts would probably be community-specific rather than countywide and because the unincorporated communities would have little ability to generate tax revenue for public services. The recently opened 24-bed hospital in Pahrump, along with the ample services available in metropolitan Las Vegas, could alleviate the scarcity of medical services in Nye County.

A.4.1.6 Summary of Socioeconomic Impacts During Construction and Operations

If 80 percent of the repository site workers lived in Nye County, there would be meaningful, measurable socioeconomic impacts in the county from construction and operations. The greater impacts would be long term and would occur during the operations analytical period. Repository-related incremental changes in employment in Nye County would generally be large during construction because the workforce at the repository would represent such a big portion of the county's current job base. The changes over the baseline in Gross Regional Product would be large because county businesses would respond to the demand for additional goods and services. Incremental changes in population during construction would be small because most construction workers would not relocate to Nye County with their families but instead would live in temporary work camps and return to out-of-county homes on days off. Changes in state and local spending would be small because agencies would not need to provide additional services for small, temporary increases in population. Increases in real disposable personal income would be moderate as the estimated 1,000 to 1,900 onsite project workers earned wages. The increases in real disposable personal income and Gross Regional Product would result in a more vibrant economy and generally would be beneficial. The increase in employment would result in increases in population, which in turn would cause the economy to grow. Growth in population can strain public services, and increases in population can change the ambiance of an area.

Nye County would experience larger socioeconomic impacts during repository operations than during construction. Incremental changes in population and spending by state and local government would be moderate in the operations analytical period—generally 3 to 5 percent over the baselines. Changes in employment and real disposable personal income would generally be large—from 5 to almost 8 percent. Changes to the county's Gross Regional Product would be even larger—more than 10 percent over the baseline. However, public services are currently at capacity. Repository-related impacts to public services could require mitigation because the unincorporated communities would have little ability to generate tax revenue for public services.

A.5 Extended Monitoring Analytical Period

Chapter 2, Section 2.1.2 of this Repository SEIS describes the four analytical periods for the Proposed Action. For purposes of analysis in this SEIS, monitoring and closure activities would end 50 years after the emplacement of the last waste package. The 10-year closure analytical period would overlap the last 10 years of monitoring activities. Chapter 4, Section 4.1 presents the estimated environmental impacts for monitoring and closure activities during the 50-year timeframe. However, DOE could extend the monitoring analytical period an additional 200 years (that is, ending 250 years after the emplacement of the last waste package). This section presents the potential additional environmental impacts that could occur as the result of an extended monitoring period beyond the initial 50 years of monitoring.

A.5.1 ENVIRONMENTAL IMPACTS OF EXTENDED MONITORING

DOE anticipates that several environmental resource categories would not have any continued impacts due to extended monitoring, or would have impacts the same as those during the initial 50 years of monitoring. In the cases of cultural resources and aesthetics, the impacts would have already occurred and, to the extent necessary, DOE would have mitigated them. New cultural resources or scenic areas would be unlikely to become of interest. In the case of socioeconomics, the workforce associated with extended monitoring would be so small it would not be perceptible in the regional or state economy. In relation to environmental justice, DOE concluded in Chapter 4, Section 4.1.13.3, that, based on the analyses performed, “no disproportionately high and adverse impacts would result from the Proposed Action.” In terms of accidents, no new scenarios or accident categories would be applicable to extended monitoring. Impacts from noise would not differ from those during the initial 50-year monitoring analytical period. There would be some noise from ventilation fans, compressors, and other machinery if DOE maintained them beyond the first 50 years of monitoring. The distances to the site boundaries would be unlikely to change.

The following sections discuss the potential additional environmental impacts of monitoring an additional 200 years after emplacement of the last waste package and repository closure.

A.5.1.1 Land Use and Ownership

As discussed in Chapter 4, Section 4.1.1.1, withdrawal of lands for repository purposes would prohibit public use of the lands. Extended monitoring would extend the unavailability of the withdrawn lands for other uses.

A.5.1.2 Air Quality

Chapter 4, Section 4.1.2.3 of this Repository SEIS presents impacts to air quality from monitoring. The analysis concluded that because surface construction, subsurface excavation, and subsurface emplacement activities would be complete, emissions would probably be substantially lower from those listed in Table 4-3. This conclusion would also apply to the extended monitoring analytical period.

A.5.1.3 Hydrology

Chapter 4, Section 4.1.3.2.3 of this Repository SEIS states that “water demand during the monitoring and closure analytical periods would be lower and of less concern and would be expected to remain as presented in the Yucca Mountain FEIS." The estimated water requirement for monitoring activities would be 7,400 cubic meters (6 acre-feet) per year and would be unlikely to change during the extended monitoring analytical period.

A.5.1.4 Biological Resources and Soils

The potential impacts to biological resources and soils due to an extended monitoring analytical period would be smaller than those DOE described in Chapter 4, Section 4.1.4 of this Repository SEIS. DOE does not anticipate additional land disturbance during the extended monitoring period that could add to disrupted or fragmented habitat; the greatly reduced workforce and level of site activities would result in a decrease in the deaths of individual species due to traffic and human activity.

A.5.1.5 Occupational and Public Health and Safety

Potential nonradiological health and safety impacts to workers would occur from industrial hazards and exposure to naturally occurring cristobalite and erionite. Potential health impacts to members of the public would be from exposure to airborne releases of naturally occurring hazardous materials and criteria pollutants.

From a radiological health and safety standpoint to workers, potential impacts would come from exposure to naturally occurring and manmade radiation and radioactive materials. There could also be exposure to members of the public from airborne releases of naturally occurring and manmade radionuclides.

A.5.1.5.1 *Nonradiological Impacts*

Chapter 4, Section 4.1.7.1.3 of this Repository SEIS describes nonradiological health impacts during monitoring. The analysis assumed that the heath and safety impacts to workers for the monitoring analytical period would be similar to those described in the Yucca Mountain FEIS. With an extended monitoring period, DOE anticipates that industrial hazard impacts for all workers would increase as follows:

Total recordable cases:	1,000 additional
Lost workday cases:	420 additional
Fatalities:	0.95 additional

From the standpoint of potential exposure to cristobalite and erionite, extended monitoring activities would be unlikely to generate large quantities of dust, and there should be reduced potential for exposure. Potential impacts to members of the public would be unlikely from naturally occurring hazardous materials or criteria pollutants because construction would be complete and there would be fewer emissions in comparison with previous periods.

A.5.1.5.2 *Radiological Impacts*

The principal contributor to radiological health impacts to workers would be from subsurface facility monitoring and maintenance activities that DOE could conduct during the extended monitoring analytical period. Potential radiological health impacts to the public from monitoring activities could result from exposure to releases of naturally occurring radon-222 and its decay products in subsurface exhaust ventilation air.

Table A-4 lists the radiological impacts from 200 years of extended monitoring.

Table A-4. Radiological impacts from 200 years of extended monitoring.

Occupational and public health and safety	Impacts for the Proposed Action	Additional impacts for 200 years of extended monitoring
Public		
MEI (probability of an LCF)	0.00032	No change
Population (LCFs)	8	18
Workers (involved and noninvolved)		
Population (LCFs)	3.5	2.8

LCF = Latent cancer fatality.
MEI = Maximally exposed individual.

A.5.1.6 Utilities, Energy, Materials, and Site Services

The extended monitoring analytical period would result in the continued consumption of energy in terms of electricity use and the consumption of fossil fuel, oils, and lubricants. There would be no additional consumption of construction materials. Table 4-29 in Section 4.1.11 lists estimates for the use of electricity and fossil fuels. The following estimates represent continued consumption of materials for the extended monitoring period:

Electricity use: 12.6 million megawatt-hours (based on 63,000 megawatt-hours per year) additional
Fossil fuel: 210 million liters (55 million gallons) additional
Oils and lubricants: 44 million liters (12 million gallons) additional

A.5.1.7 Waste and Hazardous Materials

During the extended monitoring analytical period, DOE could continue to generate sanitary sewage, low-level radioactive waste, and sanitary and industrial waste. DOE does not anticipate the generation of hazardous waste or industrial wastewater. The Department assumed that the disposition of each waste stream would continue as described in Chapter 4, Section 4.1.12 of this Repository SEIS. The following are the estimated volumes of waste that DOE would generate during the extended monitoring period:

Sanitary sewage: 656,000 cubic meters (858,000 cubic yards)
Low-level radioactive waste: 13,000 cubic meters (17,000 cubic yards) (includes solids and liquids)
Sanitary and industrial waste: 53,000 cubic meters (68,000 cubic yards)

A.5.1.8 Socioeconomics

Potential impacts to socioeconomic variables in the region of influence due to extended monitoring activities would be smaller than the impacts DOE estimated for construction and emplacement. Because direct repository employment during the extending monitoring analytical period would not involve construction or operations workers, the impacts would be the same as those for the initial 50 years of monitoring. Because the extended monitoring period would be so far in the future and would require only periodic activities from a small number of employees, DOE has not attempted to quantify the number of workers or the potential impacts. Potential impacts to associated population growth and other economic measures would be small.

A.6 Highway Routing

The Yucca Mountain FEIS (DIRS 155970-DOE 2002, Section J.3.1.3) examined the sensitivity of transportation impacts to highway routes in Nevada. In addition to analyzing the impacts of using highway routes that would meet U.S. Department of Transportation requirements for transport of spent nuclear fuel, the FEIS evaluated how the estimated impacts would differ if truck shipments of spent nuclear fuel and high-level radioactive waste for the mostly truck scenario used other highway routes in Nevada. This scenario involved the shipment of about 53,000 truck casks of spent nuclear fuel and high-level radioactive waste. The Nevada Department of Transportation examined six other routes in a 1989 study (DIRS 103072-Ardila-Coulson 1989, pp. 36 and 45). The study described the routes as follows:

- Route A. Route A begins at Interstate Highway 80 in Wendover and follows U.S. Highways 93A, 93, and 6, Nevada State Routes 318 and 375, U.S. Highway 93, Interstate Highway 15, State Route 215, and U.S. Highway 95 (through Ely, Hiko, and Las Vegas, Nevada) to Yucca Mountain.

- Route B. Route B also begins at Interstate Highway 80 in Wendover but follows U.S. Highways 93A, 93, 6, and 95 (through Ely, Tonopah, and Amargosa Valley, Nevada) to Yucca Mountain.

- Route C. Route C begins at Interstate Highway 15 in Baker, California, and follows California State Highway 127, Nevada State Route 373, and U.S. Highway 95 (through Amargosa Valley, Nevada) to Yucca Mountain.

- Route D. Route D also begins at Baker, California, but follows Interstate Highway 15, Nevada State Route 160, and U.S. Highway 95 (through Arden and Pahrump, Nevada) to Yucca Mountain.

- Route E. Route E begins at Interstate Highway 40 near Needles, California, and follows U.S. Highway 95, Nevada State Route 164, Interstate Highway 15, California State Highway 127, and U.S. Highway 95 (through Searchlight, Nevada; Baker, California; and Amargosa Valley, Nevada) to Yucca Mountain.

- Route F. Route F also begins at Interstate Highway 40 near Needles, California, but follows U.S. Highway 95, Nevada State Route 164, Interstate Highway 15, Nevada State Route 160, and U.S. Highway 95 (via Searchlight, Arden, and Pahrump, Nevada) to Yucca Mountain.

Table A-5 lists the sensitivity cases DOE evaluated based on the Nevada Department of Transportation routes, and Figure A-9 shows the routes. Tables A-6 and A-7 list the range of impacts nationally and in Nevada, respectively, of using these different routes for the mostly truck scenario. These tables compare the estimated impacts for the highways identified in the Nevada study with those estimated for shipments that would follow routes consistent with current U.S. 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 these shipments, as permitted under U.S. Department of Transportation regulations (49 CFR 397.103), the analysis assumed that shipments of spent nuclear fuel and high-level radioactive waste would enter Nevada on Interstate Highway 15 from either the northeast or southwest. The analysis also assumed that shipments traveling on Interstate Highway 15 from the northeast would use the northern Las Vegas Beltway to connect to U.S. Highway 95 and continue to the Yucca Mountain site. Shipments from the southwest on Interstate Highway 15

Table A-5. Nevada routing sensitivity cases analyzed for truck shipments.

Case	Description
1	To Yucca Mountain from Barstow, California, using I-15 to Nevada State Route 160 to U.S. Highway 95 (Nevada D and F)
2	To Yucca Mountain from Barstow using I-15 to California State Highway 127 to Nevada State Route 373 to U.S. Highway 95 (Nevada C)
3	To Yucca Mountain from Needles using U.S. Highway 95 to Nevada State Route 164 to I-15 to California State Highway 127 to Nevada State Route 373 to U.S. Highway 95 (Nevada E)
4	To Yucca Mountain from Needles using U.S. Highway 95 to Nevada State Route 164 to I-15 to Nevada State Route 160 to U.S. Highway 95 (variation of Nevada E)
5	To Yucca Mountain from Wendover using U.S. Highway 93A to U.S. Highway 93 to U.S. Highway 6 to U.S. Highway 95 (Nevada B)
6	To Yucca Mountain from Wendover using U.S. Highway 93A to U.S. Highway 93 to U.S. Highway 6 to Nevada State Route 318 to Nevada State Route 375 to U.S. Highway 93 to I-15 to the Las Vegas Beltway to U.S. Highway 95 (Nevada A)
7	To Yucca Mountain from Las Vegas using I-15 (for shipments entering Nevada at the Arizona and California borders) to U.S. Highway 95 (Spaghetti Bowl interchange)

I = Interstate Highway.

would use the southern and western Las Vegas Beltway to connect to U.S. Highway 95 and continue to the Yucca Mountain site.

On the national level, the choice of highway routes in Nevada would have very little impact on the total impacts of transporting spent nuclear fuel and high-level radioactive waste. For the base case, the analysis estimated 14 total fatalities. For Cases 1 through 7, the estimated number of fatalities would range from 13 to 14.

Transportation impacts could vary considerably at the state level depending on the highway routes DOE used in Nevada. For example, if Nevada chose Nevada Routes A or B for truck shipments to Yucca Mountain, to the exclusion of other routes, most shipments would probably go through Utah, with few going through California. If Nevada chose Nevada Routes C, D, E, or F for truck shipments to Yucca Mountain, to the exclusion of other routes, most shipments would probably go through California, with few going through Utah.

In Nevada, impacts would generally be small for all cases. For routes that used the Spaghetti Bowl interchange (Case 7) and routes that used Interstate Highway 15 and Nevada State Route 160 (Cases 1 and 3), the impacts would be about the same as those for the base case route. For Nevada Routes A and B, the impacts would be about a factor of 2 times larger than the base case route. These shipments would travel through White Pine County. For Nevada Routes C and E (Cases 2 and 4), the impacts would be about a factor of 2 times smaller than the base case route. Case 2 involves shipments that would use California State Highway 127 through Death Valley.

DOE based the results in Tables A-6 and A-7 on the shipment of about 53,000 truck casks of spent nuclear fuel and high-level radioactive waste. This Repository SEIS discusses an estimated 2,650 truck shipments of spent nuclear fuel and high-level radioactive waste. Therefore, the purpose of the results in Tables A-6 and A-7 is to provide a perspective on how transportation impacts could change based on changes in highway routing. Based on the results in Table A-6 and because truck casks would account for only about 22 percent of the total estimated number of casks in this SEIS that DOE would ship, it is likely that changes in highway routing would only result in small, if any, changes to the total estimated impacts for national transportation for the Proposed Action.

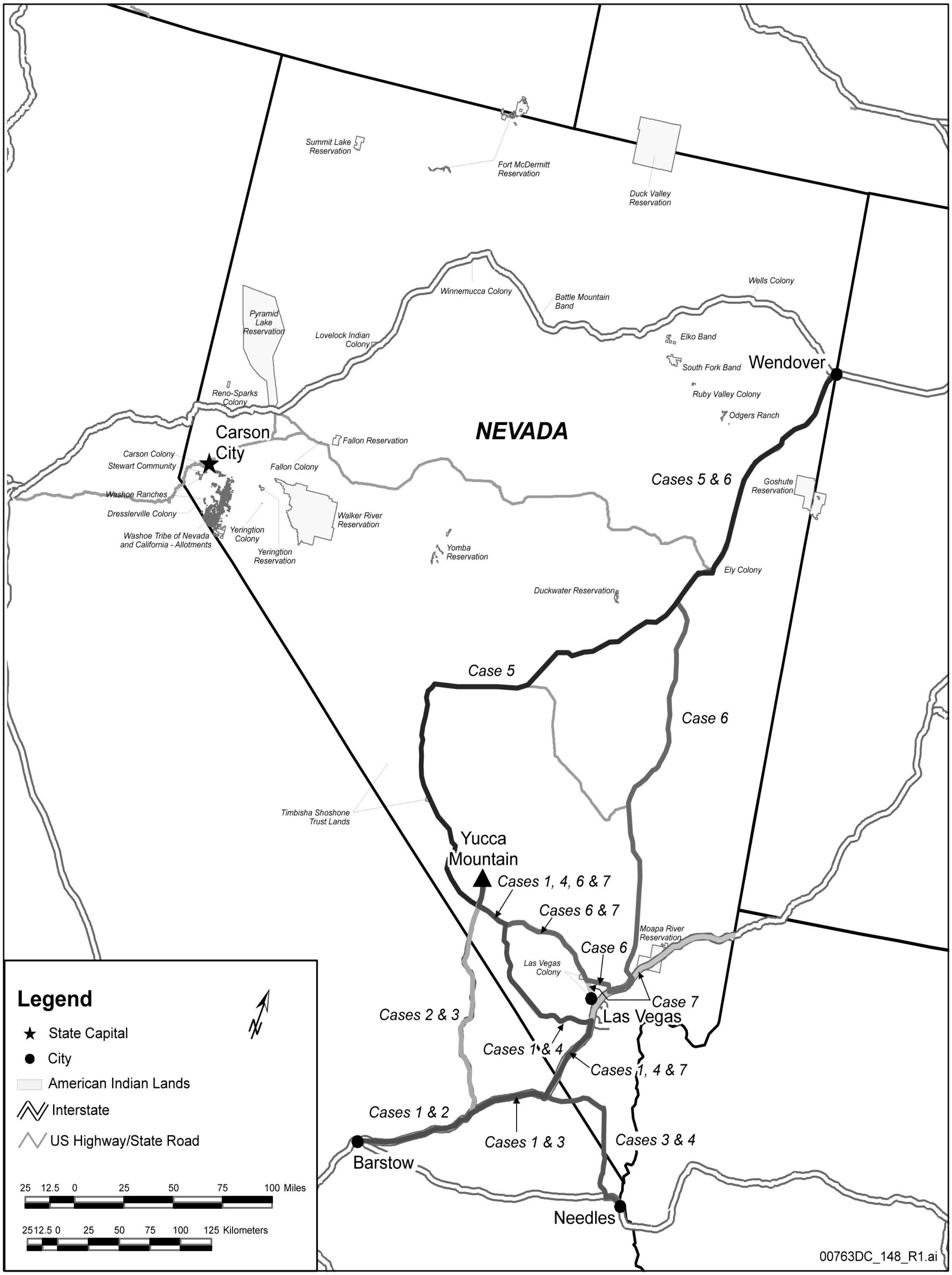

Figure A-9. Nevada routing sensitivity cases.

Table A-6. Comparison of national impacts from the mostly truck scenario routing sensitivity analyses in the Yucca Mountain FEIS.

Impact	Base Case	Case 1 Barstow via Nevada 160	Case 2 Barstow via California 127	Case 3 Needles via Nevada 160	Case 4 Needles via U.S. 95	Case 5 Wendover via U.S. 95	Case 6 Wendover via Las Vegas Beltway	Case 7 I-15 and U.S. 95 (Spaghetti Bowl)
Public incident-free dose (person-rem)	5,000	5,200	5,100	4,900	5,000	4,600	4,800	5,100
Occupational incident-free dose (person-rem)	14,000	15,000	15,000	14,000	14,000	15,000	15,000	14,000
Nonradioactive pollution health effects	0.93	0.93	0.93	0.89	0.88	0.79	0.81	1.1
Public incident-free risk of latent cancer fatality	2.5	2.6	2.6	2.4	2.5	2.3	2.4	2.6
Occupational incident-free risk of latent cancer Fatality	5.6	6	5.8	5.6	5.7	5.9	5.9	5.6
Radiological accident risk (person-rem)	0.46	0.36	0.35	0.35	0.35	0.39	0.4	0.52
Radiological accident risk of latent cancer fatality	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002	0.0002	0.0003
Traffic fatalities	4.5	4.5	4.2	4.3	4.2	4.9	5	4.5
Total fatalities	14	14	14	13	13	14	14	14

Source: DIRS 155970-DOE 2002, Section J.3.1.3, Tables J-47 and J-48.
Note: Impacts are based on 53,000 truck shipments.
I = Interstate Highway.
U.S. = U.S. Highway.

Table A-7. Comparison of Nevada impacts from the mostly truck scenario routing sensitivity analyses in the Yucca Mountain FEIS.

Impact	Base Case	Case 1 Barstow via Nevada 160	Case 2 Barstow via California 127	Case 3 Needles via Nevada 160	Case 4 Needles via U.S. 95	Case 5 Wendover via U.S. 95	Case 6 Wendover via Las Vegas Beltway	Case 7 I-15 and U.S. 95 (Spaghetti Bowl)
Public incident-free dose (person-rem)	340	180	35	170	83	360	490	480
Occupational incident-free dose (person-rem)	1,900	1,800	1,200	1,800	1,400	3,400	3,500	1,900
Nonradioactive pollution health effects	0.09	0.01	< 0.005	0.01	< 0.005	0.03	0.04	0.21
Public incident-free risk of latent cancer fatality	0.17	0.09	0.02	0.08	0.04	0.18	0.24	0.24
Occupational incident-free risk of latent cancer fatality	0.75	0.72	0.47	0.7	0.54	1.4	1.4	0.74
Radiological accident risk (person-rem)	0.052	0.005	0.002	0.004	0.002	0.015	0.027	0.11
Radiological accident risk of latent cancer fatality	0.000026	0.000003	0.000001	0.000002	0.000001	0.000008	0.000013	0.000055
Traffic fatalities	0.5	0.4	0.1	0.4	0.2	1.3	1.3	0.5
Total fatalities	1.5	1.2	0.60	1.2	0.79	2.9	3.0	1.7

Source: DIRS 155970-DOE 2002, Section J.3.1.3, Tables J-47 and J-48.
Note: Impacts are based on 53,000 truck shipments.
I = Interstate Highway.
U.S. = U.S. Highway.

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Appendix B

Nonradiological Air Quality

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B. NONRADIOLOGICAL AIR QUALITY

Potential releases of nonradiological pollutants during the construction, operation and monitoring, and closure of the proposed Yucca Mountain Repository could affect the air quality in the surrounding region. This appendix discusses the methods, data, and intermediate results the U.S. Department of Energy (DOE or the Department) used to estimate impacts from potential nonradiological releases to air for this *Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F-S1) (Repository SEIS). Chapter 4, Section 4.1.2 presents results for the Proposed Action.

Nonradiological pollutants can be categorized as hazardous and toxic air pollutants, criteria pollutants, or other substances of particular interest. Repository activities would cause the release of no or small quantities of hazardous and toxic pollutants; therefore, DOE did not consider these pollutants in the analysis. The National Ambient Air Quality Standards (40 CFR Part 50), which were established by the *Clean Air Act*, regulate concentrations of six criteria pollutants. This analysis quantitatively evaluated releases and potential impacts of four of these pollutants—carbon monoxide, nitrogen dioxide, sulfur dioxide, and particulate matter. Particulate matter has two categories: $PM_{2.5}$, particulate matter with an aerodynamic diameter of 2.5 micrometers or less, and $PM_{10,}$ particulate matter with an aerodynamic diameter of 10 micrometers or less. Sources of $PM_{2.5}$ include smoke, power plants, and gasoline and diesel engines; sources of PM_{10} include dust and gasoline and diesel engine exhaust emissions. The analysis considered the two other criteria pollutants—lead and ozone. It also considered potential releases to air of cristobalite, a form of crystalline silica that can cause silicosis and is a potential carcinogen. Workers could encounter erionite, an uncommon zeolite mineral, during underground construction, but it appears to be absent or rare at the proposed repository depth and location. Erionite would not affect air quality in the area around the repository, and DOE did not consider it in the analysis. Releases of these pollutants could occur during all project analytical periods.

Section B.1 discusses the regulatory limits for criteria pollutants and cristobalite. Section B.2 discusses the models and computer programs DOE used to estimate impacts to nonradiological air quality, and Section B.3 describes the selection of maximally exposed individuals and their locations. Section B.4 discusses meteorological data and reference concentrations of pollutants for analysis. Sections B.5 through B.7 describe the sources of pollutants and the impacts to air quality for the four analyzed activity periods of the proposed repository: construction (B.5), operations (B.6), monitoring (B.6), and closure (B.7). Section B.8 describes the sources of pollutants and the impacts to air quality from construction and operation of the proposed railroad and associated facilities. Section B.9 describes the sources of greenhouse gases, primarily carbon dioxide, during construction and operation of the proposed repository.

B.1 Regulatory Limits

Table B-1 lists the six criteria pollutants that the U.S. Environmental Protection Agency (EPA) and the State of Nevada regulate under the National Ambient Air Quality Standards or the Nevada Administrative Code along with their regulatory limits and the periods during which DOE averaged pollutant concentrations. The criteria pollutants that this section of the appendix addresses quantitatively are nitrogen dioxide, sulfur dioxide, particulate matter (both PM_{10} and $PM_{2.5}$), and carbon monoxide.

Table B-1. Criteria pollutants and regulatory limits.

Pollutant	Averaging period	NAAQS regulatory standards: Parts per million	NAAQS regulatory standards: Micrograms per cubic meter	Nevada standards
Nitrogen dioxide	Annual	0.053	100	Same
Sulfur dioxide	Annual	0.03	80	Same
	24-hour	0.14	365	Same
	3-hour[a]	0.5	1,300	Same
Carbon monoxide	8-hour	9	10,000	Same[b]
	1-hour	35	40,000	Same
PM_{10}	24-hour	(c)	150	Same
$PM_{2.5}$	Annual	(c)	15	None
	24-hour[d]	(c)	35	None
Ozone	8-hour	0.075	(e)	None
	1-hour[f]	None	None	0.12 ppm
Lead	Quarterly	(c)	1.5	Same

Sources: 40 CFR Part 50 and Nevada Administrative Code 445B.22097.

a. Secondary standard.
b. The Nevada ambient air quality standard for carbon monoxide is 9 parts per million at less than 5,000 feet above mean sea level and 6 parts per million at or above 5,000 feet.
c. Standard only reported as micrograms per cubic meter.
d. Effective December 17, 2006.
e. Standard only reported as parts per million.
f. Applies only to the 14 8-hour ozone nonattainment Early Action Compact Areas. Does not apply at Yucca Mountain.

NAAQS = National Ambient Air Quality Standards.
ppm = parts per million.

Because there would be no significant sources of airborne lead at the repository, the analysis did not perform a quantitative assessment of that pollutant. Although lead emissions can occur from concrete batch facilities, the amount of lead from concrete batching released at the Yucca Mountain site would be less than 0.40 kilogram (0.88 pound) per year. The de minimis level (the minimum threshold) for lead is 25 tons per year for conformity determination.

In addition, DOE considered ozone also but did not assess it quantitatively. The purpose of the ozone standard is to control the ambient concentration of ground-level ozone rather than the naturally occurring ozone in the upper atmosphere. Ozone is not emitted directly into the atmosphere; rather, it is created by complex chemical reactions of precursor pollutants in the presence of sunlight. The precursor pollutants are volatile organic compounds and nitrogen oxides (including nitrogen dioxide). DOE's analysis of ozone evaluated the emissions of these precursors. The major source for volatile organic compounds and nitrogen dioxide is the burning of fossil fuels. The maximum annual fuel use under the Proposed Action would be about 1.1 percent of the total diesel fuel use and about 0.021 percent of the total gasoline use in Nevada in 2004. Because about half of the State of Nevada fossil-fuel consumption is in the three-county region of Clark, Lincoln, and Nye counties (DIRS 155970-DOE 2002, p. 4-76), the maximum annual fuel use under the Proposed Action would be about 2.2 percent of the diesel fuel and about 0.04 percent of the gasoline use in those three counties in 2004.

The peak annual release of volatile organic compounds from the burning of fossil fuels would occur during the first 5 years of the operations analytical period and would be about 13,700 kilograms (30,000 pounds) (Section B.6). Because Yucca Mountain is in an attainment area for ozone, the analysis compared the estimated annual release of volatile organic compounds with the Prevention of Significant

Deterioration of Air Quality emission threshold for volatile organic compounds for stationary sources (40 CFR 52.21). The peak annual release would be well below the emission threshold of 36,000 kilograms (80,000 pounds) per year. The maximum annual concentration of nitrogen dioxide at the boundary of the analyzed land withdrawal area from the burning of fossil fuels during the operations analytical period would be about 0.11 percent of the regulatory limit. The annual emissions would be about 10 percent of the total estimated nitrogen dioxide emissions of 1.3 million kilograms (1,400 tons) in Nye County during 2002 (DIRS 177709-EPA 2006, all). About 80 percent of the existing Nye County nitrogen dioxide emissions are the result of onroad automobile and truck sources. Emissions of nitrogen dioxide due to the Proposed Action would be relatively small in comparison with the existing yearly emissions in Nye County. DOE anticipates that the impact of the ozone precursors, volatile organic compounds, and nitrogen dioxide would not cause violations of the ozone standard.

EPA revised the air quality standards for particulate matter in 2006 (40 CFR Part 50). For $PM_{2.5}$, the 2006 standards tightened the 24-hour regulatory limit from 65 to 35 micrograms per cubic meter and retained the annual regulatory limit at 15 micrograms per cubic meter. For PM_{10}, the 2006 standards retained the 24-hour regulatory limit of 150 micrograms per cubic meter but revoked the annual PM_{10} standard. EPA revoked this standard because available evidence does not suggest a link between long-term exposure to PM_{10} and health problems. The new standards took effect on December 17, 2006.

EPA withdrew the 1-hour average primary and secondary standards of 0.12 parts per million for ozone in 2005 and replaced them with 8-hour average standards of 0.08 parts per million. On March 12, 2008, the EPA revised these primary and secondary 8-hour ozone standards from 0.08 parts per million to 0.075 parts per million. The final rule was published in the *Federal Register* on March 27, 2008 (73 FR 16436), to be effective on May 27, 2008.

Cristobalite, one of several naturally occurring crystalline forms of silica (silicon dioxide), is a major mineral constituent of Yucca Mountain tuffs (DIRS 155970-DOE 2002, p. G-2). Prolonged high exposure to crystalline silica might cause silicosis, a disease characterized by scarring of lung tissue. Further, the World Health Organization lists crystalline silica as a carcinogen. Cristobalite is principally a concern for involved workers who could inhale it during subsurface excavation operations. This discussion incorporates by reference Appendix F, Section F.1.2 of the *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F; DIRS 155970-DOE 2002, pp. F-12 to F-14) (Yucca Mountain FEIS), which contains additional information on crystalline silica.

There are no limits for exposure of the general public to cristobalite. Consistent with the analysis in the Yucca Mountain FEIS (DIRS 155970-DOE 2002, p. G-3), the analysis for this Repository SEIS used a comparative benchmark of 10 micrograms per cubic meter based on a cumulative lifetime exposure calculated as 1,000 micrograms per cubic meter multiplied by years. At this level, an EPA health assessment (DIRS 103243-EPA 1996, pp. 1-5 and 7-5) states that there is a less than 1-percent chance of silicosis. Over a 70-year lifetime, this cumulative exposure benchmark would correspond to an annual average exposure concentration of about 14 micrograms per cubic meter, which DOE rounded down to 10 micrograms per cubic meter to establish a more conservative benchmark (DIRS 155970-DOE 2002, p. G-3). Additional studies of occupational exposure to respirable crystalline silica, which used higher concentration levels, have produced results that are consistent with the EPA health assessment. These studies predict that approximately 1 to 7 silicosis cases per 100 workers would occur at respirable quartz concentrations of 25 micrograms per cubic meter (DIRS 176528-CDC 2002, p. 24). This concentration was 2.5 times the benchmark level. Because the studies have shown that doubling the concentration of respirable dust can produce greater than 4 times the incidences of silicosis (DIRS 176528-CDC 2002, p. 25), the prediction of 1 to 7 silicosis cases per 100 workers is consistent with the EPA health assessment.

Members of the public and surface workers could be exposed to cristobalite. The sources of cristobalite releases would include fugitive dust from the excavated rock pile and dust emissions from subsurface excavation via exhaust ventilation. Fugitive dust from the rock pile would be the larger source. DOE would perform evaluations of airborne crystalline silica at Yucca Mountain during routine operations and tunneling. For this analysis, DOE assumed that 28 percent of the fugitive dust from the rock pile and from subsurface excavation would be cristobalite, which reflects the cristobalite content of the parent rock, which ranges from 18 to 28 percent (DIRS 104523-CRWMS M&O 1999, p. 4-81). Use of the parent rock percentage overestimates the airborne cristobalite concentration; studies of both ambient and occupational airborne crystalline silica have shown that most of this airborne material is coarse and not respirable and that larger particles deposit rapidly on the surface (DIRS 103243-EPA 1996, p. 3-26).

B.2 Computer Modeling and Analysis

DOE used the American Meteorological Society/EPA Regulatory Model (AERMOD) computer program, version 07026, to estimate the annual and short-term (24-hour or less) air quality impacts at the proposed repository. The Yucca Mountain FEIS used the Industrial Source Complex (ISC) computer model to estimate air quality impacts. The change in models occurred because EPA established AERMOD as the preferred air dispersion model in place of the Industrial Source Complex model (40 CFR Part 51, Appendix W). In addition, the AERMOD computer program provides better characterization of plume dispersion. The regulation became effective December 9, 2005.

The AERMOD model is a state-of-the-practice Gaussian plume dispersion model for assessment of pollutant concentrations from a variety of sources. It simulates transport and dispersion from sources by using an up-to-date characterization of the atmospheric boundary layer. The model uses hourly, sequential, preprocessed meteorological data to estimate concentrations for averaging times that range from 1 hour to 1 year. The program is appropriate for simple or complex terrain, and for urban or rural environments (40 CFR Part 51). It can handle multiple sources that include point, volume, and area source types. Users can model line sources as elongated area sources and define multiple receptor locations. The analysis used the AERMOD Terrain Preprocessor (AERMAP), version 06341, to prepare terrain inputs for AERMOD. AERMOD used two meteorological files during its calculations: one file defined surface boundary layer parameters, and the second defined profile variables such as wind speed, wind direction, and turbulence parameters. The AERMOD meteorological preprocessor (AERMET), version 06341, generated these meteorological inputs, which are from hourly National Weather Service surface meteorological data, twice-daily upper air data, and local surface meteorological data (DIRS 181091-EPA 2004, all).

Because DOE based the short-term pollutant concentrations on annual use or release parameters, conversion of annual parameter values to short-term values depended on the duration of the activity. The Department assumed that many repository activities would have a schedule of 250 working days per year, so the daily release would be the annual value divided by 250.

In many cases, site- or activity-specific information was not available for estimates of pollutant emissions at the Yucca Mountain site. In these cases, DOE used generic information and made conservative assumptions that tended to overestimate actual air concentrations.

Chapter 4, Section 4.1.2 summarizes total nonradiological air quality impacts for the Proposed Action. Consistent with the analysis established in the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-3 and G-4), the impacts are the sum of air quality impacts from individual sources and activities that would occur during each analyzed activity period. Individual sources and activities are described in Sections B.5 to B.7. The maximum air quality impact (that is, maximum criteria pollutant concentration) from individual sources or activities could occur at different locations around the analyzed land withdrawal area boundary, depending on the release period and the regulatory averaging time (Section B.4). These maximums would generally occur in a westerly or southerly direction due to the prevailing winds in the area. The total nonradiological air quality impacts in Section 4.1.2 are the sum of the calculated maximum concentrations regardless of direction. Therefore, the values are larger than the actual sum of the concentrations would be for a particular distance and direction. DOE selected this approach to simplify the presentation of air quality results and produce the most conservative results.

B.3 Locations of Exposed Individuals

DOE determined the locations of the public hypothetically exposed individuals by calculating the maximum ground-level pollutant concentrations. Because the public would have access only to the site boundary, the analysis followed the methodology DOE established in Appendix G of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, p. G-4) and assumed that a hypothetical individual would be present at one point on the site boundary during the entire averaging time of the regulatory limit (Table B-1).

Table B-2 lists the approximate distances from the North and South Portals to the analyzed land withdrawal area boundary, where the analysis evaluated maximally exposed individual locations. The table does not list all directions because the land withdrawal area boundaries would not be accessible to

Table B-2. Distance to the nearest point of unrestricted public access.

	From North Portal		From South Portal	
Direction	(kilometers)	(miles)	(kilometers)	(miles)
Northwest	14	8.7	15	9.3
West-northwest	12	7.5	12	7.5
West	11	6.8	11	6.8
West-southwest	14	8.7	12	7.5
Southwest	18	11	16	9.9
South-southwest	23	14	19	12
South	21	13	18	11
South-southeast	21	13	19	12
Southeast	22	14	24	15

Source: DIRS 155970-DOE 2002, Table G-2.
Note: Numbers are rounded to two significant figures.

members of the public in some directions (restricted access areas of the Nevada Test Site and the Nevada Test and Training Range). The distance to the nearest unrestricted public access in these directions would be so large that there would be no air quality impacts to the public. For the east to south-southeast directions, the distances to the land withdrawal area boundary would be large, but the terrain is such that plumes that travel in these directions tend to enter Fortymile Wash and turn south. The southern land withdrawal area boundary would be the location of a maximally exposed individual with long-term (1-year) unrestricted access, such as a resident. The short-term (1- to 24-hour) maximally exposed individual location could be the western land withdrawal area boundary, the potential location of an individual such as a hiker or hunter. No long-term access (that is, residency) could occur at this location on government-owned land. The analysis based the evaluated access periods on the exposure periods in Table B-1.

The potential location of the maximally exposed individual member of the public for surface construction outside the analyzed land withdrawal area boundary would not be at the boundary of the area. The maximally exposed person would be adjacent to the offsite construction. The analysis assumed that this individual would be 100 meters (330 feet) from the construction activities. Although 40 CFR Part 51, Appendix W does not specify an optimum receptor location, a fence line around the construction activity or the distance to the nearest building or residence is often assumed to be the closest possible location for a member of the public. Because DOE can only approximate the exact locations of construction activities and the distances to the surrounding fence lines at this time, the analysis used the approximate distance (100 meters) between existing buildings and U.S. Highway 95 as the distance between construction activities and the maximally exposed individual.

B.4 Meteorological Data and Reference Concentrations

DOE used the AERMOD computer program to estimate the concentrations of the criteria pollutants in the region of the repository. The simulations used surface and upper air meteorological data from the National Weather Service station at Desert Rock, Nevada, and onsite surface meteorological data from the meteorological station at Fortymile Wash (YMP5). DOE used meteorological station YMP5 for AERMOD simulations because the analysis calculated emission concentrations not only for activities at the repository surface facilities but also for additional activities within the analyzed land withdrawal area and for construction activities outside the land withdrawal area. Meteorological station YMP5 best represents the meteorological data for all activities inside and outside the land withdrawal area. The most recent meteorological data that are readily available to the public for Desert Rock, Nevada, are for 1984 to 1992. DOE was able to assemble a 4-year meteorological record for 1987, 1988, 1989, and 1990 of hourly data from both the National Weather Service and the onsite meteorological station. DOE preprocessed those data with AERMET for input into AERMOD.

Desert Rock is near Mercury, Nevada, approximately 44 kilometers (27 miles) east-southeast from both the geologic repository operations area and the North Construction Portal facilities. DOE used surface meteorological data from the Desert Rock station in the analysis because of its complete hourly weather data, which include cloud cover and ceiling height. This information was not available for climate stations at Yucca Mountain. DOE used onsite data from Yucca Mountain for site-specific temperature, relative humidity, wind direction, wind speed, and precipitation.

The analysis used the methodology in Section G.1.3 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-5 and G-6) and estimated unit release concentrations at the land withdrawal area boundary points of maximum exposure for ground-level release sources. The concentrations were based on release rates of 1 gram (0.04 ounce) per second for each of the five regulatory limit averaging times (annual, 24-hour, 8-hour, 3-hour, and 1-hour). Activities at the Yucca Mountain site during the construction analytical period could result in releases of pollutants over four periods in a 24-hour day [continuously, 8 hours, 12 hours (two 6-hour periods), and 3 hours]. Eleven combinations of release periods and regulatory limit averaging times would be applicable to activities at the Yucca Mountain site.

The analysis assumed that the 8-hour pollutant releases would occur from 8 a.m. to 4 p.m. and would be zero for all other hours of the day. Similarly, it assumed that that the 3-hour pollutant releases would occur from 9 a.m. to 12 p.m. and would be zero for all other hours. The 12-hour release would occur over two 6-hour periods, assumed to be from 9 a.m. to 3 p.m. and from 5 p.m. to 11 p.m.; other hours would have zero release. Continuous releases would occur throughout the 24-hour day. The estimates of all annual average concentrations assumed the releases were continuous over the year.

Table B-3 lists the maximum unit release concentrations for the 11 combinations of the site-specific release periods and regulatory limit averaging times. The AERMOD analysis used the meteorological data during a single year from 1987 through 1990 that would result in the highest unit concentration to estimate the unit concentrations and directions. Table B-3 lists the 24-hour averaged concentration for the 3- and 12-hour release scenarios because the activities of these scenarios would release only PM_{10}, which has a 24-hour regulatory limit.

Table B-3. Unit release concentrations (micrograms per cubic meter based on a release of 1 gram per second) for maximally exposed individual locations for 11 combinations of four release periods and five regulatory limit averaging times.

Release	South Portal development area	Surface geologic repository operations area and vicinity	Other locations in land withdrawal area (including access road and Gate 510)
Continuous – annual average concentration	0.025	0.027	0.0053
Continuous – 24-hour average concentration	1.6	1.2	0.10
Continuous – 8-hour average concentration	3.7	2.7	0.31
Continuous – 3-hour average concentration	6.9	4.6	0.82
Continuous – 1-hour average concentration	21	10	2.5
8-hour (8 a.m. to 4 p.m.) – 24-hour average concentration	0.86	0.41	0.10
8-hour (8 a.m. to 4 p.m.) – 8-hour average concentration	2.6	1.2	0.31
8-hour (8 a.m. to 4 p.m.) – 3-hour average concentration	6.9	3.1	0.82
8-hour (8 a.m. to 4 p.m.) – 1-hour average concentration	21	9.2	2.5
12-hour (9 a.m. to 3 p.m. and 5 p.m. to 11 p.m.) – 24-hour average concentration	1.1	0.82	0.087
3-hour (9 a.m. to 12 p.m.) – 24-hour average concentration	0.19	0.38	0.086

Note: Numbers are rounded to two significant figures.

Table B-3 lists the maximum unit release concentrations for activities at the South Portal development area and the surface geologic repository operations area and vicinity. The other locations represent construction activities that include the main access road, primary roads, borrow pits, and infrastructure power lines in the land withdrawal area.

Table B-4 lists the unit release concentrations for construction outside the analyzed land withdrawal area near the access road intersection with U.S. Highway 95. It represents activities that include a U.S. Highway 95 intersection, an offsite Sample Management Facility, and other disturbed land outside the land withdrawal area. DOE calculated the unit release concentrations at 100 meters (330 feet) from the construction activity (Section B.3). The emissions from this location would primarily be criteria pollutants from the burning of fossil fuel and PM_{10} from disturbed land.

Table B-4. Unit release concentrations (micrograms per cubic meter based on a release of 1 gram per second) and direction to maximally exposed individual locations for receptors 100 meters from surface construction activities outside the analyzed land withdrawal area.

Release	Direction from construction	Unit release concentration outside land withdrawal area
Continuous – annual average concentration	South	13
Continuous – 24-hour average concentration	South	82
Continuous – 8-hour average concentration	South	170
Continuous – 3-hour average concentration	South	300
Continuous – 1-hour average concentration	South	860
8-hour (8 a.m. to 4 p.m.) – 24-hour average concentration	East	27
8-hour (8 a.m. to 4 p.m.) – 8-hour average concentration	South	73
8-hour (8 a.m. to 4 p.m.) – 3-hour average concentration	East	200
8-hour (8 a.m. to 4 p.m.) – 1-hour average concentration	South	580
12-hour (9 a.m. to 3 p.m. and 5 p.m. to 11 p.m.) – 24-hour average concentration	South	40
3-hour (9 a.m. to 12 p.m.) – 24-hour average concentration	South	4.7

Note: Numbers are rounded to two significant figures.

Using the unit release concentration information listed in Tables B-3 and B-4, DOE calculated the estimated criteria pollutant concentrations for each source or activity (that is, the air quality impact) by multiplying the maximum unit release concentration for each averaging period by the estimated source release rate. DOE chose the maximum unit release concentration regardless of receptor direction or source location (that is, South Portal, North Portal, or other onsite location) because this is the most conservative approach. The following sections describe the source release rates and impacts for each period of activity.

B.5 Construction Analytical Period

This section describes the methods DOE used to estimate air quality impacts during the construction analytical period. The Department would begin construction of surface facilities and would complete sufficient excavation of the subsurface to support initial emplacement activities during this period.

Consistent with the methodology in Appendix G of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-1 to G-44), this analysis used calculations of the pollutant concentrations from various construction activities at the proposed repository to determine air quality impacts. To calculate impacts, DOE multiplied the estimated pollutant emission rates by the maximum unit release concentration for each averaging period (Section B.4). This produced the pollutant concentration for comparison to regulatory limits. The Department estimated short-term pollutant emission rates and concentrations using the method described in Section B.2.

The principal emission sources of PM_{10} would be fugitive dust from construction activities on the surface, excavation of rock from the repository, storage of material in the excavated rock pile, and dust emissions from concrete batch facilities. The principal sources of carbon monoxide, nitrogen dioxide, sulfur dioxide, and $PM_{2.5}$ would be fuel combustion in construction equipment and other surface vehicles. The following sections describe these sources in more detail.

B.5.1 FUGITIVE DUST EMISSIONS FROM SURFACE CONSTRUCTION

Construction activities such as earthmoving and truck traffic would generate fugitive dust. For this analysis, and consistent with the methodology in the Yucca Mountain FEIS, DOE assumed that all surface construction activities and associated fugitive dust releases would occur during 250 working days per year with one 8-hour shift per day. The EPA-preferred method would be to break the construction activities into their component activities (for example, earthmoving and truck traffic) and calculate the emissions for each component. However, information to that detail was not available for the construction analytical period, so DOE took a generic, conservative approach similar to that in the Yucca Mountain FEIS. The estimated release rate of total suspended particulates (particulates with aerodynamic diameters of 30 micrometers or less) would be 0.27 kilogram per square meter (1.2 tons per acre) per month (DIRS 101824-EPA 1995, pp. 13.2.3-1 to 13.2.3-7). The Department based this estimated rate on measurements from the construction of apartment buildings and shopping centers.

Although the estimated release rate of total suspended particulates would be 0.27 kilogram per square meter (1.2 tons per acre) per month, the amount of PM_{10} emissions would be less than that amount. Many of the total suspended particulates from construction would be in the 10- to 30-micrometer range and would tend to settle rapidly (DIRS 102180-Seinfeld 1986, pp. 26 to 31). Experiments on dust emission due to construction found that at 50 meters (160 feet) downwind of the source, a maximum of 30 percent of the remaining suspended particulates at respirable height were in the PM_{10} range (DIRS 103678-Midwest Research Institute 1988, pp. 22 to 26). Based on this factor, only 30 percent of the 0.27 kilogram per square meter per month of total suspended particulates, or 0.081 kilogram per square meter (0.36 ton per acre) per month, would be emitted as PM_{10} from construction activities. Because DOE based the default emission rate on continuous emissions over 30 days, the daily PM_{10} emission rate would be 0.0027 kilogram per square meter per day (0.012 ton per acre), or 0.00011 kilogram per square meter (0.00050 ton per acre) per hour. Although normal dust suppression activities would reduce PM_{10} emissions, the analysis took no credit for such activities.

The estimation of the annual and 24-hour average PM_{10} emission rates required an estimate of the size of the area DOE would disturb along with the unit area emission rate [0.00011 kilogram per square meter (0.00050 ton per acre) per hour] times 8 hours of construction per day. The analysis assumed that site preparation activities during the construction analytical period would disturb the entire land area required for construction at the surface geologic repository operations area and vicinity and the South Portal development area, even though DOE would not build all facilities during that period. The analysis estimated that 20 percent of the total disturbed land area would be actively involved in construction activities at any given time; this was based on the total disturbed area at the end of the construction period divided by the 5 years that construction activities would last. Table B-5 lists the total area of disturbance at repository operations areas. Similarly, the analysis assumed that storage preparation activities would

Table B-5. Land area [square kilometers (acres)] disturbed during the construction analytical period.

Operations area	Disturbed land
North and South portal areas	
North Portal site	2.76 (680)
Topsoil storage location near North Portal site	0.061 (15)
North Portal site ancillary support facilities	0.14 (35)
North Portal site protective forces administrative facility	0.081 (20)
Aging pads	0.57 (140)
Subsurface intake/exhaust shafts (and access roads)	0.243 (60)
South Portal area	0.081 (20)
Excavated rock pile (muck storage)	0.81 (200)
Rail Equipment Maintenance Yard and associated rail facilities	0.405 (100)
Other: In land withdrawal area	
Main access road	2.27 (560)
Gate 510 security complex	0.11 (27)
Primary roads	0.405 (100)
Aggregate quarry/engineered fill quarry	0.405 (100)
Infrastructure: Power lines	0.12 (30)
Other: Outside land withdrawal area	
Intersection at U.S. Highway 95	0.113 (28)
Disturbed land outside the land withdrawal area	0.26 (64)
Infrastructure: Offsite Sample Management Facility	0.012 (3.0)
Total land disturbance	8.8 (2,200)
Area disturbed per year	1.8 (440)

Source: DIRS 182827-Morton 2007, all.
Note: Totals might differ from sums.

disturb the entire land area required for excavated rock storage (for both the construction and operations analytical periods), although DOE would use only a portion of the area for storage during the construction period. Table B-6 lists fugitive dust emissions from surface construction; Table B-7 lists estimated air quality impacts from fugitive dust as a pollutant concentration and as a percent of the applicable regulatory limit. Because DOE based the calculation of the PM_{10} emissions solely on the area of disturbed land, the calculations are independent of the number, specific location, or type of structures the Department would construct on the disturbed land.

Fugitive dust from construction would produce small PM_{10} concentrations at the analyzed land withdrawal boundary. The maximum 24-hour average concentration of PM_{10} for construction in the land withdrawal area would be less than 20 percent of the regulatory limit. The maximum 24-hour average concentration of PM_{10} for construction outside the land withdrawal area could be approximately 40 percent of the regulatory limit at a receptor distance of 100 meters (330 feet) from the construction source.

B.5.2 FUGITIVE DUST EMISSIONS FROM SUBSURFACE EXCAVATION

The excavation of rock from the repository would release fugitive dust. Consistent with the methodology in the Yucca Mountain FEIS, this analysis assumed that subsurface excavation activities would take place 250 days per year in three 8-hour shifts per day. Excavation would generate dust in the tunnels, some of which would emit to the surface atmosphere through the ventilation system. DOE estimated the amount of dust the ventilation system would emit by using engineering judgment and best available information (DIRS 104494-CRWMS M&O 1998, p. 37). Table B-8 lists the release rates of PM_{10} for excavation

Table B-6. Fugitive dust releases from surface construction (PM_{10}).

Location/period	Pollutant emission (kilograms)[a]	Emission rate (grams per second)
North and South portal areas		
Annual[b]	230,000	7.2
24-hour	910	31[c]
Other: Inside land withdrawal area		
Annual[b]	150,000	4.6
24-hour	580	20[c]
Other: Outside land withdrawal area		
Annual[b]	17,000	0.54
24-hour	68	2.4[c]
Total		
Annual[b]	390,000	12
24-hour	1,600	54[b]

Note: Numbers are rounded to two significant figures; therefore, totals might differ from sums.
a. To convert kilograms to pounds, multiply by 2.2046.
b. NAAQS annual PM_{10} regulatory limit revoked December 17, 2006; therefore, DOE did not consider the annual PM_{10} impact further. The annual pollutant emission is listed here for comparison purposes only.
c. Based on an 8-hour release period.
DOE = U.S. Department of Energy.
NAAQS = National Ambient Air Quality Standards.

Table B-7. Estimated fugitive dust air quality impacts (micrograms per cubic meter) from surface construction (PM_{10}).

Operations area	Period	Maximum concentration[a]	Regulatory limit	Percent of limit[a]
North and South portal areas (receptors at boundary of land withdrawal area)	24-hour	27	150	18
Other: In land withdrawal area (receptors at boundary of land withdrawal area)	24-hour	2.1	150	1.4
Other: Outside land withdrawal area [receptors 100 meters (330 feet) from construction activity]	24-hour	64	150	43

a. Numbers are rounded to two significant figures.

Table B-8. Fugitive dust (PM_{10}) releases from excavation activities.

Period	Emission (kilograms)[a]	Emission rate (grams per second)
Annual	920	0.029
24-hour	3.7	0.043[b]

Source: DIRS 155970-DOE 2002, Table G-7; amount of rock excavated by the Proposed Action is within the range evaluated by the Yucca Mountain FEIS.
Note: Numbers are rounded to two significant figures.
a. To convert kilograms to pounds, multiply by 2.2046.
b. Based on a 24-hour release period.

activities. Table B-9 lists estimated air quality impacts from fugitive dust as a pollutant concentration in air and as a percentage of the regulatory limit.

Table B-9. Fugitive dust (PM_{10}) and cristobalite air quality impacts (micrograms per cubic meter) from excavation activities.

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
PM_{10}	24-hour	0.067	150	0.045
Cristobalite	Annual	0.00022	10[b]	0.0022

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.
b. This value is a benchmark; there is no regulatory limit for exposure of cristobalite to the general public (Section B.1).

Fugitive dust emissions from excavation would produce small offsite PM_{10} concentrations. The maximum 24-hour average concentration of PM_{10} would be less than 0.05 percent of the regulatory standard.

Dust from excavation would contain cristobalite, a form of crystalline silica that occurs naturally in Yucca Mountain tuffs. The analysis estimated the annual amounts of cristobalite releases by multiplying the amount of released dust (Table B-8) by the percentage of cristobalite in the parent rock (28 percent). Table B-9 lists potential air quality impacts for releases of cristobalite from excavation of the repository. Because there are no public exposure limits for cristobalite, DOE compared the annual average concentration to a derived benchmark level for the prevention of silicosis (Section B.1). The offsite cristobalite concentration would be less than 0.003 percent of this benchmark.

B.5.3 FUGITIVE DUST FROM EXCAVATED ROCK PILE

The storage of rock from the repository on the excavated rock pile would generate fugitive dust. The unloading of the rock and subsequent smoothing of the rock pile, as well as wind erosion, would release dust. Consistent with the methodology in the Yucca Mountain FEIS, DOE used the total suspended particulate emission for active storage piles to estimate fugitive dust emission. The equation is:

$$E = 1.9 \times (s \div 1.5) \times [(365 - p) \div 235] \times (f \div 15) \qquad \text{(Equation B-1)}$$

where

- E = total suspended particulate emission factor [kilogram per day per hectare (1 hectare = 0.01 square kilometer = 2.5 acres)]
- s = silt content of aggregate (percent)
- p = number of days per year with 0.25 millimeter (0.0098 inch) or more of precipitation
- f = percentage of time wind speed exceeds 5.4 meters per second (12 miles per hour) at pile height.

This analysis assumed the same variables as those used in Section G.1.4.3 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-9 to G-11): s is equal to 4 percent, based on the average silt content of limestone quarrying material; p is 37.75 days; and f is 16.5 percent. Thus, E is equal to 780 kilograms of total particulates per day per square kilometer (6.9 pounds per day per acre). Using the assumption that only about 50 percent of the total particulates would be PM_{10} (DIRS 103676-Cowherd et al. 1988, pp. 4-17 to 4-37), the emission rate for PM_{10} would be 390 kilograms per day per square kilometer (3.5 pounds per day per acre).

The analysis in this Repository SEIS used the size of the area that would be actively involved in storage and maintenance to estimate fugitive dust from disposal and storage. The unloading of excavated rock and the subsequent contouring of the pile would actively disturb only a portion of the excavated rock pile, and only that portion would be an active source of fugitive dust. The analysis assumed that either natural processes or DOE stabilization measures would stabilize the rest of the rock pile, which would release small amounts of dust. The application of dust suppression measures to the active area of the pile would reduce the calculated releases.

DOE used the calculations in Section G.1.4.3 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-9 and G-10) as the basis of its estimate of the size of the active portion of the excavated rock pile because the amount of excavated rock in the Proposed Action would be within the range of the FEIS analysis. DOE assumed the area of the rock pile would be between 0.26 and 0.28 square kilometer (0.10 to 0.11 square mile), the height of the pile would be between 6 and 8 meters (20 and 26 feet), and the average annual active area would be between 0.10 and 0.11 square kilometer (0.039 and 0.042 square mile). The analysis assumed the maximum release of PM_{10} during construction would be 44 kilograms (97 pounds) per 24-hour period. The emission rate would be 0.51 gram per second.

Table B-10 lists estimated air quality impacts from fugitive dust as a pollutant concentration and as a percent of the applicable regulatory limit. The table also lists potential air quality impacts from releases of cristobalite. The analysis used the same methods as those in Section B.5.2, in which DOE assumes that cristobalite would be 28 percent of the fugitive dust released.

Table B-10. Fugitive dust (PM_{10}) and cristobalite air quality impacts (micrograms per cubic meter) from the excavated rock pile during the construction analytical period.

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
PM_{10}	24-hour	0.80	150	0.53
Cristobalite	Annual	0.0038	10[b]	0.038

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.
b. This value is a benchmark; there is no regulatory limit for exposure of cristobalite to the general public (Section B.1).

Fugitive dust emissions from the excavated rock pile would produce small offsite PM_{10} concentrations. The maximum 24-hour average concentration of PM_{10} would be approximately 0.5 percent of the regulatory standard. The offsite cristobalite concentration would be less than 0.04 percent of the benchmark.

B.5.4 FUGITIVE DUST FROM CONCRETE BATCH FACILITY

During the construction analytical period, three concrete batch plants would emit fugitive dust. Two plants would have a capacity of 190 cubic meters (250 cubic yards) per hour and one would have a capacity of 115 cubic meters (150 cubic yards) per hour. For this analysis and consistent with the methodology in Section G.1.4.4 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-11 and G-12), DOE assumed that the three plants would run 3 hours a day and 250 days per year. The three facilities would have a combined capacity of 495 cubic meters (650 cubic yards) of concrete per hour, 1,500 cubic meters (2,000 cubic yards) per day, and 370,000 cubic meters (480,000 cubic yards) per year. However, the Proposed Action would require an average of only 65,000 cubic meters (85,000 cubic yards) per year, or 260 cubic meters (340 cubic yards) per day during the construction period. Table B-11 lists emission factor estimates for a concrete batch facility (DIRS 182386-EPA 2006, pp. 11.12-4 and 11.12-5).

Table B-11. Dust (PM_{10}) release rates for a concrete batch facility (kilograms per 1,000 kilograms of concrete).[a]

Source/activity	Emission rate
Aggregate transfer	0.0017
Sand transfer	0.00051
Cement unloading to elevated storage silo	0.23
Weight hopper loading	0.0013
Mixer loading (central mix)	0.067

Source: DIRS 182386-EPA 2006, p. 11.12-4.
Notes: EPA updated emission rates in June 2006. Numbers are rounded to two significant figures.
a. To convert kilograms to pounds, multiply by 2.2046.
EPA = U.S. Environmental Protection Agency.

Table B-12 lists the particulate matter emission rates of the concrete batch facilities. The emission rate calculations assume that 1 cubic meter (1.3 cubic yards) of concrete weighs about 2,400 kilograms (5,300 pounds). The maximum concentration of PM_{10} for a 24-hour period during construction would be 6.6 micrograms per cubic meter at the boundary of the land withdrawal area, which is 4.4 percent of the regulatory limit.

Table B-12. Particulate matter (PM_{10}) release rates for concrete batch facilities during the construction analytical period.

Period	Emission (kilograms)[a]	Emission rate (grams per second)
Annual[b]	47,000	1.5
24-hour	190	17[c]

Note: Numbers are rounded to two significant figures.
a. To convert kilograms to pounds, multiply by 2.2046.
b. NAAQS annual PM_{10} regulatory limit revoked December 17, 2006; therefore, DOE did not calculate annual PM_{10} impacts. The annual pollutant emission is listed here for comparison purposes only.
c. Based on a 3-hour release period.
DOE = U.S. Department of Energy.
NAAQS = National Ambient Air Quality Standards.

B.5.5 FUGITIVE DUST FROM EXCAVATED ROCK REMOVAL

Excavated rock from construction of the Exploratory Studies Facility is still at the North Portal. In preparation for construction of the repository, DOE would remove approximately 600,000 cubic meters (800,000 cubic yards) of fill and excavated rock, which the Department would either use during construction or move to an excavated rock pile in the South Portal development area (Chapter 2, Section 2.1.3).

DOE used the emission factor for aggregate handling and storage piles to estimate fugitive dust emission from movement of the excavated rock (DIRS 182386-EPA 2006, all). The equation is:

$$E = k(0.0016)\frac{\left(\frac{U}{2.2}\right)^{1.3}}{\left(\frac{M}{2}\right)^{1.4}} \text{ (kilograms per metric ton)} \qquad \text{(Equation B-2)}$$

where

E = emission factor
k = particle size multiplier (dimensionless)
U = mean wind speed, meters per second
M = material moisture content (percent)
Kilograms per metric ton = 1,000 kilograms.

For this analysis, k is equal to 0.35 for PM_{10} (DIRS 177709-EPA 2006, p. 13.2.4-4), U is equal to 1.8 meters per second (DIRS 155970-DOE 2002, p. 3-15), and M is equal to 3.4 percent (DIRS 177709-EPA 2006, p. 13.2.4-2). Therefore, the emission factor E is equal to 0.000205 kilogram of PM_{10} per kilogram of transferred material (0.41 pound per ton).

Table B-13 lists fugitive dust emissions from the excavated rock pile removal. Table B-14 lists estimated air quality impacts from fugitive dust as the pollutant concentration in air and as the percent of the applicable regulatory limit.

Table B-13. Fugitive dust (PM_{10}) releases from excavated rock pile removal.

Period	Cubic meters of rock moved[a]	Kilograms of rock moved[b,c]	Pollutant emission (kilograms)[b]	Emission rate (grams per second)
Annual[d]	600,000	910,000,000	190,000	5.9
24-hour[e]	2,400	3,700,000	750	26[f]

Note: Numbers are rounded to two significant figures.
a. To convert cubic meters to cubic yards, multiply by 1.3079.
b. To convert kilograms to pounds, multiply by 2.2046.
c. Assumes 1 cubic meter of packed earth weighs 1,522 kilograms.
d. NAAQS annual PM_{10} regulatory limit revoked December 17, 2006; therefore, DOE did not calculate annual PM_{10} impact. The annual pollutant emission is listed here for comparison purposes only.
e. Based on 250 working days per year.
f. Based on an 8-hour release period.
DOE = U.S. Department of Energy.
NAAQS = National Ambient Air Quality Standards.

Table B-14. Fugitive dust (PM_{10}) air quality impacts (micrograms per cubic meter) from excavated rock pile removal during the construction analytical period.

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
PM_{10}	24-hour	22	150	15
Cristobalite	Annual	0.044	10[b]	0.44

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.
b. This value is a benchmark; there is no regulatory limit for exposure of cristobalite to the general public (Section B.1).

B.5.6 EXHAUST EMISSIONS FROM CONSTRUCTION EQUIPMENT

Diesel- and gasoline-powered vehicles and equipment would emit the criteria pollutants carbon monoxide, nitrogen dioxide, sulfur dioxide, and particulate matter (PM_{10} and $PM_{2.5}$) during the construction analytical period. DOE estimated emissions from diesel equipment by applying standard EPA emission rates for nonroad diesel construction equipment to the amount of fuel the equipment would use (DIRS 174089-EPA 2004, all). Because legislation has mandated newer and cleaner diesel equipment after 2003, DOE estimated the emission factors from Tier 3 emissions standards (typically 2006 to 2010 model-year equipment). The emission factors assumed construction equipment with an engine size between 176 and 300 horsepower. The EPA emission rates are in grams per horsepower-hour, so DOE converted liters of diesel fuel to horsepower-hours.

Table B-15 lists the emission rates for an average piece of construction equipment. Table B-16 lists the estimated average amount of fuel that DOE would use per year during the construction analytical period and the equivalent horsepower-hours. Table B-17 lists pollutant releases from construction equipment. Table B-18 lists the air quality impacts from construction equipment emission as the pollutant concentration in air and percent of the applicable regulatory limit.

Table B-15. Pollutant emission rates (grams per horsepower-hour)[a] for construction equipment.

Pollutant	Estimated emission	
	Diesel[b]	Gasoline[c]
Carbon monoxide	0.7475	37.1
Nitrogen dioxide	2.5	4
Sulfur dioxide	0.004964	0.1147
PM_{10}	0.15	0.1565
$PM_{2.5}$	0.1455	0.1565[d]
Hydrocarbons	0.1836	1.9

Note: Assumes the horsepower rating for construction equipment is between 176 and 300 horsepower.
a. To convert grams to ounces, multiply by 0.035274.
b. Source: DIRS 174089-EPA 2004, p. A6.
c. Source: DIRS 182387-EPA 1997, all; DIRS 103679-EPA 1991, pp. II-7-1 and II-7-7.
d. Assumes PM_{10} is 100-percent $PM_{2.5}$.

Table B-16. Average amount of fuel use per year during the construction analytical period and equivalent horsepower-hours.

Location consumed[a]	Diesel (liters)[b]	Diesel (hp-hr)	Gasoline (liters)[b]	Gasoline (hp-hr)
In land withdrawal area	3,500,000	19,000,000	150,000	830,000
Outside land withdrawal area	160,000	870,000	6,900	38,000
Total	3,600,000	20,000,000	160,000	870,000

Note: Numbers rounded to two significant figures; therefore, totals might differ from sums.
a. DOE estimated the amount of fuel use in and outside the land withdrawal area by multiplying the percentage of disturbed land in or outside the area by the total amount of fuel use during the construction period.
b. To convert liters to gallons, multiply by 0.26418.
DOE = U.S. Department of Energy.
hp-hr = horsepower-hour.

B.6 Operations and Monitoring Analytical Periods

This section describes the methods DOE used to estimate air quality impacts during the operations and monitoring analytical periods. The operations period would begin on receipt of a license to receive and possess radiological materials and would last up to 50 years. During the operations period, DOE would complete surface construction Phases 2, 3, and 4; continue subsurface development; and construct and operate the North Construction Portal. These activities would occur while the receipt, handling, aging, emplacement, and monitoring of waste were occurring.

Table B-17. Pollutant release rates from surface equipment during the construction analytical period.

Pollutant	Period	Mass of pollutant per averaging period (kilograms)[a]	Emission rate[b] (grams per second)
Construction in land withdrawal area			
Nitrogen dioxide	Annual	51,000	1.6
Sulfur dioxide	Annual	190	0.0060
	24-hour	0.76	0.026
	3-hour	0.28	0.026
Carbon monoxide	8-hour	180	6.2
	1-hour	22	6.2
PM_{10}	24-hour	12	0.41
$PM_{2.5}$	Annual	2,900	0.092
	24-hour	12	0.40
Construction outside land withdrawal area			
Nitrogen dioxide	Annual	2,300	0.074
Sulfur dioxide	Annual	8.7	0.00028
	24-hour	0.035	0.0012
	3-hour	0.013	0.0012
Carbon monoxide	8-hour	8.3	0.29
	1-hour	1.0	0.29
PM_{10}	24-hour	0.55	0.019
$PM_{2.5}$	Annual	130	0.0042
	24-hour	0.53	0.018

Note: Numbers are rounded to two significant figures.
a. To convert kilograms to pounds, multiply by 2.2046.
b. Based on an 8-hour release for averaging periods of 24 hours or less.

The monitoring analytical period would begin at the completion of the operations analytical period and would continue for 50 years after the emplacement of the final waste package. Activities during the monitoring period would include maintenance of active ventilation for up to 50 years, remote inspections of waste packages, continued investigations to support predictions of postclosure repository performance, and retrieval of waste packages to correct detected problems, if necessary. No construction activities would occur. Due to a major decline in activities during the monitoring period, the impacts to air quality would be much less than those during the construction or operations periods.

For this Repository SEIS and consistent with the methodology in Section G.1.5 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-16 to G-21), workers would use the following schedule for activities during the operations and monitoring analytical periods: three 8-hour shifts a day, 5 days a week, 50 weeks a year. Maintenance of the excavated rock pile would occur in one 8-hour shift a day, 5 days a week, 50 weeks a year.

The analysis estimated air quality impacts by calculating pollution concentrations from operations and monitoring activities. It developed emission rates for each activity that would result in pollutant releases and multiplied the emission rates by the unit release concentrations (Section B.4) to calculate the pollutant concentrations for comparison with regulatory limits.

Table B-18. Air quality impacts from construction equipment during the construction analytical period (micrograms per cubic meter).

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
Construction in land withdrawal area (receptors at boundary of land withdrawal area)				
Nitrogen dioxide	Annual	0.043	100	0.043
Sulfur dioxide	Annual	0.00016	80	0.00020
	24-hour	0.023	365	0.0062
	3-hour	0.18	1,300	0.014
Carbon monoxide	8-hour	16	10,000	0.16
	1-hour	130	40,000	0.32
PM_{10}	24-hour	0.36	150	0.24
$PM_{2.5}$	Annual	0.0024	15	0.016
	24-hour	0.34	35	1.0
Construction outside land withdrawal area [receptors 100 meters (330 feet) from construction activity]				
Nitrogen dioxide	Annual	1.0	100	1.0
Sulfur dioxide	Annual	0.0040	80	0.0051
	24-hour	0.032	365	0.0088
	3-hour	0.24	1,300	0.019
Carbon monoxide	8-hour	21	10,000	0.21
	1-hour	170	40,000	0.42
PM_{10}	24-hour	0.51	150	0.34
$PM_{2.5}$	Annual	0.057	15	0.38
	24-hour	0.49	35	1.4

a. Numbers are rounded to two significant figures.

The principal sources of particulate matter would be dust emissions from surface construction (which would include an aging pad), concrete batch facility operations, excavation, and storage in the excavated rock pile. Surface construction would occur during the first 5 years of the operations analytical period. Emissions from the North Portal boiler, standby generators, and emergency generators would be sources of nitrogen dioxide, sulfur dioxide, carbon monoxide, and $PM_{2.5}$. Fuel combustion from waste handling equipment, surface construction equipment, and equipment to maintain the excavated rock pile would be additional sources of these criteria pollutants. The following sections describe these sources in greater detail.

B.6.1 FUGITIVE DUST FROM SURFACE CONSTRUCTION

Construction of the remaining surface facilities, the North Construction Portal, and the remaining aging pad during the operations analytical period would emit fugitive dust. For this analysis and consistent with the methodology in Section G.1.5 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, p. G-16), DOE assumed that some construction would disturb portions of land already disturbed during the construction analytical period.

This Repository SEIS assumed the disturbance of an equal amount of land every year during the 5 years of surface construction in the operations analytical period. Table B-19 lists the areas surface construction would disturb. The estimated annual amount of land disturbed during the operations period would be about 21 percent of that during the construction analytical period.

Table B-19. Land area (square kilometers)[a] disturbed during the operations analytical period.

Description	Total disturbed land	Percent disturbed during operations period	Land disturbed during operations period	Land disturbed per year during operations period[b]
North Portal site	2.8	50	1.4	0.28
Aging pads	0.57	75	0.43	0.085
Surface geologic repository operations area and vicinity	0.081	100	0.081	0.016
Totals[c]			1.9	0.38

a. To convert square kilometers to acres, multiply by 247.1.

b. Assume that surface construction would occur during only the first 5 years of the operations period and that equal amounts of land would be disturbed during each of those 5 years.

c. Numbers are rounded to two significant figures; therefore, totals might differ from sums.

The estimated PM_{10} emissions and emission rates during the operations analytical period would be 21 percent of the total during the construction analytical period (Section B.5.1, Table B-6) based on the amount of land disturbed. The PM_{10} concentration would be about 3.9 percent of the regulatory limit. Although normal dust suppression activities would reduce PM_{10} emissions, the analysis took no credit for such activities.

B.6.2 FUGITIVE DUST FROM CONCRETE BATCH FACILITY

For this Repository SEIS and consistent with the methodology in Section G.1.5.2 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-16 and G-17), DOE assumed that the concrete batch facilities it used during construction would operate during the first 4 years of the operations analytical period. The Proposed Action would require an average of 41,600 cubic meters (54,000 cubic yards) per year, or 170 cubic meters (220 cubic yards) per day during those 4 years. The dust release rate and potential air quality impacts for the operations period would be about 64 percent of those for the construction analytical period (Section B.5.4). The PM_{10} concentration would be about 2.8 percent of the regulatory limit.

B.6.3 FUGITIVE DUST FROM SUBSURFACE EXCAVATION

This section summarizes and incorporates by reference Section G.1.5.3 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, p. G-17). The excavation of rock from the repository would generate fugitive dust in the drifts and some of the dust would reach the atmosphere through the repository ventilation system. The subsurface excavation activity during the operations analytical period would be similar to the activity during the construction analytical period; thus, fugitive dust emission rates from excavation during operations would be similar to those during the construction period. The fugitive dust release rate and potential air quality impacts for excavation of rock would be the same as those in Section B.5.2 for construction.

Tables B-8 and B-9 list the impacts of fugitive dust from subsurface excavation during construction. Air quality impacts from cristobalite releases during subsurface excavation would be the same as those in Table B-9. The PM_{10} concentration would be 0.045 percent of the regulatory limit, and the cristobalite concentration would be 0.0022 percent of the benchmark.

B.6.4 FUGITIVE DUST FROM EXCAVATED ROCK PILE

The storage of rock on the excavated rock pile would release fugitive dust during the operations analytical period. For this Repository SEIS and consistent with the methodology in Section G.1.5.4 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-17 to G-19), the fugitive dust emissions and release rate would depend on the active area of the excavated rock pile. While the land area DOE would use for storage of excavated rock during the operations period would be nearly twice as large as that used during the construction analytical period, the active area per year would be approximately 50 percent as large due to the larger number of years over which continued development would occur. The annual emissions, emission rate, and maximum concentration of PM_{10} for the operations period would be 50 percent of those for the construction period (Section B.5.3). The PM_{10} concentration would be 0.27 percent of the regulatory limit, and the cristobalite concentration would be 0.019 percent of the benchmark.

B.6.5 EXHAUST EMISSIONS FROM SURFACE EQUIPMENT

Surface equipment would emit carbon monoxide, nitrogen dioxide, sulfur dioxide, and particulate matter during surface operations, excavated rock pile maintenance, and surface facility construction. Consistent with the methodology in Section G.1.5.5 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-19 to G-20), the analysis used the same method to determine air quality impacts from surface equipment during operations as that for construction (Section B.5.6).

During the first 5 years of the operations analytical period, while construction activities were occurring, the annual diesel-fuel use would be 101 percent of that during the construction analytical period. Annual gasoline use during those 5 years would be 488 percent of that during the construction period. The increase in gasoline use would be due to the use of trucks, cars, and four-wheel drive vehicles during operations activities.

After the 5 years of construction activities, the annual diesel-fuel use would be 55 percent of that during construction. The decrease in diesel-fuel use would be a direct result of the completion of surface construction and the associated decrease in the use of construction equipment. Annual gasoline use would be 539 percent of that during the construction analytical period. Gasoline use would not decrease in comparison with the construction period because few construction vehicles would use gasoline and the number of gasoline-powered vehicles for operations would increase after the 5 years of construction.

Table B-20 lists the pollution release rates during the first 5 years of the operations analytical period, when the total amount of release would be greatest. Table B-21 lists the air quality impacts from surface equipment emissions. Because volatile organic compounds are a precursor for ozone production, DOE's analysis of ozone evaluated the quantity of volatile organic compounds emitted annually during the operations period. Approximately 12,000 kilograms (26,000 pounds) of hydrocarbons would be released annually by surface equipment during operations.

Table B-20. Pollutant release rates from surface equipment during the first 5 years of the operations analytical period.

Pollutant	Period	Mass of pollutant per averaging period (kilograms)[a]	Emission rate[b] (grams per second)
Nitrogen dioxide	Annual	67,000	2.1
Sulfur dioxide	Annual	580	0.019
	24-hour	2.3	0.081
	3-hour	0.88	0.081
Carbon monoxide	8-hour	690	24
	1-hour	86	24
PM_{10}	24-hour	15	0.51
$PM_{2.5}$	Annual	3,600	0.11
	24-hour	14	0.50

Note: Numbers are rounded to two significant figures.
a. To convert kilograms to pounds, multiply by 2.2046.
b. Based on an 8-hour release for averaging periods of 24 hours or less.

Table B-21. Air quality impacts from surface equipment during the first 5 years of the operations analytical period (micrograms per cubic meter).

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
Nitrogen dioxide	Annual	0.056	100	0.056
Sulfur dioxide	Annual	0.00049	80	0.00061
	24-hour	0.070	365	0.019
	3-hour	0.56	1,300	0.043
Carbon monoxide	8-hour	61	10,000	0.62
	1-hour	490	40,000	1.2
PM_{10}	24-hour	0.44	150	0.29
$PM_{2.5}$	Annual	0.0030	15	0.020
	24-hour	0.43	35	1.2

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.

B.6.6 EXHAUST EMISSIONS FROM BOILERS AND GENERATORS

Diesel plant heating boilers in the surface geologic repository operations area and vicinity would emit carbon monoxide, nitrogen dioxide, sulfur dioxide, and particulate matter. The basis for the emission calculations was fuel consumption during the 5-year period of increasing operations activities, when the annual total emissions would be greatest for the operations analytical period due to emissions from construction equipment. The boilers would be industrial water tube boilers. Table B-22 lists the emission factors for a commercial/industrial diesel boiler with a size of 10 to 100 million British thermal units per hour (EPA type SCC 1-03-005-02). The diesel boilers would consume an average of 13 million liters (3.4 million gallons) per year during the initial 5-year period and about 17 million liters (4.5 million gallons) per year at full operations. Table B-23 lists pollutant releases by diesel boilers during the operations

Table B-22. Pollutant emission rates for commercial/industrial diesel boiler.

	Estimated emission	
Pollutant	Pounds per 1,000 gallons diesel burned[a]	Kilograms per 1,000 liters diesel burned[b]
Carbon monoxide	5	0.60
Nitrogen dioxide (uncontrolled)	20	2.4
Sulfur dioxide	0.21[c]	0.026
PM_{10}	2.4	0.29
$PM_{2.5}$	2.1	0.26

Source: EPA Factor Information Retrieval (FIRE) software version 6.25.
a. Actual emission factor from EPA FIRE 6.25.
b. Calculated emission factor.
c. Assumes 0.0015 percent sulfur in fuel (15 parts per million).
EPA = U.S. Environmental Protection Agency.

Table B-23. Pollutant release rates from diesel boilers during first 5 years of the operations analytical period.

Pollutant	Period	Mass of pollutant per averaging period (kilograms)[a]	Emission rate[b] (grams per second)
Nitrogen dioxide	Annual	31,000	0.98
Sulfur dioxide	Annual	330	0.010
	24-hour	1.3	0.046
	3-hour	0.49	0.046
Carbon monoxide	8-hour	31	1.1
	1-hour	3.9	1.1
PM_{10}	24-hour	15	0.51
$PM_{2.5}$	Annual	3,300	0.10
	24-hour	13	0.46

Note: Numbers are rounded to two significant figures.
a. To convert kilograms to pounds, multiply by 2.2046.
b. Based on an 8-hour release for averaging periods of 24 hours or less.

period. Table B-24 lists the air quality impacts from boiler emissions. Approximately 860 kilograms (1,900 pounds) of total organic carbon would also be released annually by boilers and would add to the amount of volatile organic compounds released during operations.

The air quality impacts from the boilers during full repository operations would be 130 percent of the results in Tables B-23 and B-24; the boilers' fuel consumption would be 130 percent greater during full operations than during the initial 5-year period. Even though impacts from boilers would be greater during full repository operations, the annual total emissions from all sources would be greater during the 5-year period of increasing operations because of the large quantity of fuel burned by construction vehicles during that period. DOE combined the impact from boiler emissions with impacts from the 5-year period of surface construction to calculate the most conservative combined impact.

Table B-24. Air quality impacts from diesel boilers during the first 5 years of the operations analytical period (micrograms per cubic meter).

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
Nitrogen dioxide	Annual	0.026	100	0.026
Sulfur dioxide	Annual	0.00028	80	0.00035
	24-hour	0.039	365	0.011
	3-hour	0.31	1,300	0.024
Carbon monoxide	8-hour	2.8	10,000	0.028
	1-hour	22	40,000	0.055
PM_{10}	24-hour	0.44	150	0.29
$PM_{2.5}$	Annual	0.0028	15	0.018
	24-hour	0.39	35	1.1

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.

The emergency and standby diesel generators would emit carbon monoxide, nitrogen dioxide, sulfur dioxide, and particulate matter. The analysis assumed that the generators would be 4,500 kilowatts. The basis for the emission calculations would be annual fuel consumption during the operations analytical period. It also assumed that annual diesel-fuel use for the generators would be constant through the operations period and would not be affected by the increasing repository operations during the first 5 years of the period.

Table B-25 lists the emission factors for a large, stationary diesel engine (EPA type SCC 2-02-004-01). Table B-26 lists the amount of fuel consumed per year by the diesel generators. Table B-27 lists pollutant releases by diesel generators during the operations analytical period. In addition, the generators would release approximately 850 kilograms (1,900 pounds) of volatile organic compounds annually. Table B-28 lists the air quality impacts from diesel generator emissions.

Table B-25. Pollutant emission rates for large, stationary diesel engine.

	Estimated emissions	
Pollutant	Pounds per 1,000 gallons diesel burned[a]	Kilograms per 1,000 liters diesel burned[b]
Carbon monoxide	116	14
Nitrogen dioxide (uncontrolled)	438	52
Sulfur dioxide	0.207[c]	0.025
PM_{10}	7.85	0.94
$PM_{2.5}$	7.55	0.90

Source: EPA FIRE software version 6.25.
a. Actual emission factor from EPA FIRE 6.25.
b. Calculated emission factor.
c. Assumes 0.0015 percent sulfur in fuel (15 parts per million).
EPA = U.S. Environmental Protection Agency.

Table B-26. Amount of fuel consumed per year by diesel generators.

Generator type	Fuel use per year (liters)	Fuel use per year (gallons)
Emergency diesel generator	160,000	42,000
Standby diesel generator	670,000	180,000
Total	830,000	220,000

Table B-27. Pollutant release rates from diesel generators during the operations analytical period.

Pollutant	Period	Mass of pollutant per averaging period (kilograms)[a]	Emission rate[b] (grams per second)
Nitrogen dioxide	Annual	44,000	1.4
Sulfur dioxide	Annual	21	0.00066
	24-hour	0.083	0.0029
	3-hour	0.031	0.0029
Carbon monoxide	8-hour	46	1.6
	1-hour	5.8	1.6
PM_{10}	24-hour	3.1	0.11
$PM_{2.5}$	Annual	760	0.024
	24-hour	3.0	0.10

Note: Numbers are rounded to two significant figures.
a. To convert kilograms to pounds, multiply by 2.2046.
b. Based on an 8-hour release for averaging periods of 24 hours or less.

Table B-28. Air quality impacts from diesel generators during the operations analytical period (micrograms per cubic meter).

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
Nitrogen dioxide	Annual	0.037	100	0.037
Sulfur dioxide	Annual	0.000018	80	0.000022
	24-hour	0.0025	365	0.00068
	3-hour	0.020	1,300	0.0015
Carbon monoxide	8-hour	4.2	10,000	0.042
	1-hour	33	40,000	0.083
PM_{10}	24-hour	0.094	150	0.062
$PM_{2.5}$	Annual	0.00063	15	0.0042
	24-hour	0.090	35	0.26

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.

B.7 Closure Analytical Period

This section describes the methods DOE used to estimate air quality impacts during the closure analytical period at the proposed repository. The closure period would last 10 years and would overlap the last 10 years of the monitoring analytical period. Activities during the closure period would include decontamination of the surface handling facilities, backfilling, sealing of subsurface-to-surface openings, construction of monuments to mark the site, decommissioning and demolition of surface facilities, and restoration of the surface to its approximate condition before repository construction.

For this Repository SEIS and consistent with the methodology in Section G.1.6 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-21 to G-25), DOE estimated air quality impacts by calculating pollutant concentrations from closure activities. The analysis developed emission rates for each activity that would result in release of pollutants and then multiplied the rates by the unit release concentration (Section B.4) to calculate the pollutant concentration for comparison with the regulatory limits.

The sources of particulate matter would be emissions from the backfill plant (discussed below in Section B.7.1) and concrete batch facility, fugitive dust from closure activities on the surface, and fugitive dust from the reclamation of material from the excavated rock pile for backfill. The principal source of nitrogen dioxide, sulfur dioxide, and carbon monoxide during closure would be fuel combustion. The following sections describe these sources in more detail.

B.7.1 FUGITIVE DUST FROM BACKFILL ACTIVITIES

This section summarizes, incorporates by reference, and updates Section G.1.6.1 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, p. G-21). DOE assumed that much of the backfill would be processed rock from the excavated rock pile. The rock would be separated, crushed, screened, and washed to enhance the characteristics useful for closure backfill. As much as 91 metric tons (100 tons) an hour would be processed in a facility that would run 6 hours a shift, 2 shifts per day, 5 days a week, 50 weeks a year during the closure analytical period. DOE assumed the PM_{10} release amount would be 12,000 kilograms (26,000 pounds) per year, or 49 kilograms (110 pounds) per 24-hour period. The 24-hour emission rate would be 1.1 grams per second, based on a 12-hour release period. The maximum concentration of PM_{10} would be 1.2 micrograms per cubic meter, which is 0.82 percent of the regulatory limit.

B.7.2 FUGITIVE DUST FROM THE CONCRETE BATCH FACILITY

The design and operational plans included in the application for a construction authorization no longer include the use of concrete during the closure analytical period. Therefore, there would be no additional emissions from a concrete batch plant during this period.

B.7.3 FUGITIVE DUST FROM CLOSURE ACTIVITIES

This section summarizes, incorporates by reference, and updates Section G.1.6.3 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, p. G-23). DOE assumed that closure activities such as smoothing and reshaping of the excavated rock pile and demolition of buildings would produce virtually the same fugitive dust releases as construction activities because they would disturb nearly the same amount of land. However, because the activities would occur over a 10-year period rather than a 5-year period, the annual emissions would be lower. Sources of dust from surface demolition and decommissioning activities would include the North Portal area and roads, South Portal area and roads, ventilation shaft areas and access roads, the excavated rock pile, concrete batch plant, and aging pads. The analysis assumed that closure would not affect sites outside the land withdrawal area such as an intersection near U.S. Highway 95 and an offsite Sample Management Facility. Table B-29 lists PM_{10} release rates. The maximum concentration of PM_{10} would be 22 micrograms per cubic meter, which is 15 percent of the regulatory limit.

B.7.4 FUGITIVE DUST FROM EXCAVATED ROCK PILE

This section summarizes, incorporates by reference, and updates Section G.1.6.4 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-24 and G-25). DOE assumed that fugitive dust would occur from the removal of excavated rock from the rock pile during backfill operations. The amount of excavated rock in the Proposed Action is within the range evaluated by the FEIS. Consistent with Table G-38 in the FEIS, DOE assumed the PM_{10} release amount would be 30 kilograms (66 pounds) per 24-hour period,

Table B-29. Fugitive dust releases from surface demolition and decommissioning (PM_{10}).

Period	Pollutant emission (kilograms)[a]	Emission rate (grams per second)
Annual[b]	190,000	5.9
24-hour	740	26[c]

Notes: Numbers are rounded to two significant figures. Assumes 10 years for closure.
a. To convert kilograms to pounds, multiply by 2.2046.
b. National Ambient Air Quality Standard annual PM_{10} regulatory limit revoked December 17, 2006; therefore, DOE did not consider annual PM_{10} impact further. The annual pollutant emission is listed for comparison purposes only.
c. Based on an 8-hour release period.
DOE = U.S. Department of Energy.

with an emission rate of 0.35 gram per second, based on continuous release. Table B-30 lists PM_{10} air quality impacts from the excavated rock pile. Table B-30 also lists potential air quality impacts for releases of cristobalite. The analysis used the same methods as those in Section B.5.2 for the construction analytical period, in which DOE assumed cristobalite would be 28 percent of the fugitive dust releases, based on its percentage in the parent rock.

Table B-30. Fugitive dust (PM_{10}) and cristobalite air quality impacts (micrograms per cubic meter) from the excavated rock pile during the closure analytical period.

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
PM_{10}	24-hour	0.55	150	0.37
Cristobalite	Annual	0.0026	10[b]	0.026

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.
b. This value is a benchmark; there is no regulatory limit for exposure of cristobalite to the general public (Section B.1).

B.7.5 EXHAUST EMISSIONS FROM SURFACE EQUIPMENT

This section summarizes, incorporates by reference, and updates Section G.1.6.5 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, p. G-25). The consumption of diesel fuel by surface equipment and backfilling equipment would emit carbon monoxide, nitrogen dioxide, sulfur dioxide, and particulate matter (PM_{10} and $PM_{2.5}$) during the closure analytical period. DOE assumed the annual amount of diesel-fuel use during closure would be 2 million liters (530,000 gallons). Table B-31 lists pollutant releases from diesel-fuel use for the combination of surface equipment and backfilling equipment. Table B-32 lists air quality impacts. Exhaust emissions would be substantially less than those during the construction analytical period.

Table B-31. Pollutant release rates from surface and backfilling equipment during the closure analytical period.

Pollutant	Period	Mass of pollutant per averaging period (kilograms)[a,b]	Emission rate[c] (grams per second)
Nitrogen dioxide	Annual	27,000	0.87
Sulfur dioxide	Annual	55	0.0017
	24-hour	0.22	0.0076
	3-hour	0.082	0.0076
Carbon monoxide	8-hour	33	1.1
	1-hour	4.1	1.1
PM_{10}	24-hour	6.6	0.23
$PM_{2.5}$	Annual	1,600	0.051
	24-hour	6.4	0.22

Note: Numbers are rounded to two significant figures.
a. Mass of pollutant was calculated by using diesel emission factors from Table B-15.
b. To convert kilograms to pounds, multiply by 2.2046.
c. Based on an 8-hour release for averaging periods of 24 hours or less.

Table B-32. Air quality impacts from diesel equipment during the closure analytical period (micrograms per cubic meter).

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
Nitrogen dioxide	Annual	0.023	100	0.023
Sulfur dioxide	Annual	0.000045	80	0.000056
	24-hour	0.0065	365	0.0018
	3-hour	0.052	1,300	0.0040
Carbon monoxide	8-hour	2.9	10,000	0.029
	1-hour	24	40,000	0.059
PM_{10}	24-hour	0.20	150	0.13
$PM_{2.5}$	Annual	0.0013	15	0.0090
	24-hour	0.19	35	0.55

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.

B.8 Quantification of Emissions Associated with the Rail Alignment in the Analyzed Land Withdrawal Area

This section describes the methods DOE used to estimate air quality impacts from the railroad in the analyzed land withdrawal area. The Rail Alignment EIS contains a more complete description of the proposed railroad. DOE calculated all air quality concentrations at the boundary of the land withdrawal area.

B.8.1 RAIL CONSTRUCTION: FUGITIVE DUST EMISSIONS DURING THE CONSTRUCTION ANALYTICAL PERIOD

Activities associated with constructing the rail line would generate fugitive dust. Crystalline silica could be present in the rock DOE used as ballast and, thus, in fugitive dust. For this analysis, and consistent with the Rail Alignment EIS, DOE assumed that all rail construction activities and associated fugitive dust releases would occur during a 12-hour workday with 250 working days per year. Estimated PM_{10}

releases in the analyzed land withdrawal area from track construction would be about 160,000 kilograms (350,000 pounds) per year, or 650 kilograms (1,400 pounds) per day. The daily emission rate would be about 15 grams per second. The maximum concentration of PM_{10} at the boundary of the land withdrawal area would be about 57 micrograms per cubic meter, which would be about 38 percent of the regulatory limit. Consistent with the methodology in the Rail Alignment EIS, these estimates assumed a 74-percent best management practice reduction of fugitive dust emissions. The highest maximum concentration of PM_{10} would be at the receptor location along the west boundary of the land withdrawal area. This receptor would be less than 500 meters (1,600 feet) from the rail line.

B.8.2 RAIL CONSTRUCTION: EXHAUST EMISSIONS DURING THE CONSTRUCTION ANALYTICAL PERIOD

Diesel-powered vehicles and equipment would emit the criteria pollutants carbon monoxide, nitrogen dioxide, sulfur dioxide, and particulate matter (both PM_{10} and $PM_{2.5}$) during the construction of the rail line in the analyzed land withdrawal area. DOE based its calculation of emissions on the types of equipment it would use during construction, the number of operating hours for the equipment, and the hourly emission factors. The Department used Tier 1 emission standards to obtain conservative estimates of emissions for rail activities. The highest maximum concentration of all criteria pollutants would be at the receptor location along the west boundary of the land withdrawal area. This receptor would be less than 500 meters (1,600 feet) from the location of the rail line. Table B-33 lists estimated pollutant releases from construction equipment. Table B-34 lists estimated air quality impacts from construction equipment emissions as the pollutant concentration in air and percent of the applicable regulatory limit.

Table B-33. Rail construction pollutant release rates in the analyzed land withdrawal area from surface equipment during the construction analytical period.

Pollutant	Period	Mass of pollutant per averaging period (kilograms)[a]	Emission rate[b] (grams per second)
Nitrogen dioxide	Annual	590,000	19
Sulfur dioxide	Annual	420	0.013
	24-hour	1.7	0.038
	3-hour	0.62	0.038
Carbon monoxide	8-hour	1,800	42
	1-hour	230	42
Carbon dioxide	Annual	44,000,000	1,400
PM_{10}	24-hour	140	3.2
$PM_{2.5}$	Annual	34,000	1.1
	24-hour	140	3.1

Note: Numbers are rounded to two significant figures.
a. To convert kilograms to pounds, multiply by 2.2046.
b. Based on a 12-hour release for averaging periods of 24 hours or less.

Table B-34. Rail construction air quality impacts from construction equipment in the analyzed land withdrawal area during the construction analytical period (micrograms per cubic meter).

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
Nitrogen dioxide	Annual	2.7	100	2.7
Sulfur dioxide	Annual	0.0019	80	0.0024
	24-hour	0.15	365	0.040
	3-hour	0.61	1,300	0.047
Carbon monoxide	8-hour	250	10,000	2.5
	1-hour	2,000	40,000	5.1
PM_{10}	24-hour	12	150	8.2
$PM_{2.5}$	Annual	0.16	15	1.0
	24-hour	12	35	34

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.

B.8.3 RAIL FACILITY CONSTRUCTION: EXHAUST EMISSIONS DURING THE CONSTRUCTION ANALYTICAL PERIOD

Diesel-powered vehicles and equipment would emit the criteria pollutants carbon monoxide, nitrogen dioxide, sulfur dioxide, and particulate matter (both PM_{10} and $PM_{2.5}$) during the construction of the Rail Equipment Maintenance Yard and associated facilities in the land withdrawal area. DOE based its calculation of emissions on the types of equipment it would use during construction, the number of operating hours for the equipment, and the hourly emission factors. The Department used Tier 1 emission standards to obtain conservative estimates of emissions for rail activities. Table B-35 lists estimated pollutant releases from construction equipment. Table B-36 lists estimated air quality impacts from construction equipment emissions as the pollutant concentration in air and percent of the applicable regulatory limit.

Table B-35. Rail Equipment Maintenance Yard pollutant release rates from surface equipment during the construction analytical period.

Pollutant	Period	Mass of pollutant per averaging period (kilograms)[a]	Emission rate[b] (grams per second)
Nitrogen dioxide	Annual	84,000	2.7
Sulfur dioxide	Annual	71	0.0022
	24-hour	0.28	0.0098
	3-hour	0.11	0.0098
Carbon monoxide	8-hour	300	11
	1-hour	38	11
Carbon dioxide	Annual	7,500,000	240
PM_{10}	24-hour	22	0.76
$PM_{2.5}$	Annual	5,300	0.17
	24-hour	21	0.73

Note: Numbers are rounded to two significant figures.
a. To convert kilograms to pounds, multiply by 2.2046.
b. Based on an 8-hour release for averaging periods of 24 hours or less.

Table B-36. Rail Equipment Maintenance Yard air quality impacts from construction equipment during the construction analytical period (micrograms per cubic meter).

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
Nitrogen dioxide	Annual	0.071	100	0.071
Sulfur dioxide	Annual	0.000058	80	0.000073
	24-hour	0.0084	365	0.0023
	3-hour	0.067	1,300	0.0052
Carbon monoxide	8-hour	27	10,000	0.27
	1-hour	220	40,000	0.54
PM_{10}	24-hour	0.65	150	0.43
$PM_{2.5}$	Annual	0.0044	15	0.030
	24-hour	0.63	35	1.8

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.

B.8.4 RAIL FACILITY EMISSIONS DURING OPERATIONS ANALYTICAL PERIOD

Air emissions from rail facilities in the analyzed land withdrawal area would occur during the operations period. They would include emissions from the Rail Equipment Maintenance Yard operations, vehicles, switch train locomotives, and fuel storage tanks. Table B-37 lists annual pollutant releases from these activities. Table B-38 lists air quality impacts from rail facilities and activities.

Table B-37. Annual pollutant emissions (kilograms)[a] from rail facilities and activities during the operations analytical period.

Pollutant	Rail Equipment Maintenance Yard	Rail Equipment Maintenance Yard trucks	Rail Equipment Maintenance Yard switch train locomotives	Fuel oil storage	Total rail facility emissions
Nitrogen dioxide	34,000	170	360,000	0	400,000
Sulfur dioxide	800	1.0	210	0	1,000
Carbon monoxide	10,000	190	110,000	0	120,000
Carbon dioxide	930,000	110,000	41,000,000	0	42,000,000
PM_{10}	1,100	9.6	11,000	0	12,000
$PM_{2.5}$	1,000	8.9	9,600	0	11,000
Hydrocarbons	4,100	89	27,000	150	31,000

Note: Numbers are rounded to two significant figures; therefore, totals might differ from sums.
a. To convert kilograms to pounds, multiply by 2.2046.

B.9 Greenhouse Gases

This section describes the methods DOE used to estimate emissions of greenhouse gases, primarily carbon dioxide during construction, operations, and all combined analytical periods at the proposed repository.

Carbon dioxide, which is a greenhouse gas, is emitted by the burning of fossil fuels such as diesel and gasoline. Construction equipment, surface vehicles, boilers, and generators would use the greatest amount of fossil fuel during the construction and operations analytical periods. Carbon dioxide is also emitted by concrete batch plants during the manufacture of concrete. Although human activities can

Table B-38. Air quality impacts from rail facilities and activities during the operations analytical period (micrograms per cubic meter).

Pollutant	Period	Maximum concentration[a]	Regulatory limit	Percent of regulatory limit[a]
Nitrogen dioxide	Annual	0.33	100	0.33
Sulfur dioxide	Annual	0.00086	80	0.0011
	24-hour	0.12	365	0.034
	3-hour	0.98	1,300	0.075
Carbon monoxide	8-hour	42	10,000	0.42
	1-hour	340	40,000	0.84
PM_{10}	24-hour	1.4	150	0.94
$PM_{2.5}$	Annual	0.0089	15	0.060
	24-hour	1.3	35	3.6

Note: Receptors at boundary of land withdrawal area.
a. Numbers are rounded to two significant figures.

produce other greenhouse gases such as methane and nitrous oxide, construction and operations activities would release only carbon dioxide in meaningful quantities. Therefore, DOE considered only carbon dioxide in this Repository SEIS.

Repository activities would not release methane in meaningful quantities because its primary emission sources are the production (not the combustion) of fossil fuels, agricultural activities, and the decay of organic waste in municipal solid waste landfills. None of these sources are part of the repository Proposed Action. Similarly, repository activities would not release nitrous oxide in meaningful quantities because its primary emission sources are agricultural activities. Although burning fossil fuel can emit small quantities of nitrous oxide, fossil-fuel combustion is a minor portion (less than 16 percent) of total nitrous oxide emissions in the United States (DIRS 185422-EPA 2006, all). As a consequence, the amount of nitrous oxide emitted by the burning of fossil fuels at the repository would not be meaningful.

The EPA emission factors for criteria pollutants do not include emission factors for carbon dioxide. Therefore, rather than having a different carbon dioxide emission factor for each different fuel-burning source (as for criteria pollutants), DOE used one emission factor for all diesel-fuel consumption and one emission factor for all gasoline consumption. The emission factor for the burning of diesel fuel is 22.23 pounds of carbon dioxide per gallon of diesel fuel (2.7 kilograms per liter), and the emission factor for the burning of gasoline is 19.37 pounds of carbon dioxide per gallon of gasoline (2.3 kilograms per liter) (DIRS 185297-EPA 2004, p. 2). Table B-39 lists the annual carbon dioxide emissions during the construction and operations analytical periods of the repository, based on the amount of diesel and gasoline consumed, and the total amount of carbon dioxide emitted from fossil-fuel burning during all analytical periods.

For carbon dioxide emissions from concrete manufacturing, DOE used an emission factor of 320 kilograms of carbon dioxide per cubic meter of concrete produced (DIRS 185469-Flowers and Sanjayan 2007, all). This is equivalent to 539 pounds of carbon dioxide per cubic yard of concrete. Table B-40 lists the annual carbon dioxide emissions during the construction and operations analytical periods of the repository, based on the amount of concrete produced per year and the total amount of carbon dioxide emitted from concrete batch plants during all analytical periods. Concrete manufacturing was estimated to occur during the first 4 years of the operations period while construction continued (DIRS 182713-Morton 2007, all).

Table B-39. Carbon dioxide emissions due to repository fossil-fuel burning during construction, operations, and all analytical periods.

Fuel	Fuel use (gallons)[a]	Fuel use (liters)	Carbon dioxide emissions (million pounds)[b]	Carbon dioxide emissions (million metric tons)[c]
Construction analytical period (annual)				
Maximum annual diesel	1,500,000	5,500,000	32	0.015
Maximum annual gasoline	47,000	180,000	0.90	0.00041
Maximum annual fossil fuel	1,500,000	5,700,000	33	0.015
Operations analytical period (annual)				
Maximum annual diesel	5,300,000	20,000,000	120	0.054
Maximum annual gasoline	220,000	850,000	4.3	0.0020
Maximum annual fossil fuel	5,600,000	21,000,000	120	0.056
All analytical periods (total)				
Total diesel	190,000,000	740,000,000	4,300	2.0
Total gasoline	8,200,000	31,000,000	160	0.072
Total fossil fuel	200,000,000	770,000,000	4,500	2.0

Notes: Numbers are rounded to two significant figures; therefore, totals might differ from sums.

a. Sources: DIRS 182211-Morton 2007, p. 2; DIRS 182210-Morton 2007, all; DIRS 155970-DOE 2002, p. 4-73. DOE has presented this measure in English units because of common statutory and public use.

b. To convert pounds to kilograms, multiply by 0.45359. DOE presented this measure in English units because of common statutory and public use.

c. To convert metric tons to tons, multiply by 1.1023.

Table B-40. Carbon dioxide emissions due to repository concrete batch plants during construction, operations, and all analytical periods.

Period	Concrete use (cubic meters)[a]	Concrete use (cubic yards)	Carbon dioxide emissions (million metric tons)[b]
Construction analytical period (annual)	65,000	85,000	0.021
Operations analytical period (annual)	41,600	54,000	0.013
All analytical periods (total)	490,000	640,000	0.16

Note: Numbers are rounded to two significant figures; therefore, totals might differ from sums.

a. Source: DIRS 182713-Morton 2007, all.

b. To convert metric tons to tons, multiply by 1.1023.

Carbon dioxide emissions from repository analytical periods can be compared to the overall State of Nevada emissions of carbon dioxide produced by existing activities. An estimated 47.9 million metric tons of carbon dioxide emissions were created in Nevada in 2004 (DIRS 185316-EIA n.d., all). During the construction analytical period, the annual amount of carbon dioxide produced by the combination of fossil-fuel burning and concrete manufacturing would be about 0.036 million metric tons, or 0.075 percent of 2004 Nevada carbon dioxide emissions. During the operations analytical period, while concrete batch plants were operating, the annual amount of carbon dioxide produced by the combination of fossil-fuel burning and concrete manufacturing would be about 0.069 million metric tons, or 0.14 percent of the 2004 Nevada carbon dioxide emissions. The total carbon dioxide emissions during all analytical periods would be about 2.2 million metric tons.

Carbon dioxide emissions from repository analytical periods can also be compared to the overall U.S. emissions of carbon dioxide produced by existing activities. An estimated 6,089 million metric tons of carbon dioxide emissions were created in the United States in 2005 (DIRS 185248-EPA 2007, all). During the construction analytical period, the annual amount of carbon dioxide produced by the combination of fossil-fuel burning and concrete manufacturing would be about 0.00059 percent of 2005 U.S. carbon dioxide emissions. During the operations analytical period, the annual amount of carbon dioxide produced by the combination of fossil-fuel burning and concrete manufacturing would be about 0.0011 percent of the 2005 U.S. carbon dioxide emissions.

In addition to the carbon dioxide emissions associated with the repository in Table B-39, carbon dioxide emissions associated with the railroad would occur in the analyzed land withdrawal area. Tables B-33, B-35, and B-37 list these emissions. During the construction analytical period, the annual carbon dioxide emissions associated with the railroad in the analyzed land withdrawal area would be approximately 52,000 metric tons (57,000 tons). This would be about 0.11 percent of the State of Nevada 2004 carbon dioxide emissions and compares with 36,000 metric tons (39,000 tons) of carbon dioxide emissions for activities at the repository. During the operations analytical period, the annual carbon dioxide emissions associated with the railroad in the analyzed land withdrawal area would be approximately 42,000 metric tons (46,000 tons). This would be about 0.087 percent of the State of Nevada 2004 carbon dioxide emissions and compares with 69,000 metric tons (76,000 tons) for activities associated with the repository.

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Appendix C

Floodplain/Wetlands Assessment for the Proposed Yucca Mountain Geologic Repository

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C. FLOODPLAIN/WETLANDS ASSESSMENT FOR THE PROPOSED YUCCA MOUNTAIN GEOLOGIC REPOSITORY

This appendix presents the floodplain and wetlands assessment for the Proposed Action to construct, operate, monitor, and eventually close a geologic repository at Yucca Mountain in southern Nevada for the disposal of spent nuclear fuel and high-level radioactive waste. Section C.1 describes the regulatory basis and history for the assessment. Section C.2 describes the Proposed Action in terms of activities that could affect floodplains and wetlands in the vicinity of Yucca Mountain, and Section C.3 characterizes the relevant existing environment. Section C.4 describes potential effects on floodplains (see Section C.1.2 for a discussion of effects on wetlands). Sections C.5 and C.6 discuss mitigation measures DOE would use and alternatives to the Proposed Action, respectively. Section C.7 contains the findings of the floodplains and wetlands assessment.

C.1 Introduction

Pursuant to Executive Order 11988, *Floodplain Management*, each federal agency, when it conducts activities in a floodplain, is to take actions to reduce the risk of flood damage; minimize the impacts 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. The U.S. Department of Energy (DOE or the Department) issued regulations that implement these Executive Orders (10 CFR Part 1022, "Compliance with Floodplain/Wetlands Environmental Review Requirements"). In accordance with the terms of these regulations, specifically 10 CFR 1022.11(d), DOE must prepare a floodplain assessment for proposed actions that would take place in floodplains and a wetlands assessment for proposed actions that would take place in wetlands. This appendix addresses DOE's obligations to perform a floodplain and wetlands assessment under 10 CFR Part 1022. The remainder of this section addresses pertinent past actions and decisions that could affect this assessment.

Congress enacted the *Nuclear Waste Policy Act of 1982* (Public Law 97-425, 96 Stat. 2201, January 7, 1983) to address the accumulation of spent nuclear fuel and high-level radioactive waste at 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. In 1987, Congress amended the Act (NWPA; 42 U.S.C. 10101 et seq.) by redirecting DOE to determine the suitability of only Yucca Mountain in southern Nevada.

In 1989, DOE published "Notice of Floodplain/Wetlands Involvement" (54 FR 63187, February 9, 1989) for site characterization studies at Yucca Mountain. The purpose of these studies was to determine the suitability of Yucca Mountain to isolate nuclear waste. DOE prepared a floodplain assessment (DIRS 104559-YMP 1991, all) and issued a Statement of Findings (56 FR 49765, October 1, 1991). In 1992, DOE prepared a second floodplain assessment on the cumulative impacts of surface-based investigations and the location of part of the Exploratory Studies Facility in the 100-year floodplain of a wash at Yucca Mountain (DIRS 103197-YMP 1992, all) and published the associated Statement of Findings (57 FR 48363, October 23, 1992). Both Statements of Findings concluded that the benefits of locating activities and structures in floodplains outweighed potential adverse impacts to the floodplains and that alternatives to these actions were not reasonable.

The NWPA requires that a final environmental impact statement (EIS) accompany any recommendation by the Secretary of Energy to the President to construct a repository. As part of the EIS process, and following the requirements of 10 CFR Part 1022, DOE issued "Notice of Floodplain and Wetlands Involvement" (64 FR 31554, June 11, 1999). The Notice requested comments from the public on potential impacts on floodplains and wetlands from the construction of a rail line or an 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 DOE selected. DOE received no comments from the public.

In February 2002, DOE completed the *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F; DIRS 155970-DOE 2002, all) (Yucca Mountain FEIS). Appendix L of the Yucca Mountain FEIS contained a floodplain and wetlands assessment prepared in accordance with 10 CFR Part 1022. The assessment examined the potential effects of repository construction and operation and construction of either a rail line or an intermodal transfer station and its associated heavy-haul truck route on (1) floodplains near the Yucca Mountain site and (2) floodplains and areas that might have wetlands along the five rail corridors and the five heavy-haul truck routes. In the assessment Statement of Findings, DOE concluded that the proposed actions at Yucca Mountain would be (1) unlikely to increase the risk of future flood damage, (2) unlikely to increase the impact of floods on human health and safety, or (3) unlikely to harm the natural, beneficial values of the floodplains because there are no human activities or facilities upstream or downstream that such activities could affect. In addition, DOE committed to a more detailed floodplains evaluation and wetlands delineation along the selected route for transport of spent nuclear fuel and high-level radioactive waste to the Yucca Mountain site. The Yucca Mountain FEIS identified rail as DOE's preferred mode of transportation, but did not identify a preference among the five rail corridors in Nevada.

By July 9, 2002, the recommendation to make Yucca Mountain the site for development of a geologic repository for spent nuclear fuel and high-level radioactive waste had passed from the Secretary of Energy to the President, then to Congress, and both the House of Representatives and the Senate had passed a joint resolution to approve the site. On July 23, 2002, the President signed Public Law 107-200, *Yucca Mountain Development Act of 2002*, which paved the way for DOE to seek licenses from the U.S. Nuclear Regulatory Commission (NRC) to build and operate a repository at Yucca Mountain.

In "Notice of Preferred Nevada Rail Corridor" (68 FR 74951, December 29, 2003), DOE named the Caliente rail corridor as its preferred route for construction of a rail line in Nevada. DOE published the corresponding Record of Decision (69 FR 18557) on April 8, 2004, and on the same date published "Notice of Intent to Prepare an Environmental Impact Statement for the Alignment, Construction, and Operation of a Rail Line to a Geologic Repository at Yucca Mountain, Nye County, NV" (69 FR 18565). On October 13, 2006, the Department amended the scope of the Rail Alignment EIS to include the Mina rail corridor in addition to the Caliente rail corridor (71 FR 60484). On the same day, the Department published a "Notice of Intent to Prepare a Supplement to the Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, NV" (71 FR 60490).

The purpose of this *Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County,* *Nevada* (DOE/EIS-0250F-S1) (Repository SEIS) is to evaluate the potential environmental impacts of the current repository design and operational plans.

Likewise, this floodplain/wetlands assessment updates the floodplain and wetlands assessment that DOE included with the Yucca Mountain FEIS to address current repository design and operational plans. Specifically, this assessment addresses potential effects of two elements: (1) the repository facility layout and design, and (2) a group of infrastructure improvements that DOE recently proposed to do in the near-term, before starting repository construction actions. This latter element consists of several different actions at and near Yucca Mountain that DOE considers necessary to continue ongoing activities and tests in a manner that ensures the health and safety of workers and visitors. DOE documented the proposed infrastructure improvements in the *Draft Environmental Assessment for the Proposed Infrastructure Improvements for the Yucca Mountain Project, Nevada* (DIRS 178817-DOE 2006, all), which it made available for public review on July 6, 2006 (Notice of Availability, 71 FR 38391). DOE has incorporated Appendix A of the draft environmental assessment, "Floodplain and Wetlands Assessment for the Proposed Infrastructure Improvements for the Yucca Mountain Project, Nevada," into this assessment.

The Nevada Rail Corridor SEIS and Rail Alignment EIS include an appendix containing a separate floodplain and wetlands assessment that provides a detailed floodplains evaluation and wetlands delineation along the Caliente and Mina rail corridors. As a result, this Repository SEIS (in contrast to the corresponding assessment in the Yucca Mountain FEIS) does not address potential impacts to floodplains and wetlands along the transportation corridors. There is, however, some overlap in the floodplains addressed in this document and those assessed in the Rail Alignment EIS because the rail line would cross some of the same drainage features at and near Yucca Mountain that repository construction would affect.

C.1.1 FLOODPLAIN DATA REVIEW

This assessment examines the potential effects of repository construction and operations on floodplains at and near the Yucca Mountain site. The floodplains of concern are those associated with Fortymile Wash, Busted Butte Wash (also known as Dune Wash), Drill Hole Wash, and Midway Valley Wash (also known as Sever Wash) (Figure C-1). These usually dry washes can fill with flowing water after very heavy, sustained rain or rapid snow melt.

Title 10 CFR 1022.4 defines a flood or flooding as ". . . a temporary condition of partial or complete inundation of normally dry land areas from the overflow of inland or tidal waters, or the unusual and rapid accumulation of runoff of surface waters from any source." It identifies floodplains that must be considered in the floodplain assessment as the base floodplain and the critical-action floodplain. The base floodplain is the area inundated by a flood having a 1-percent chance of occurrence in any given year (a 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 (a 500-year floodplain). Critical action is 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. DOE considered the critical-action floodplain because it could use petroleum-based fuel, oil, lubricants, and other hazardous materials during the construction of repository facilities, including upgrades of roads, and because it could transport spent nuclear fuel and high-level radioactive waste across washes and manage them at facilities adjacent to washes.

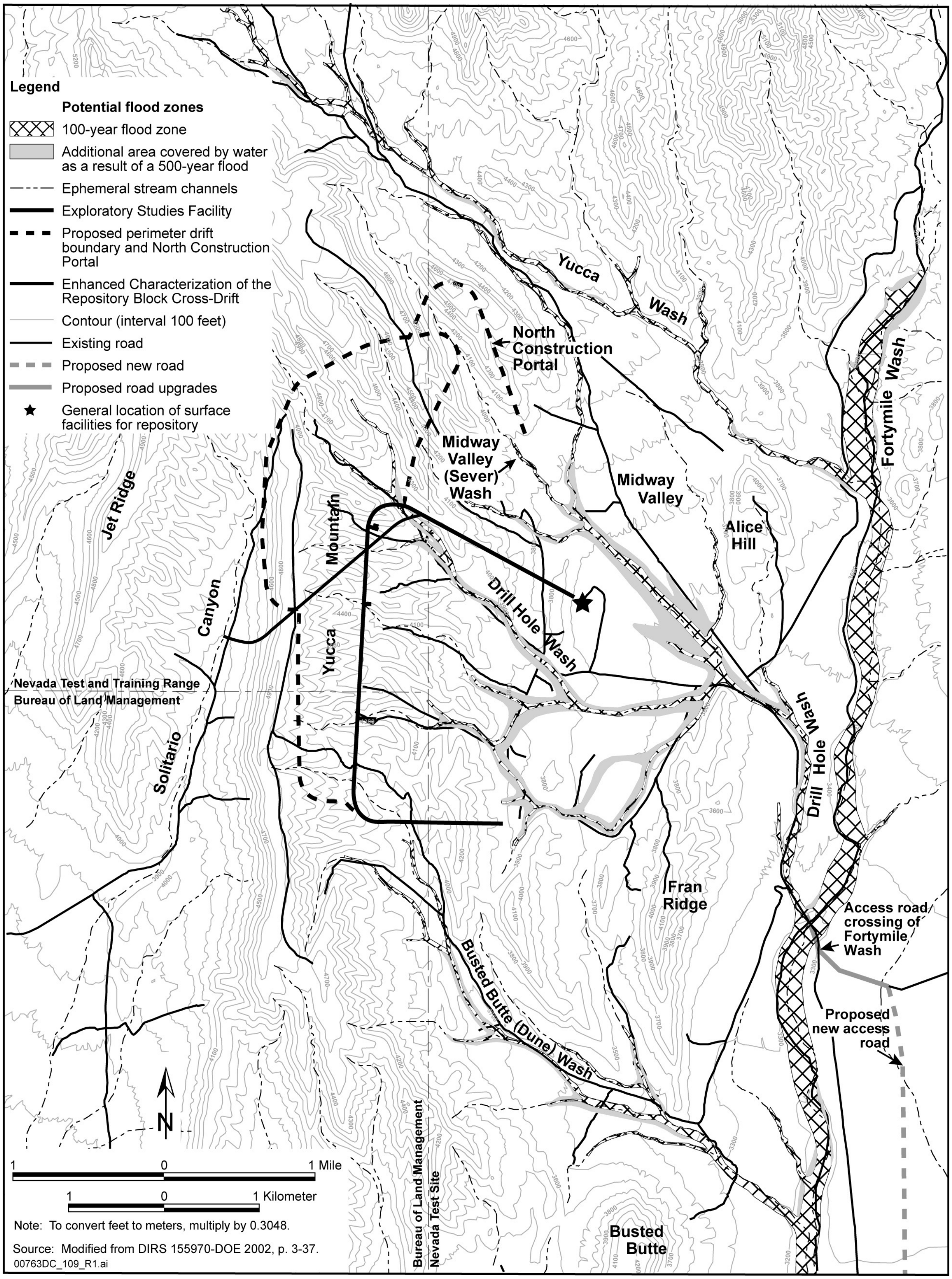

Figure C-1. Yucca Mountain site topography, drainage channels, and floodplains.

FLOOD TERMINOLOGY FROM 10 CFR PART 1022

Flood or flooding:
A temporary condition of partial or complete inundation of normally dry land area from the overflow of inland or tidal waters, or the unusual and rapid accumulation or runoff of surface waters from any source.

Floodplain:
The lowlands adjoining inland and coastal waters and relatively flat areas and flood-prone areas of offshore islands.

Title 10 CFR 1022.11 requires DOE to use Flood Insurance Rate Maps or Flood Hazard Boundary Maps to determine if a proposed action would be in the base or critical-action floodplain. On federal or state lands for which Flood Insurance Rate Maps or Flood Hazard Boundary Maps are not available, the Department must seek flood information from the appropriate land management agency or from agencies with expertise in floodplain analysis. Therefore, DOE asked the U.S. Geological Survey to complete a flood study of Fortymile Wash and its principal tributaries (which include Busted Butte, Drill Hole, and Midway Valley washes) and outline areas of inundation from 100- and 500-year floods (DIRS 180001-Squires and Young 1984, Plate 1). Figure C-1 shows the lateral extents of 100- and 500-year floods within these drainages.

In a related evaluation, DOE determined if the Caliente and Mina rail alignments would cross jurisdictional waters of the United States under Section 404 of the *Clean Water Act* (DIRS 183595-PBS&J 2006, all). Findings from this evaluation that were related to drainage channels on the east side of Yucca Mountain that an alignment would cross were of interest to this assessment. If drainage channels that repository actions affected qualified as waters of the United States, the qualification would not affect the requirements or applicability of including the drainage channels in this assessment. However, if the repository action involved construction or other work in waters of the United States, DOE would seek authorization pursuant to Section 404 of the *Clean Water Act* for the discharge of fill material in connection with construction of the repository.

According to the waters of the United States evaluation, the Amargosa River is an interstate water and, because Fortymile Wash is a tributary, it is a potential water of the United States under the jurisdiction of the U.S. Army Corps of Engineers (DIRS 183595-PBS&J 2006, p. 4). The washes that drain the east side of Yucca Mountain flow into Fortymile Wash and meet the same criteria for possibly qualifying as waters of the United States. For the last segment of the rail alignment, which would terminate at the Yucca Mountain site, the evaluation identified three ephemeral washes on the east side of Yucca Mountain as potential waters of the United States that the rail alignment would cross. From Figure 3E in the rail evaluation (DIRS 183595-PBS&J 2006, Appendix A, Figure 3E), the identified crossings appear to include two associated with Busted Butte Wash and one associated with Drill Hole Wash. (The evaluated rail alignment would not go as far north as Midway Valley Wash.) Although these evaluations were specific to the points along the washes where the rail alignment would cross, they imply that, under Corps of Engineers guidelines of the time, washes along the east side of Yucca Mountain as well as Fortymile Wash could qualify as waters of the United States.

In June 2007, the U.S. Environmental Protection Agency (EPA) and U.S. Army Corps of Engineers released interim guidance that addresses the jurisdiction over waters of the United States under the *Clean Water Act* (72 FR 31824, June 8, 2007). This guidance was a result of Supreme Court decisions that occurred after the DOE evaluation. Based on this guidance, it is likely that the drainages on the east side of Yucca Mountain that DOE currently considers potential waters of the United States might not be considered as such. Before undertaking construction in these washes, DOE would request that the Corps of Engineers determine the limits of jurisdiction under Section 404 of the *Clean Water Act*.

C.1.2 WETLANDS DATA REVIEW

Title 10 CFR Part 1022 requires DOE to determine if the Proposed Action would affect wetlands and, if necessary, to conduct a wetlands assessment. As required by 10 CFR 1022.11(c), DOE examined the following information in relation 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 (DIRS 147930-FWS 1995, all).

- U.S. Department of Agriculture, Soil Conservation Service Local Identification Maps. The Soil Conservation Service (now the 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. The surveys indicate there are no naturally occurring hydric soils at Yucca Mountain (DIRS 104592-CRWMS M&O 1999, pp. 2 to 6).

- U.S. Geological Survey Topographic Maps. Topographic maps of the vicinity (for example, DIRS 147932-USGS 1983, all) do not show springs, permanent streams, or other indications of wetlands.

- Regional or Local Government-Sponsored Wetlands or Land-Use Inventories. DOE has conducted a wetlands inventory of the Nevada Test Site (DIRS 101833-Hansen et al. 1997, p. 1-161). The closest naturally occurring wetlands to Yucca Mountain are on the upper west slope of Fortymile Canyon, 6 kilometers (3.7 miles) north of the North Portal and outside the area of any construction or other land disturbance associated with the repository.

Based on this information, DOE concluded that a wetlands assessment is not necessary to comply with 10 CFR Part 1022 because there are no wetlands that the Proposed Action could affect.

C.2 Project Description

Under the Proposed Action, the Yucca Mountain site would be the nation's geologic repository and DOE would ship spent nuclear fuel and high-level radioactive waste to the site for a period of up to 50 years. For this analysis, DOE assumed that emplacement of spent nuclear fuel and high-level radioactive waste would begin in 2017, after a 5-year construction analytical period. The discussion that follows has two parts. Section C.2.1 discusses the Proposed Action in the vicinity of the Yucca Mountain site. Section C.2.2 discusses proposed infrastructure improvements that would affect floodplains.

C.2.1 PROPOSED ACTIONS AT YUCCA MOUNTAIN

The preliminary layout of surface facilities in the geologic repository operations area shows these facilities would be in the primary natural drainage channel and associated floodplains of Midway Valley Wash and a short portion of the northern branch of Drill Hole Wash (Figure C-1). Construction of new roads or upgrades to existing roads and possibly placement of the large volumes of excavated rock, or muck, from the subsurface as DOE developed the repository emplacement area would probably affect other washes that drain the east side of Yucca Mountain (Busted Butte Wash and other portions of Drill Hole Wash).

A combination of drainage-control features would protect facilities in the geologic repository operations area from flash floods. DOE would build dikes and drainage ditches to surround much of the geologic repository operations area and other associated surface facilities to redirect runoff from outside the area. Exile Hill, although not shown on Figure C-1, is basically a raised rock on the side slope of Yucca Mountain where the North Portal starts. An existing diversion channel on the hill protects the west side of the operations area from runoff from that direction. DOE would integrate the Exile Hill diversion channel into the overall drainage-control features. In the operations area, new ditches, improved drainage channels, and stormwater detention ponds in the low eastern and southern sides of the diked area would control runoff. Culverts in the dikes would allow stormwater in the detention ponds to leave the area in a controlled (throttled) manner to join the natural drainage channel that runs through the gap between Fran Ridge to the south and Alice Hill to the north. From the gap between the two hills, where Midway Valley Wash joins Drill Hole Wash (Figure C-1), drainage would flow to the southeast and south in its current natural course to Fortymile Wash.

Construction in the geologic repository operations area would involve significant earthwork (excavation and filling) to establish the necessary foundations for buildings and the installation of utilities. As noted above, surface-water control measures (such as ditches, improved channels, and stormwater ponds) would be an element of the construction activities. Much of this work would be in, or over, areas shown in Figure C-1 as land where water would otherwise spread during times of flash flooding (that is, in floodplain areas). However, with the planned drainage-control features, this would no longer be the case. Because the affected natural drainage channels in this case originate at Yucca Mountain, changes would occur fairly high in the drainage system. The ditches and dikes DOE would construct to keep overland flow out of the operations area would intercept or block relatively minor channels, which are dry most of the time.

The U.S. Geological Survey mapped the 100- and 500-year floodplains of Fortymile Wash and its principal tributaries, as described in Section C.1.1 and shown in Figure C-1. DOE used another technique, referred to as the probable maximum flood method [based on American National Standards Institute and American Nuclear Society Standards for Nuclear Facilities (DIRS 103071-ANS 1992, all)] to estimate maximum flood volumes for specific segments of washes adjacent to planned Yucca Mountain facilities (DIRS 100530-Blanton 1992, all; DIRS 108883-Bullard 1992, all). In more recent studies, DOE has calculated probable maximum flood volumes and associated inundation areas that would result with consideration of tentative locations for surface facilities (DIRS 157928-BSC 2002, all; DIRS 169464-BSC 2004, all). These studies were a means to generate flooding criteria for the more detailed design of these facilities. The probable maximum flood method is widely used in hydrologic designs for structures critical to public safety, and federal regulations require the use of this method for the design of dam spillways, large detention basins, major bridges, and nuclear facilities. The method is a very conservative approach to generate the most severe flood volume reasonably possible for the location under evaluation, which is larger than even the 500-year flood. The 100-year, 500-year, or probable maximum flood would not be high enough to reach the entrances to the subsurface facilities at either the North or South portal. Studies are currently underway to generate probable maximum flood values for drainage channels near the planned location of the North Construction Portal to ensure that it too would be outside all possible flood levels. Some support facilities outside the North Portal would be in the natural flood zones for the 100-year, 500-year, and the more extensive probable maximum flood. DOE would design drainage-control measures to ensure the protection of those surface facilities that are important to safety against all reasonably possible floods. DOE would protect other central operations area facilities (those not important to safety) to withstand 100-year floods.

C.2.2 PROPOSED INFRASTRUCTURE ACTIONS

The existing access road to the Yucca Mountain surface facilities crosses about 460 meters (1,500 feet) of Fortymile Wash (Figure C-1) at grade; that is, it is directly on the surface of the wash and does not contain culverts. At this location, the wash contains several braided channels, and the occasional floods in Fortymile Wash flow across the road unimpeded. As the water subsides, rock debris in the road can make it impassable until heavy equipment removes the debris.

DOE proposes to replace the existing road where it crosses Fortymile Wash. The new road would be higher and drainage structures would channel floodwaters under the road (DOE would determine roadway and drainage improvements through further design). DOE would design this type of road upgrade to accommodate a 100-year flow, but the final design could consider a range of flood frequencies and a cost-benefit analysis. The culverts and associated dikes and other features that would modify the stream flow would also be designed to minimize erosion upstream and downstream of the crossing. DOE would use heavy earthmoving equipment to construct the road in accordance with standard road construction practices. This equipment would use petroleum-based fuels, oils, lubricants, and other hazardous materials, which DOE would store outside the 500-year floodplain (Figure C-1). The Department would obtain construction aggregate from existing borrow pits and concrete from local vendors.

On the west side of Fortymile Wash, the existing access road continues northward about 3.5 kilometers (2.2 miles) to a point where it is next to a 1.5-meter (4.9-foot)-wide ditch that is in the area where Drill Hole Wash and Midway Valley Wash merge and then drain toward Fortymile Wash (Figure C-1). Improvement of the access road could affect the drainage channel in the area, but the effects would be beneficial because DOE would size the drainage area to accommodate flow in the wash more appropriately. The access road from U.S. Highway 95 north to near the Fortymile Wash crossing would also involve segments of new road construction. The new road segments would cross many small washes. Because these washes are small, this assessment does not consider the effects of road construction to their associated floodplains further. It is noted, however, that design analyses, including hydrologic studies, would be performed as necessary to support design of drainage features for all segments of new road construction and would be required for road work within the Nevada Department of Transportation right-of-way in order to obtain the necessary approvals.

C.3 Existing Environment

Fortymile Wash is about 150 kilometers (93 miles) long and drains an area of about 810 square kilometers (200,000 acres) to the east and north of Yucca Mountain (Figure C-1). The wash continues south 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 riverbed extends another 100 kilometers (60 miles) before it ends in Death Valley.

Busted Butte Wash and Drill Hole Wash drain the east side of Yucca Mountain and flow into Fortymile Wash (Figure C-1); Midway Valley Wash is a tributary to Drill Hole Wash. Busted Butte Wash drains an area of 17 square kilometers (4,200 acres) and Drill Hole Wash drains an area of 40 square kilometers (9,900 acres).

Chapter 3 of this Repository SEIS describes the existing environment at and near Yucca Mountain, which includes Fortymile, Busted Butte, Drill Hole, and Midway Valley washes. The following sections summarize important aspects of the environment that pertain to this floodplain assessment.

C.3.1 FLOODING

Water flow in the four washes is infrequent. The dry, semiarid climate and meager precipitation [which averages 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. Table C-1 lists estimated peak discharges for the base (100-year) and critical-action (500-year) floodplains in Fortymile, Busted Butte, and Drill Hole washes.

Table C-1. Estimated peak discharges along washes at Yucca Mountain.

Name	Drainage area [square kilometers (acres)]	100-year flood peak discharge [cubic meters per second (cubic feet per second)]	500-year flood peak discharge [cubic meters per second (cubic feet per second)]
Fortymile Wash	810 (200,000)	340 (12,000)	1,640 (58,000)
Busted Butte Wash	17 (4,200)	40 (1,400)	184 (6,500)
Drill Hole Wash[a]	40 (9,900)	65 (2,300)	283 (10,000)

Source: DIRS 180001-Squires and Young 1984, p. 2.
a. Includes, as tributaries, Midway Valley Wash in the area of the North Portal and the wash in the area of the South Portal.

The Nevada Test Site access road to Yucca Mountain crosses Fortymile Wash in the area where it is joined by Drill Hole Wash. The next nearest manmade structure in Fortymile Wash is U.S. Highway 95, about 21 kilometers (13 miles) south of the confluence of Drill Hole and Fortymile washes. The portion of the community of Amargosa Valley that was once known as Lathrop Wells is the nearest population center to Yucca Mountain, about 22 kilometers (14 miles) to the south along U.S. Highway 95 and 4.8 kilometers (3 miles) east of Fortymile Wash.

Flooding in the region is often localized. A flash flood in one or more of the washes that drains to Fortymile Wash, for example, might not result in any notable flow in Fortymile Wash. Although infrequent, storm and runoff conditions can be extensive enough to result in flow throughout the drainage system. “Modern Flooding and Runoff of the Amargosa River, Nevada-California, Emphasizing Contributions of Fortymile Wash” (DIRS 155679-Glancy and Beck 1998, all) documented conditions during March 1995 and February 1998 when Fortymile Wash and the Amargosa River flowed simultaneously through their primary channels to Death Valley. The 1995 incident was the first documented case of this flow condition, though undocumented incidents probably occurred during the preceding 30 years when there were several instances for which records show sections of the primary channels flowing with floodwater.

C.3.2 WETLANDS

There are no springs, perennial streams, hydric soils, or naturally occurring wetlands in the affected areas at Yucca Mountain.

C.3.3 BIOLOGY

Vegetation at and near Fortymile Wash is typical of the Mojave Desert. The mix or association of vegetation in the wash, which is dominated by the shrubs white bursage (*Ambrosia dumosa*), creosote bush (*Larrea tridentate*), white burrobush (*Hymenoclea salsola*), and heathgoldenrod (*Ericameria paniculata*) differs somewhat from other vegetation associations at Yucca Mountain (DIRS 104589-CRWMS M&O 1998, pp. 5 to 7). No plant species grow exclusively in the floodplains. In addition, none of the more than 180 known plant species at Yucca Mountain is endemic to the area.

No documented mammals, reptiles, or bird species at Yucca Mountain are restricted to or dependent on the floodplains, and these species are widespread throughout the region. Studies have found no amphibians at Yucca Mountain.

The only plant or animal species at Yucca Mountain that the EPA has classified under the *Endangered Species Act* is the desert tortoise (*Gopherus agassizii*), which is threatened. Yucca Mountain is at the northern edge of the range of the desert tortoise (DIRS 101915-Rautenstrauch et al. 1994, p. 11). Desert tortoises occur in the floodplain of Fortymile Wash, but their abundance there and elsewhere at Yucca Mountain is low in comparison with other parts of their range farther south and east (DIRS 102869-CRWMS M&O 1997, pp. 6 to 11). DOE generated *Environmental Baseline File for Biological Resources* (DIRS 104593-CRWMS M&O 1999, all), which included summary information on the ecology of the desert tortoise population at Yucca Mountain.

Several animal and plant species that the Bureau of Land Management or the State of Nevada have classified as sensitive occur at Yucca Mountain (Section 3.1.5.1.3 of this Repository SEIS). These species can occur in the floodplains at and near Yucca Mountain but are not dependent on habitat there (DIRS 104590-CRWMS M&O 1998, p. 8; DIRS 103159-CRWMS M&O 1998, pp. 22 and 23; DIRS 103654-Steen et al. 1997, pp. 19 to 29).

C.3.4 ARCHAEOLOGY

Years of research at and near Yucca Mountain have discovered 830 archaeological sites, and that number increases to well over 1,000 when including isolated artifacts, some of which are in Fortymile Wash. These sites range from small scatters of lithic (stone) artifacts to campsites and quarries. They indicate that American Indian populations have occupied the Yucca Mountain region for at least 12,000 years. Fortymile Wash was an important crossroad where several trails converged from such distant places as Owens Valley, Death Valley, and the Avawatz Mountains. A draft programmatic agreement among DOE, the Advisory Council on Historic Preservation, and the Nevada State Historic Preservation Office has been prepared for cultural resources management related to activities that would be associated with development of a repository at Yucca Mountain. While this agreement is in negotiation among the concurring parties, DOE is abiding by Section 106 of the *National Historic Preservation Act of 1966* (16 U.S.C. 470) process.

C.4 Floodplain Effects

Title 10 CFR 1022.13(a)(2) requires a floodplain assessment to discuss the positive and negative, direct and indirect, and long- and short-term effects of a proposed action on an affected floodplain. In addition, the assessment must evaluate the effects on lives and property, and on natural and beneficial values of floodplains. If DOE finds no practicable alternative to the location of activities in floodplains, it would design or modify its actions to minimize potential harm to or in the floodplains. The floodplains DOE assessed are areas of normally dry washes that are temporarily and infrequently inundated from runoff, including during 100-year or more intense (and less frequent) floods. The following sections address effects specific to repository development actions at Yucca Mountain, effects from infrastructure actions, and effects common to both sets of actions.

C.4.1 EFFECTS AT YUCCA MOUNTAIN

Construction of the proposed repository and the associated surface support facilities could affect each of the three primary washes that drain the east side of Yucca Mountain. The most affected would be Midway Valley Wash, which DOE would reroute so it could construct facilities adjacent to the North Portal entrance of the repository and protect them from potential flash flooding. A short portion of the northern branch of Drill Hole Wash (Figure C-1) would be similarly affected (that is, DOE would reroute the natural drainage in this portion of the wash). Road construction and road upgrades would probably affect the other primary washes that drain the east side of Yucca Mountain in this area (Busted Butte Wash and the other portions of Drill Hole Wash), but these effects would occur at crossings with drainage structures, as necessary, or at grade rather than drainage channel reroutes. DOE expansion of existing or new rock storage piles into existing drainage channels could require drainage rerouting for relatively short distances.

DOE would construct facilities for the receipt and management of spent nuclear fuel and high-level radioactive waste close to the North Portal of the repository, which would be the access point to the subsurface area for emplacement of the nuclear waste. The Department would build dikes around this area on the southwest, southeast, northeast, and around to the north sides. Exile Hill, the location of the North Portal, and an existing drainage channel on the hill would protect the west side from runoff. Outside the diked area, natural drainage channels would carry runoff except in areas where dikes intercepted channels and runoff. In those areas, runoff would flow along the dike until the flow reached another natural drainage point. Runoff would concentrate in the gap between Fran Ridge to the south and Alice Hill to the north, in the same place it now exits the area and drains (via the lower section of Drill Hole Wash) into Fortymile Wash. The main access road into the geologic repository operations area would come through this same gap; DOE would build drainage structures under the road as necessary for runoff to reach the natural drainage channels. Inside the diked portion of the geologic repository operations area, a combination of new ditches and improved channels would manage runoff. They would direct runoff to the low eastern and southeastern portions of the diked area, where stormwater detention ponds and culverts would drain accumulated water through the dikes. Water that went though the dikes would join the natural drainage channels to the natural gap and on to Fortymile Wash.

Construction across washes that involved the placement of drainage structures would reduce the area through which floodwaters naturally flow. During large floods, bodies of water could develop on the upstream side of each crossing and slowly drain through drainage structures. This would be an intended result of the design of the dikes and stormwater detention ponds in the geologic repository operations area. In the case of road crossings, if the flood occurred quickly and was sufficiently large, water could flow over the road and continue downstream, which could damage the road. 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 floodplains because there are no human activities or facilities upstream or downstream that floods could affect. If runoff or floodwater was held on the upstream side of a drainage feature, there would be a potential for sediment to fall out of the flow and accumulate in the channel. These areas would be subject to periodic maintenance, as necessary, to remove and dispose of accumulated sediment.

C.4.2 EFFECTS FROM INFRASTRUCTURE ACTIONS

The floodplain of Fortymile Wash is normally dry, but runoff, such as would occur during 100- or 500-year floods, can temporarily and infrequently inundate it. Improvement of the existing access road where it crosses Fortymile Wash would reduce the area through which floodwaters naturally flow. During large floods, bodies of water could develop on the upstream side of the crossing 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 nearby human activities or facilities upstream or downstream that they would affect. A sufficiently large flood in Fortymile Wash could create a temporary large lake upstream of the improved road that would slowly drain through the drainage structures. If the flood occurred quickly and was sufficiently large, the dammed water could flow over the road and continue downstream. Some road damage could occur, but the damage would be unlikely to 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 nearby human activities or facilities downstream that floods would affect.

During flood events, sediment would probably accumulate on the upstream side of the Fortymile Wash crossing. DOE would have to remove this material periodically so future floodwaters would have sufficient space to accumulate, rather than overflow the structures during later smaller floods. When necessary, DOE would remove this material by truck and dispose of it appropriately. Under natural conditions, this sediment would have continued downstream and been deposited as the floodwater receded. In comparison with the total amount of sediment that floodwater moves along the entire length of the washes, the amount that accumulated behind the crossing would be small.

During a 100- or 500-year flood, there would be no preferred channels; most channels across the entire width of Fortymile Wash would fill with water (Figure C-1). Therefore, the road would not cause preferential flow in a particular channel or alter the velocity or direction of flow on the floodplain.

C.4.3 EFFECTS COMMON TO BOTH SETS OF ACTIONS

Potential construction across washes and over large areas of a wash, as in the case of Midway Valley Wash, would require the removal of desert vegetation and the disturbance of soil and alluvium. These actions could affect wildlife habitat and individual animals, including the threatened desert tortoise. In 2000, DOE consulted with the U.S. Fish and Wildlife Service about the effects on the desert tortoise from construction, operations, monitoring, and eventual closure of a repository at Yucca Mountain. The Fish and Wildlife Service concluded in a Biological Opinion in 2001 that it was unlikely that these activities would jeopardize the desert tortoise (DIRS 155970-DOE 2002, Appendix O, pp. 21 to 22). This opinion and its associated incidental-take provisions are applicable to the construction, operations, monitoring, and closure analytical periods of the Proposed Action. As directed in the Biological Opinion, DOE would conduct surveys for tortoises or their nests and eggs for avoidance or relocation before surface-disturbing activities, and would perform other mitigation measures delineated in the opinion.

Construction in the floodplains could affect unidentified cultural resources. Before construction, archaeologists would survey the area in accordance with the Programmatic Agreement currently being finalized among DOE, the Advisory Council on Historic Preservation, and the Nevada State Historic Preservation Office. This agreement will address the performance of cultural resources management during the licensing and repository development phases. Cultural resources surveys during previous phases were in accordance with an earlier Programmatic Agreement with the Advisory Council on Historic Preservation (DIRS 104558-DOE 1988, p. 5). DOE would avoid cultural sites if possible; if not possible, DOE would conduct a data recovery program for the sites in accordance with the Programmatic Agreement being negotiated (Section C.3.4). The Department would preserve artifacts from and knowledge about the site. Improved access to the area could lead to indirect impacts, which could include unauthorized excavation or collection of artifacts. Workers would receive required training on the protection of these resources from excavation or collection.

Potential indirect impacts on flora and fauna would include increased emissions of fugitive dust, elevated noise levels, and increased human activities. Emissions of fugitive dust would be short-term and unlikely to have a significant effect on vegetation or wildlife. Significant long-term impacts to wildlife from the temporary increase in noise during construction would be unlikely. Wildlife displaced during construction would probably return after the completion of construction.

Periodic maintenance activities, such as sediment removal and drainage structure repair or replacement, would probably have effects similar to those of construction, but generally of smaller magnitude and shorter duration. Before performing maintenance actions, DOE would take measures similar to those described for construction to identify any flora, fauna, or cultural resources of concern and, as appropriate, identify mitigation measures.

There are no perennial sources of surface water at or downstream from the Yucca Mountain site that the proposed construction activities or periodic maintenance actions would affect.

Construction would not substantially affect the quality or the quantity of groundwater that normally recharges through Fortymile Wash. Water infiltration could increase somewhat after large floods as standing water slowly entered the ground behind crossing or diked areas. The total volume of these water bodies would be a few thousand cubic meters (a few acre-feet) at most, and much of the water would gradually drain through culverts or evaporate before it infiltrated deep into the ground where it might eventually reach the water table, about 300 meters (980 feet) below the surface at Fortymile Wash.

DOE would control the use of petroleum fuels, oils, lubricants, and other hazardous materials during construction, would clean up spills promptly and, if necessary, remediate the soil and alluvium. Cleanup and remediation would also occur if there was a hazardous material release during transport to the site on the access road. The small amount of such materials that reached the ground would have little, if any, potential to affect groundwater.

The nearest residents are about 22 kilometers (14 miles) to the south, along U.S. Highway 95 in the community of Amargosa Valley, a few kilometers east of Fortymile Wash. If floodwaters from a 100- or 500-year flood reached this far downstream, there would be no measurable increase in the flood velocity or sediment load attributable to construction activities for the Yucca Mountain project in comparison with natural conditions. Therefore, disturbances to the floodplains of Fortymile, Busted Butte, Drill Hole, and Midway Valley washes would have no adverse impacts on lives and property downstream. Moreover, impacts to these floodplains would be insignificant in both the short and long terms in comparison to the erosion and deposition that occur naturally and erratically in these washes and floodplains.

During operation of the repository, the fall of a truck or railcar that carried spent nuclear fuel or high-level radioactive waste into Busted Butte, Drill Hole, Midway Valley, or Fortymile washes would be extremely unlikely. However, if this occurred, the shipping casks, which are designed to prevent the release of radioactive materials during an accident, would remain intact. DOE would recover the casks and transport them to the repository. No adverse impacts to surface-water or groundwater quality from such accidents would occur.

DOE has identified no positive or beneficial impacts to the floodplains of Busted Butte, Drill Hole, Midway Valley, or Fortymile washes from the proposed repository and infrastructure actions.

C.5 Mitigation Measures

According to 10 CFR 1022.13(a)(3), DOE must address measures to mitigate the adverse impacts of actions in floodplains, which include but are not limited to minimum grading requirements, runoff controls, design and construction constraints, and protection of ecologically sensitive areas. This section discusses floodplain mitigation measures that DOE would consider in the vicinity of Yucca Mountain and, where necessary and feasible, implement in the washes.

Adverse impacts to the affected floodplains would be small. Even during 100- and 500-year floods, differences in the rate and distribution of erosion and sedimentation caused by the proposed construction would probably not be measurably different from existing conditions. Upgrades to access roads and placement of excavated rock storage piles in the site area would have little effect on erosion and sedimentation from flooding events. DOE would perform hydrologic studies as necessary and design the drainage structures, dikes, improved channels, and other features it would install to modify stream flow to minimize erosion upstream and downstream. In addition, DOE would follow its reclamation guidelines for site clearance, topsoil salvage, erosion and runoff control, recontouring, revegetation, construction practices, and site maintenance (DIRS 154386-YMP 2001, all). The Department would minimize disturbance of surface areas and vegetation, maintain natural contours to the maximum extent feasible, stabilize slopes to minimize erosion, and avoid unnecessary off-road vehicle travel. Storage of hazardous materials during construction would be outside the floodplains.

Before construction began, DOE would require preconstruction surveys to ensure the work would not affect sensitive biological or archaeological resources. In addition, these surveys would determine the site's reclamation potential. If construction could threaten important biological or archaeological resources, and modification or relocation of the item under construction or improvement was not reasonable, DOE would incorporate mitigation measures into the design of the work. These measures would include relocation of sensitive species, avoidance of archaeological sites, or data recovery if avoidance was not feasible. In that case, DOE would evaluate the cultural resources for their importance and eligibility for inclusion in the *National Register of Historic Places*, and would collect and document artifacts at eligible sites in accordance with Section 106 of the *National Historic Preservation Act* and the Programmatic Agreement negotiated between DOE, the Advisory Council on Historic Preservation, and the Nevada State Historic Preservation Office (Section C.3.4). In the years after construction, DOE would take similar actions before performing any maintenance to determine if work could affect sensitive biological resources that might have moved back into the area or newly identified archeological resources.

If there were spills of hazardous materials during construction of the facilities and roads or during transport to the repository, DOE would quickly clean the spill and remediate the soil and alluvium. Storage of hazardous materials would be away from floodplains to decrease the probability of an inadvertent spill in these areas.

C.6 Alternatives

According to 10 CFR 1022.13(a)(3), DOE must consider alternatives to a proposed action. DOE has addressed alternatives in relation to sites for surface construction for both the repository and infrastructure upgrades.

C.6.1 ALTERNATIVES TO ACTIONS AT YUCCA MOUNTAIN

The long history of alternatives that DOE has considered has led to the Proposed Action at Yucca Mountain. The geologic disposal of radioactive waste has been the focus of more than 40 years of scientific research. After an extensive consideration of options, Congress enacted the *Nuclear Waste Policy Act of 1982*, which specified that DOE will dispose of spent nuclear fuel and high-level radioactive waste underground in deep geologic repositories. In the 1987 amendment, Congress directed DOE to study only Yucca Mountain to determine its suitability as a repository. On July 9, 2002, Congress passed a joint resolution that approved Yucca Mountain as the site for development of a geologic repository. As a result, the only alternative to the Proposed Action that DOE considered in the 2002 Yucca Mountain FEIS and this Repository SEIS is the No-Action Alternative. Under the No-Action Alternative, DOE would avoid additional impacts or effects on floodplains at and near Yucca Mountain, but would not meet its legal obligation to develop a repository.

In the framework of repository development, DOE could have designed a surface facility layout with less disturbance to existing drainage channels and floodplains than that described in this assessment. However, avoidance of all effects to floodplains is unreasonable. DOE would base its ultimate design of surface facilities and their exact layouts on optimization of the efficiency of those facilities in the performance of their functions and, more importantly, in the protection of the health and safety of the people who would work in those facilities and adjacent areas. Given the relatively minor effects on floodplains from the Proposed Action, protection of the health and safety of the workers and a facility layout that optimizes their efficiency are more significant criteria. There is no practicable alternative that would affect floodplains less.

C.6.2 ALTERNATIVES TO INFRASTRUCTURE ACTIONS

To operate a repository at Yucca Mountain, DOE would require a road that crossed Fortymile Wash to access facilities west of the Wash. Consideration of a new access road across the Wash is unreasonable if the existing road, if improved, would adequately meet DOE operational needs. Moreover, a new access road across the Wash at a different location would increase environmental damage and costs. Because of these concerns, DOE eliminated a new access road across the Wash from detailed consideration.

Selection of the No-Action Alternative would avoid additional impacts to Fortymile Wash. DOE could use the existing road, but this alternative would not meet the Department's operational needs.

C.7 Floodplain Statement of Findings

Consistent with the presentations in this assessment, this section contains a Floodplain Statement of Findings for those actions at the Yucca Mountain site and for the infrastructure actions that would affect only Fortymile Wash.

C.7.1 STATEMENT OF FINDINGS FOR ACTIONS AT YUCCA MOUNTAIN

Facilities that DOE would build at the Yucca Mountain site would encroach on the primary natural drainage channel and associated floodplains of Midway Valley Wash and a short portion of the northern branch of Drill Hole Wash. Construction of new roads or upgrades to existing roads and possible placement of the large volumes of excavated rock from the subsurface would probably affect other washes that drain the east side of Yucca Mountain (Busted Butte Wash and portions of Drill Hole Wash). Because Yucca Mountain has been designated as the site for development of a geologic repository, DOE maintains that there are no practicable alternatives to the locations of facilities, roads, and materials in floodplains at the Yucca Mountain site. The ultimate design and layout of surface facilities would optimize the efficiency of their functions and protect the health and safety of workers. DOE would avoid floodplains associated with the normally dry drainage channels at Yucca Mountain to the extent these other criteria would not be jeopardized.

Construction of new facilities and roads and upgrades to existing facilities and roads would affect floodplains in the vicinity of the Yucca Mountain site. To provide adequate protection for these facilities from flash flooding, DOE would dike areas and reroute natural drainage channels. In areas where roads crossed existing washes, the Department would generally install drainage structures (unless the crossing was at grade); construction activities could reduce the area through which floodwaters naturally flow. However, none of these impacts would be likely to 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 floods could affect.

The No-Action Alternative would avoid additional impacts or effects on floodplains at and near Yucca Mountain, but would not achieve DOE's legal obligation under the NWPA to develop a repository for the nation's spent nuclear fuel and high-level radioactive waste.

During construction and operations at the Yucca Mountain site, DOE would avoid disturbance of sensitive species, cultural resources, and floodplains whenever possible. If avoidance was not practicable, the Department would use standard mitigation practices to minimize the potential impacts to floodplains. Procedures would include preconstruction and biological surveys to identify and relocate sensitive species; avoidance of archaeological sites (or data recovery if avoidance was not feasible); modification of designs and implementation of good engineering practices such as minimizing the size of disturbance areas, salvaging topsoil, preserving natural contours, and controlling surface erosion or runoff;

reclamation and revegetation of disturbed areas; and use of established guidelines for hazardous materials storage and spill response.

DOE would construct some surface facilities in floodplains in accordance with all applicable requirements, which include state or local floodplain protection standards. If Busted Butte Wash, Drill Hole Wash, or Midway Valley Wash qualified as a jurisdictional water of the United States, the Department would obtain the appropriate permit, or permits, from the U.S. Army Corps of Engineers for actions in those washes. DOE would base its planning and actions on consultations with the Corps of Engineers.

C.7.2 STATEMENT OF FINDINGS FOR INFRASTRUCTURE ACTIONS

Effects to the floodplain of Fortymile Wash would occur from improvements to the existing access road where it crosses Fortymile Wash. Construction activities could reduce the area through which floodwaters naturally flow. However, none of these actions would be likely to increase the risk of future flood damage, increase the impact of floods on human health and safety, harm the natural and beneficial values of the floodplains because there are no nearby human activities or facilities upstream or downstream that floods could affect. There are no delineated wetlands at or near Yucca Mountain.

Under the No-Action Alternative, no new impacts to the floodplain of Fortymile Wash would occur, but DOE would not meet its operational needs.

During construction and upgrade activities, DOE would use standard mitigation practices to minimize potential impacts to the floodplain of Fortymile Wash. Procedures would include preconstruction surveys to identify and, if necessary, relocate sensitive species and avoid cultural sites; modification of designs and implementation of good engineering practices such as minimizing the size of disturbances, salvaging topsoil, preserving natural contours, and controlling surface erosion and runoff; reclamation and revegetation of disturbed areas; and use of established guidelines for hazardous materials storage and spill response.

DOE would perform its proposed infrastructure actions in the floodplain of Fortymile Wash in accordance with all applicable requirements, which include state or local floodplain protection standards. If Fortymile Wash qualified as a jurisdictional water of the United States, DOE would obtain the appropriate permit from the U.S. Army Corps of Engineers for the action. DOE would base its planning and actions on consultations with the Corps of Engineers.

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Appendix D

Radiological Health Impacts Primer and Estimation of Preclosure Radiological Health Impacts

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D. RADIOLOGICAL HEALTH IMPACTS PRIMER AND ESTIMATION OF PRECLOSURE RADIOLOGICAL HEALTH IMPACTS

This appendix contains information that supports the estimates of preclosure human health and safety impacts in this *Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F-S1) (Repository SEIS). Preclosure impacts would occur during construction, operations, monitoring, and closure of the proposed repository. (Chapter 5 and Appendix F discuss postclosure repository performance; Appendix E discusses potential radiological impacts of accidents.)

Section D.1 is a primer that explains the nature of radiation, the origin of radiation in the context of radiological impacts, and how radiation interacts with the human body to produce health impacts. Section D.2 describes releases of radiological materials to the atmosphere that would affect involved and noninvolved workers and the public. Section D.3 describes the affected populations of these groups and the hypothetical maximally exposed workers and members of the public among those populations. Section D.4 discusses the methodology and data the analysis used to estimate occupational and public health impacts and presents the detailed results.

D.1 Radiological Health Impacts Primer

This section discusses the concepts of human health impacts as a result of exposure to radiation.

D.1.1 RADIATION

Radiation is the emission and propagation of energy through space or through a material in the form of waves or bundles of energy called photons or in the form of high-energy subatomic particles. Radiation generally results from atomic or subatomic processes that occur naturally.

The most common kind of radiation is electromagnetic radiation, which consists of photons. Electromagnetic radiation occurs over a range of wavelengths and energies. People are most commonly aware of visible light, which is part of the spectrum of electromagnetic radiation. Types of radiation of longer wavelengths and lower energy include infrared, which heats an exposed material, and radio waves. Types of electromagnetic radiation of shorter wavelengths and higher energy (which are more penetrating) include ultraviolet, which causes sunburn, and x-rays and gamma radiation.

Ionizing radiation is radiation that has sufficient energy to displace electrons from atoms or molecules to create ions. It can be electromagnetic (for example, x-rays or gamma radiation) or subatomic particles (for example, alpha, beta, or neutron radiation). The ions have the ability to interact with other atoms or molecules; in biological systems, this interaction can cause damage in the tissue or organism.

D.1.2 RADIOACTIVITY

Radioactivity is the property or characteristic of an unstable atom to undergo spontaneous transformation (to disintegrate or decay) with the emission of energy as radiation. The emitted radiation is usually ionizing. The result of radioactive decay is the transformation of an unstable atom (a radionuclide) into a different atom, which releases energy (as radiation) as it reaches a more stable, lower-energy configuration.

Radioactive decay produces three main types of ionizing radiation—alpha particles, beta particles, and gamma or x-rays. Each of these types can have different characteristics and levels of energy and, therefore, different abilities to penetrate and interact with atoms in the human body. Because each type has different characteristics, each requires different amounts of material to stop (or shield) the radiation. Alpha particles are the least penetrating; a thin layer of material such as a single sheet of paper stops them. However, if radioactive atoms (called radionuclides) emit alpha particles inside the body when they decay, there is a concentrated deposition of energy near the point where the decay occurs. Shielding beta particles requires thicker layers of material such as several reams of paper or several centimeters of wood or water. Shielding from gamma rays, which are highly penetrating, requires several centimeters to several meters of heavy material (for example, concrete or lead). A gamma ray disperses energy along the line of passage through the body in contrast to the local energy deposition by an alpha particle. Some gamma radiation can pass through the body without interaction.

In a nuclear reactor, heavy atoms such as uranium and plutonium can undergo another process, called fission, after the absorption of a subatomic particle (usually a neutron). In fission, a heavy atom splits into two lighter atoms and releases energy in the form of radiation and the kinetic energy of the two new lighter atoms. These lighter atoms are called fission products. The fission products are usually unstable and undergo radioactive decay toward a more stable state. Some of the heavy atoms might not fission after they absorb a subatomic particle. A new nucleus forms instead that tends to be unstable (like fission products) and undergo decay. The decay of fission products and unstable heavy atoms, some of which can generate neutrons by spontaneous fission or by alpha interaction, is the source of the radiation from spent nuclear fuel and high-level radioactive waste that makes these materials hazardous in terms of potential human health impacts.

D.1.3 EXPOSURE TO RADIATION AND RADIATION DOSE

Radiation that originates outside the body is external or direct radiation. Such radiation can come from an x-ray machine or from radioactive materials that directly emit radiation, such as radioactive waste or radionuclides in soil. Shielding, such as lead, between the source of the radiation and the exposed individual can reduce or eliminate the exposure. Internal radiation originates inside a person's body after an intake of radioactive material through ingestion or inhalation. Once the material is in the body, its chemical behavior and how the body metabolizes it affect the potential for damage to the body. If the material is soluble, bodily fluids might dissolve it, transport it to various body organs, and deposit it there. If the material is insoluble, it might move rapidly through the gastrointestinal tract if it was ingested or deposit in the lungs if it was inhaled.

Exposure to ionizing radiation is expressed in terms of absorbed dose, which is the amount of energy that is imparted to matter per unit mass. Often simply called dose, it is a fundamental concept in the measurement and quantification of the effects of exposure to radiation. The unit of absorbed dose is the rad. The different types of radiation have different effects in damage to cells of biological systems. With the use of a radiation-specific quality factor, the dose equivalent concept accounts for the absorbed dose and the relative effectiveness of the type of ionizing radiation damage to biological systems. The unit of dose equivalent is the rem.

There are several additional concepts in quantifying the effects of radiation on humans. The effective dose equivalent method quantifies effects of radionuclides in the body through estimation of the susceptibility of the different tissues in the body to radiation to produce a tissue-specific weighting factor, which is based on the susceptibility of that tissue to cancer. The unit of effective dose equivalent is the rem. The sum of the products of each affected tissue's estimated dose equivalent multiplied by its specific weighting factor is the effective dose equivalent for a particular type of exposure. The potential effects from a one-time ingestion or inhalation of radioactive material are calculated over a period of 50 years to account for radionuclides that have long half-lives and long residence times in the body. The result is the committed effective dose equivalent. Total effective dose equivalent is the sum of the committed effective dose equivalents from radionuclides in the body and the dose equivalent from radiation sources external to the body. All estimates of radiation dose in this Repository SEIS, unless specifically noted otherwise, are total effective dose equivalents in rem or millirem.

More detailed information on the concepts of radiation dose and dose equivalent is available in Report 115 from the National Council on Radiation Protection and Measurements (DIRS 101857-NCRP 1993, all) and Publication 60 from the International Commission on Radiological Protection (DIRS 101836-ICRP 1991, all).

The factors for conversion of estimates of radionuclide intake (by inhalation or ingestion) or external exposure to radionuclides [by groundshine or cloudshine (immersion)] to radiation dose are dose conversion factors or dose coefficients. The International Commission on Radiological Protection and federal agencies such as the U.S. Environmental Protection Agency (EPA) publish these factors (DIRS 172935-ICRP 2001, all; DIRS 175544-EPA 2002, all), which are based on original recommendations of the International Commission on Radiological Protection (DIRS 101836-ICRP 1991, all) and incorporate the dose coefficients from International Commission on Radiological Protection Publication 72 (DIRS 152446-ICRP 1996, all).

The radiation dose to an individual or to a group of people can be expressed as the total received dose or as a dose rate, which is dose per unit time (usually an hour or a year). Population dose is the total dose to an exposed population; person-rem is the unit. Population dose (or collective dose) is the sum of the individual dose to each member of a population. For example, if 100 workers each received 0.1 rem, the population dose would be 10 person-rem.

D.1.4 BACKGROUND RADIATION

Nationwide, on average, members of the public receive approximately 360 millirem of radiation per year from natural and manmade sources (DIRS 101855-NCRP 1987, p. 53). About 60 millirem per year are from medical radiation and consumer products. About 300 millirem are from natural sources (DIRS 100472-NCRP 1987, p. 149). The largest natural sources are radon-222 and its radioactive decay products in homes and buildings, which contribute about 200 millirem per year. Additional natural sources include radioactive material in the Earth (primarily the uranium and thorium decay series and potassium-40) and cosmic rays from space that make it through the atmosphere. In relation to exposures from human activities, the combined doses from weapons testing fallout, consumer and industrial products, and air travel (cosmic radiation) account for the remaining approximately 3 percent of the total annual dose. Nuclear fuel-cycle facilities contribute 0.05 millirem per year, less than 0.1 percent of the total dose.

D.1.5 IMPACTS TO HUMAN HEALTH FROM EXPOSURE TO RADIATION

Exposures to radiation or radionuclides are often characterized as being acute or chronic. Acute exposures occur over a short period, typically 24 hours or less. Chronic exposures occur over longer periods (months to years) and are usually continuous over the period, even though the dose rate might vary. For a given dose of radiation, chronic exposure is usually less harmful than acute exposure because the dose rate (dose per unit time, such as rem per hour) is lower, which provides more opportunity for the body to repair damaged cells.

D.1.5.1 Acute Exposures at High Dose Rates

Exposures to high levels of radiation at high dose rates over a short period (less than 24 hours) can result in acute radiation effects. Minor changes in blood characteristics might occur at exposures in the range of 25 to 50 rad. The external symptoms of radiation sickness begin to appear following acute exposures of about 50 to 100 rad and can include anorexia, nausea, and vomiting. More severe symptoms occur at higher doses and can include death at doses higher than 200 to 300 rad of total body irradiation, depending on the level of medical treatment. Information on the effects of acute exposures on humans is the result of studies of the survivors of the Hiroshima and Nagasaki bombings and from studies after a number of accidental acute exposures.

Acute exposures have occurred after detonations of nuclear weapons in wartime and during weapons testing, and in other events that involved testing of nuclear materials. Exposures could also occur during other activities, such as medical procedures involving radiation, at processing plants that use radiation to irradiate food, and during weld radiography.

D.1.5.2 Chronic Exposures at Low Dose Rates

The analysis for this Repository SEIS assumed all doses would be at low dose rates. Such exposures can be chronic (continuous or nearly continuous), such as those cask handlers and health physics technicians would receive. In some instances, exposures to low levels of radiation would be intermittent (for example, infrequent exposures to persons along the transportation routes DOE would use to ship spent nuclear fuel and high-level radioactive waste to the proposed repository). Cancer induction is the principal potential risk to human health from exposure to low levels of radiation. The estimation of cancer induction is a statistical process in that exposure to radiation conveys only a chance of incurring cancer, not a certainty. Further, cancer induction in individuals can occur from other causes, such as exposure to chemical agents.

D.1.6 DOSE-TO-HEALTH-EFFECT CONVERSION FACTORS

Cancer is the principal potential risk to human health from exposure to low or chronic levels of radiation. Radiological health impacts are expressed as the incremental changes in the number of expected fatal cancers (latent cancer fatalities) for populations and as the incremental increases in the lifetime probability of an individual contracting a fatal cancer. The estimates are based on the received dose and on dose-to-health-effect conversion factors that were recommended by the Interagency Steering Committee on Radiation Standards (DIRS 174559-Lawrence 2002, all) and by updated DOE guidance (DIRS 178579-DOE 2004, pp. 22 to 24). The Steering Committee consists of eight federal agencies (EPA, NRC, DOE, the U.S. Department of Defense, the U.S. Department of Homeland Security, the U.S.

Department of Transportation, the Occupational Safety and Health Administration, and the U.S. Department of Health and Human Services), three federal observer agencies (the Office of Science and Technology Policy, the Office of Management and Budget, and the Defense Nuclear Facilities Safety Board), and observer agencies from two states (Illinois and Pennsylvania). The Committee estimated that, for the general population and workers, a population dose of 1 person-rem would yield 0.0006 excess latent cancer fatality.

Sometimes, calculations of the number of latent cancer fatalities in relation to dose do not yield whole numbers and, especially in environmental applications, can yield values less than 1. For example, if each individual in a population of 100,000 received a total radiation dose of 0.001 rem, the population dose would be 100 person-rem and the corresponding estimated number of latent cancer fatalities would be 0.06 (100,000 persons × 0.001 rem × 0.0006 latent cancer fatalities per person-rem). How should one interpret a nonintegral number of latent cancer fatalities, such as 0.06? The answer is to interpret the result as a statistical estimate; that is, 0.06 is the average number of latent cancer fatalities that would result if the same exposure situation occurred to many different groups of 100,000 people. For most groups, no one would incur a latent cancer fatality from the 0.001-rem radiation dose each member had received. In a small fraction of the groups (about 6 percent), 1 latent cancer fatality would result, and in exceptionally few groups, 2 or more latent cancer fatalities would occur. The average number of latent cancer fatalities for all the groups would be 0.06. The most likely outcome for any single group is no latent cancer fatalities.

D.1.7 COMPARISON WITH OTHER DOSE-TO-HEALTH-EFFECT CONVERSION FACTORS

The updated dose-to-health-effect conversion factor of 0.0006, which this Repository SEIS uses, is similar to the lethality-adjusted cancer risk coefficients from the International Commission on Radiological Protection of 0.00041 per person-rem for workers and 0.00055 per person-rem for individuals among the general population (DIRS 185466-ICRP 2007, p. 53). It is also similar to the conversion factors from the National Research Council in *Health Risks from Exposure to Low Levels of Ionizing Radiation, BEIR VII Phase 2* (DIRS 181250-National Research Council 2006, p. 15), which range from 0.00041 to 0.00061 latent cancer fatality per person-rem for solid cancers and 0.00005 to 0.00007 latent cancer fatality per person-rem for leukemia, and to the age-specific dose-to-health-effect conversion factor of 0.000575 latent cancer fatality per person-rem from the EPA (DIRS 153733-EPA 2000, Table 7.3, p. 179).

D.1.8 LINEAR NO-THRESHOLD MODEL

The premise of the linear no-threshold model is that there is some risk, even at the lowest radiation doses. The Committee on the Biological Effects of Ionizing Radiation reviewed the linear no-threshold model (DIRS 181250-National Research Council 2006, p. 9). The Committee examined arguments that low doses of radiation are more harmful than the linear no-threshold model suggests, and it concluded that radiation health effects research, as a whole, does not support this view.

D.1.9 RADIATION HORMESIS

The premise of radiation hormesis is that a threshold or decrease in effect exists at low radiation doses, and that use of the linear no-threshold model exaggerates the health effects of low levels of ionizing radiation. The Committee on the Biological Effects of Ionizing Radiation reviewed the issue of radiation hormesis (DIRS 181250-National Research Council 2006, pp. 9 and 10). The Committee did not accept the hypothesis that the risks are lower than the linear no-threshold model predicts, that they are nonexistent, or that low doses of radiation could even be beneficial. The Committee concluded that there is always some risk, even at low doses.

D.1.10 OTHER RADIATION HEALTH EFFECTS

Table D-1 lists other health effects such as nonfatal cancers and genetic effects that can occur as a result of chronic exposure to radiation. The International Commission on Radiological Protection evaluated these other health effects (DIRS 185466-ICRP 2007, p. 53).

Table D-1. Detriment-adjusted nominal risk coefficients for cancer and heritable effects from exposure to radiation.

Population	Cancer (per rem)	Heritable effects (per rem)	Total (per rem)
Whole population	5.5×10^{-4}	2×10^{-5}	5.7×10^{-4}
Adults	4.1×10^{-4}	1×10^{-5}	4.2×10^{-4}

Source: DIRS 185466-ICRP 2007, p. 53.
Note: Numbers are rounded to two significant figures; therefore, totals might differ from sums.

The dose-to-health-effect conversion factors for cancer in Table D-1, 0.00041 per person-rem for workers and 0.00055 per person-rem for individuals among the general population, are based on cancer incidence data but include consideration of cancer lethality and life impairment. In addition, Table D-1 lists dose-to-health-effect conversion factors for heritable effects—0.00001 per person-rem for workers and 0.00002 per person-rem for individuals among the general population. The total detriment, 0.00042 per person-rem for workers and 0.00057 per person-rem for individuals among the general population, is consistent with the recommended factor of 0.0006. While DOE recognizes the existence of health effects other than fatal cancers, it has chosen to quantify the impacts in this Repository SEIS in terms of latent cancer fatalities, in part because the other health effects are a small portion of the total detriment from exposure to radiation.

Radiation exposure increases the risk of other diseases, particularly cardiovascular disease, in persons who receive high therapeutic doses and in atomic bomb survivors and others who receive more modest doses.

The Committee on the Biological Effects of Ionizing Radiation reviewed the issue of health effects other than cancer (DIRS 181250-National Research Council 2006, p. 8). The Committee concluded that there was no direct evidence of increased risk of noncancer diseases at low doses and that data were inadequate to quantify this risk if it exists. Radiation exposure increases the risk of some benign tumors, but the Committee concluded that data were inadequate to quantify this risk.

D.1.11 PRENATAL EXPOSURE

Studies of prenatal exposure or exposure in early life to diagnostic x-rays have shown that there is a significantly increased risk of leukemia and childhood cancer from a diagnostic dose of 1 to 2 rem to the embryo or fetus in utero (DIRS 181250-National Research Council 2006, pp. 172 and 173). In recognition of this, DOE and NRC regulations (10 CFR 835.206 and 10 CFR 20.1208, respectively) specifically address protection of declared pregnant workers from radiation, in which they limit the exposure of the embryo or fetus to 0.5 rem during the period from conception to birth.

D.2 Atmospheric Releases of Radioactive Materials

There would be two major types and sources of radionuclide releases to the air from project activities at the proposed repository. The ventilation exhaust air from the subsurface facility would contain naturally occurring radon-222 and its decay products during all project analytical periods (construction, operations, monitoring, and closure) (Section D.2.1). Handling and transfer of commercial spent nuclear fuel in the surface Wet Handling Facility and aging of transportation, aging, and disposal (TAD) and dual-purpose canisters inside aging overpacks in the Aging Facility during operations would release manmade radioactive materials (Section D.2.2). There would be other minor sources of release from the subsurface repository: neutron activation of ventilation air in the emplacement drifts, release of neutron-activated rock dust to the air from the emplacement drift walls, and resuspension of surface contamination on waste packages to the air in the emplacement drifts (Section D.2.3). As indicated in Section D.5.1, almost all (99.8 percent) of the potential health impacts to the public would be from exposure to naturally occurring radon-222 and its decay products released in subsurface exhaust ventilation air.

D.2.1 RELEASE OF RADON-222 AND RADON DECAY PRODUCTS FROM THE SUBSURFACE FACILITY

In the subsurface facility, radon-222 would diffuse continuously from the rock into the air. Radioactive decay of the radon would produce radon decay products during transport through the ventilation system. The primary radionuclide members of the radon-222 decay chain are polonium-218, lead-214, and bismuth-214. Exhaust ventilation air would carry the radon-222 and the radon decay products that originated from the host rock. For this analysis, DOE based the estimates of radon-222 releases and radon decay product concentrations in the subsurface facility on concentration data from the Exploratory Studies Facility and the concentration calculation results for a fully developed repository (DIRS 164380-BSC 2003, all; DIRS 167021-BSC 2003, all).

In calculating radon releases over time, the analysis assumed that the releases would increase linearly over the 5-year construction analytical period and the first 22 years of the beginning of the 50-year operations analytical period. The maximum annual radon release would begin after the completion of excavation, last the final 28 years of the operations period, and continue through the monitoring analytical period. During the monitoring period, forced ventilation would continue at the same rate, as would the radon release rate. Monitoring and maintenance activities would last for 50 years. Releases of radon and its decay products during the closure analytical period duration of 10 years would decrease linearly as crews gradually sealed openings. The initial release rate would be the same as that of the monitoring period and would decrease to none. Figure D-1 shows the estimated radon release rate as a function of time.

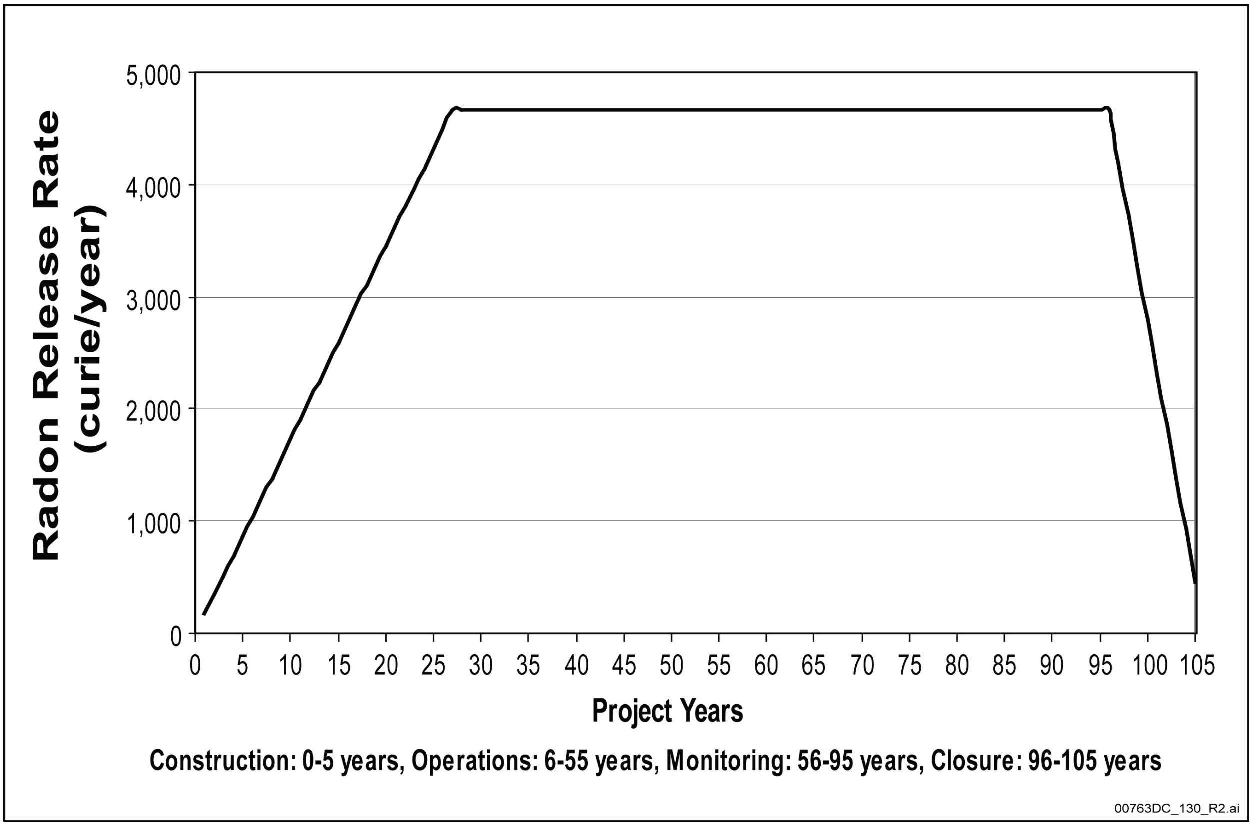

Figure D-1. Radon release rate as a function of time.

D.2.2 RELEASES OF RADIONUCLIDES FROM SURFACE FACILITIES

As explained in Chapter 2 of this Repository SEIS, DOE assumed that 90 percent of the commercial spent nuclear fuel would arrive at the proposed repository in TAD canisters. Although DOE has a small amount of spent nuclear fuel of commercial origin that it could ship to the repository uncanistered in a cask, consistent with the analysis in the Yucca Mountain FEIS, this Repository SEIS assumes that it would transport and receive all DOE spent nuclear fuel and high-level radioactive waste in disposable canisters. None of the canisters of DOE materials would require opening at the repository; workers would place them directly into waste packages. Therefore, releases from these canisters during normal operations would not occur. About 10 percent of the commercial spent nuclear fuel would arrive at the repository either as uncanistered fuel or in dual-purpose canisters. Nondisposable canisters would require opening in the Wet Handling Facility, where workers would handle uncanistered spent nuclear fuel and nondisposable canisters underwater using remote-control equipment underwater to load the fuel into TAD canisters for eventual placement in a waste package.

Commercial spent nuclear fuel contains encapsulated uranium, transuranic elements, fission products, and activation products in the structural materials of the fuel assemblies or as crud on the exterior of the fuel assemblies. Small amounts of radioactive materials would be released into the pool of the Wet Handling Facility and the exhaust ventilation air. The water would capture most of the materials, which would become part of the low-level radioactive waste stream that DOE would manage as described in Chapter 4, Section 4.1.12 of this Repository SEIS. The materials that entered the exhaust ventilation air would be filtered, but the radioactive gases and a small percentage of the particulates in the canisters or shipping containers would be released to the atmosphere under normal operating conditions.

The Aging Facility, which would stage and age spent nuclear fuel, would be the only surface facility other than the Wet Handling Facility with the potential to release radioactive materials to the environment during normal operations. Radionuclides released from the Aging Facility would be the resuspension of loose surface contamination on TAD and dual-purpose canisters inside aging overpacks. The following sections describe the assumptions and methods for estimation of these releases.

D.2.2.1 Airborne Release Radionuclide Composition

Airborne releases during normal operations would occur in the Wet Handling Facility during processing of uncanistered fuel and fuel from dual-purpose canisters. Because DOE would receive 90 percent of the commercial spent nuclear fuel in TAD canisters, potential airborne releases would be only from the remaining portion of the waste stream. To estimate conservatively the magnitude of radioactive releases from the Wet Handling Facility, the analysis assumed that all pressurized-water-reactor spent nuclear fuel would consist of the same composition of radionuclides as that estimated for a pressurized-water-reactor fuel assembly with 4.2-percent initial enrichment, 50,000 megawatt-days per metric ton of heavy metal (MTHM) burnup rate, and 10-year cooling time, and all boiling-water-reactor spent nuclear fuel would consist of the same composition of radionuclides as that estimated for a boiling-water-reactor fuel assembly with 4-percent initial enrichment, 50,000 megawatt-days per MTHM burnup rate, and 10-year cooling time (DIRS 180185-BSC 2007, Section 7). These fuel compositions bound the expected annual average characteristics of the fuel that has the potential to contribute to airborne releases during normal operations in the Wet Handling Facility (DIRS 180185-BSC 2007, Section 7). These bounding representative spent fuel assembly characteristics were determined (DIRS 180185-BSC 2007, all) by analyzing yearly average fuel characteristics using the waste stream scenario (DIRS 180258-BSC 2007, all) developed based on loading commercial spent nuclear fuel in TAD canisters beginning in 2011 and shipping the youngest fuel that is greater than or equal to 5 years old first beginning in 2017.

DOE based the radioactive surface contamination level it used to estimate radionuclide releases from the Aging Facility during normal operations on 0.0001 microcurie per square centimeter for beta-gamma emitters and low-toxicity alpha emitters and 0.00001 microcurie per square centimeter for all other alpha emitters. These surface contamination levels represent the maximum permissible surface contamination limits on the exterior of a shipping package (49 CFR 173.443, Table 9). The analysis used cobalt-60 to bound the dose contribution of beta-gamma emitters and low-toxicity alpha emitters, and americium-241 to bound the dose contribution of all other alpha emitters. The analysis determined that the release rate based on the staging capacity of the Aging Facility for cobalt-60 would be 0.029 curie per year and the release rate for americium-241 would be 0.0029 curie per year from the Aging Facility (DIRS 185287-BSC 2008, Section 6.2.2).

D.2.2.2 Release Parameters

DOE based the parameters for release estimates primarily on NRC guidance and the use of data and experience from operating nuclear power plants. Releases of gases and materials from a spent nuclear fuel rod would occur only in the event of fuel failures in which the cladding of the fuel cracked or leaked. NRC guidance indicates that less than 1 percent of commercial spent nuclear fuel would have failed fuel rods (DIRS 149756-NRC 2000, p. 9-12; DIRS 160582-NRC 2003, Attachment, Table 7.1). To estimate crud releases, the analysis assumed 15 percent of the crud surface activity would become loose from the fuel surfaces and 10 percent of the loose crud would become airborne during normal operations. The 15-percent loose fraction is from NRC guidance (DIRS 149756-NRC 2000, p. 9-12; DIRS 160582-NRC 2003, Attachment, Table 7.1). The 10-percent airborne release fraction is the bounding release fraction for the case in which venting gases pressurized the volume in which loose powdering surface contamination existed (DIRS 103756-DOE 1994, p. 5-22). Table D-2 lists the radionuclide release fractions. Each fraction, except that for crud, is the fraction of the total radionuclide inventory in a commercial spent nuclear fuel rod. The fraction for crud is applicable to all fuel rods, and the fractions for other groups are applicable only to the failed fuel rods in a fuel assembly.

Table D-2. Airborne release fractions by radionuclide group.

Radionuclide group	Spent nuclear fuel nuclide	Release fraction[a]
Gases	Hydrogen-3 Carbon-14 Chlorine-36 Krypton-85 Iodine-129	0.3
Volatiles	Cesium-134 Cesium-137	0.0002
Crud	Cobalt-60 Iron-55	0.015[b]
Fuel fines	Particulates	0.00003

a. Source: DIRS 149756-NRC 2000, p. 9-12; DIRS 160582-NRC 2003, Attachment, Table 7.1.
b. Source: DIRS 149756-NRC 2000, p. 9-12; DIRS 160582-NRC 2003, Attachment, Table 7.1; DIRS 103756-DOE 1994, p. 5-22.

The analysis used the release fractions, a decontamination factor of 10,000 for a two-stage high-efficiency particulate air filter system in the Wet Handling Facility, the analyzed schedule of receipts, and the design capacity of the Wet Handling Facility to estimate the amount of radionuclides handling activities would release to the environment as a result of normal operations. Table D-3 lists radionuclide releases for an annual throughput of 3,600 MTHM of commercial spent nuclear fuel; 10 percent of this amount (360 MTHM per year) would require handling in the Wet Handling Facility. The listed radionuclides are those the analysis determined to be important for dose calculation based on the selection criteria in NRC guidance (DIRS 149756-NRC 2000, p. 9-11; DIRS 160582-NRC 2003, Attachment, Section 3). These nuclides represent more than 99.8 percent of the total radionuclide source term activity and contribute more than 99.9 percent of the calculated offsite dose from the release of manmade radionuclides. The table includes all gaseous nuclides.

D.2.3 AIRBORNE RELEASES FROM SUBSURFACE FACILITY

During normal operations of the subsurface repository, in addition to the continuous release of radon-222 through the ventilation exhaust, three mechanisms could generate additional airborne releases of radioactive materials: neutron activation of ventilation air in the emplacement drifts, release of neutron-activated rock dust to the air from the emplacement drift walls, and release of radioactive surface contamination from waste packages in the emplacement drifts. The waste package surface contamination resuspension release was estimated based on the recommended surface contamination levels for waste packages prior to placement in the repository (DIRS 164177- Edwards and Yuan 2003, Section 6.1). During repository operation, an operational procedure for waste package contamination surveys would be required to demonstrate that removable surface contamination in excess of the contamination levels is not present on the waste packages. The derived contamination levels represent the average concentration of radioactivity on the external surfaces of a waste package that would not be exceeded before the waste package was transported to the subsurface repository. The derivation of the contamination level is based on the requirement that the annual average concentrations of radioactive material released at the repository shaft exhaust do not exceed the airborne effluent concentration limit specified in Table 2 of Appendix B to 10 CFR Part 20. Table D-3 lists the estimated annual releases of radionuclides from the subsurface facility under normal operating conditions (DIRS 172487-BSC 2005, Tables 13, III-1, III-4).

Table D-3. Maximum annual releases from normal operations.[a]

Subsurface facility releases		Surface facility releases		
Radionuclide	Curies per year	Radionuclide	Curies per year	
			PWR	BWR
Activated air[b]		Wet Handling Facility releases[c,d]		
Nitrogen-16	5.8	Hydrogen-3	5.5×10^{2}	5.7×10^{2}
Argon-41	1.5×10^{1}	Carbon-14	9.6×10^{-1}	1.1
Activated dust[e]		Chlorine-36	1.9×10^{-2}	1.9×10^{-2}
Nitrogen-16	2.1×10^{-5}	Krypton-85	7.1×10^{3}	6.3×10^{3}
Sodium-24	3.7×10^{-3}	Iodine-129	5.2×10^{-2}	5.0×10^{-2}
Aluminum-28	4.0×10^{-3}	Cesium-134	6.2×10^{-4}	4.7×10^{-4}
Silicon-31	5.2×10^{-4}	Cesium-137	9.2×10^{-3}	8.7×10^{-3}
Potassium-42	8.0×10^{-4}	Barium-137m	8.6×10^{-2}	8.2×10^{-3}
Iron-55	8.2×10^{-5}	Crud (cobalt-60)	1.9×10^{-2}	1.5×10^{-1}
Waste package surface contamination[f]		Crud (iron-55)	2.4×10^{-1}	2.7×10^{-1}
Cobalt-60	2.9×10^{-3}	Strontium-90	9.3×10^{-4}	9.0×10^{-4}
Nickel-63	6.3×10^{-6}	Yttrium-90	9.3×10^{-4}	9.0×10^{-4}
Strontium-90	6.8×10^{-4}	Ruthenium-106	5.2×10^{-5}	3.3×10^{-5}
Yttrium-90	6.8×10^{-4}	Antimony-125	8.9×10^{-6}	6.5×10^{-6}
Cesium-137	6.8×10^{-3}	Promethium-147	1.4×10^{-4}	1.1×10^{-4}
Promethium-147	3.0×10^{-6}	Europium-154	5.4×10^{-5}	4.2×10^{-5}
Samarium-151	5.3×10^{-6}	Europium-155	1.1×10^{-5}	1.0×10^{-5}
Europium-154	1.7×10^{-5}	Plutonium-238	6.3×10^{-5}	5.5×10^{-5}
Plutonium-238	5.7×10^{-5}	Plutonium-239	4.1×10^{-6}	2.9×10^{-6}
Plutonium-239	4.4×10^{-6}	Plutonium-240	7.3×10^{-6}	6.9×10^{-6}
Plutonium-240	7.9×10^{-6}	Americium-241	2.7×10^{-5}	2.0×10^{-5}
Americium-241	4.9×10^{-5}	Plutonium-241	1.2×10^{-3}	8.5×10^{-4}
Plutonium-241	6.2×10^{-4}	Americium-243	5.2×10^{-7}	4.7×10^{-7}
Americium-243	5.5×10^{-7}	Curium-243	3.6×10^{-7}	3.0×10^{-7}
Curium-243	2.6×10^{-7}	Curium-244	5.9×10^{-5}	5.0×10^{-5}
Curium-244	3.4×10^{-5}	Aging Facility releases[g]		
Naturally occurring radioactivity[h]		Cobalt-60	2.9×10^{-2}	
Radon-222	4.7×10^{3}	Americium-241	2.9×10^{-3}	

a. The listed source-term nuclides would contribute more than 99.9 percent of the total dose to the population offsite.
b. Source: DIRS 172487-BSC 2005, Table III-1.
c. Based on Wet Handling Facility throughput of 360 MTHM per year and a decontamination factor of 10,000 for a two-stage high-efficiency particulate air filter system in the Wet Handling Facility.
d. DOE chose the fuel type (PWR or BWR) that produces the highest dose for each receptor location.
e. Source: DIRS 172487-BSC 2005, Table III-4.
f. Source: DIRS 172487-BSC 2005, Table 13.
g. Source: DIRS 185287-BSC 2008, Section 6.2.2.
h. Assumes a fully excavated repository from DIRS 167021-BSC 2003, p. 37.

BWR = Boiling-water reactor.
PWR = Pressurized-water reactor.

The principal pathways by which airborne radioactivity from the repository could reach workers or the public would be (1) direct external exposure from radionuclides in the air and on the ground, (2) inhalation of radioactivity into the lungs followed by redistribution to other organs of the body, and (3) ingestion of radioactivity in foodstuffs for offsite members of the public.

D.3 Affected Populations and Individuals

Radiological impacts are measured in terms of doses to individuals and to populations. A dose is a measure of the amount of energy that radiation deposits in the body. A number of terms describe radiation doses. This analysis examined two dose categories: individual dose and population dose. Individual dose is a measure of the maximum dose to an individual. Population dose is a measure of the dose to the population outside the repository boundary or a group of workers inside the repository boundary; it is the sum of the doses to the individuals in the population or group of workers.

This section describes the four analyzed population groups and the locations of the maximally exposed individuals in each group: (1) the general population within 84 kilometers (52 miles) of the proposed repository, (2) the noninvolved worker population at the Nevada Test Site, (3) the noninvolved worker population at the repository, and (4) the involved worker population at the repository.

Members of the public, involved workers, and noninvolved workers could be exposed to atmospheric releases of radionuclides from repository activities. In this analysis, estimated noninvolved worker population doses from radon releases include population doses for both involved and noninvolved workers.

D.3.1 PUBLIC

The closest residents to the repository would be in the Armargosa Valley. The analysis assumed the maximally exposed member of the public would be a hypothetical individual who resided continuously for 70 years at a location in the unrestricted public access area that could receive the highest radiation exposure. The atmospheric dispersion calculations indicated this location would be 19 kilometers (12 miles) in the south-southeast direction for releases from the geologic repository operations area and 18 kilometers (11 miles) in the south-southeast direction for releases from subsurface facilities (DIRS 183160-BSC 2007, Tables 18 and 24). The release points for radon and other subsurface facility releases would include the South Portal and one to six exhaust ventilation shafts. Normal operations releases of manmade radionuclides from the surface geologic repository operations area would occur from the Wet Handling Facility and Aging Facility.

Table D-4 lists the estimated average population distribution for 2067 of about 117,000 within 84 kilometers (52 miles) of the proposed repository. The analysis based this number on projected changes in the region, which includes the towns of Amargosa Valley, Beatty, Pahrump, and Indian Springs and the surrounding rural areas. The analysis used information from state and local sources (Chapter 3, Section 3.1.8). The table lists the population in the vicinity of Pahrump even though part of the population would be beyond the 84-kilometer region. The analysis calculated both annual population dose and cumulative dose for the Proposed Action duration of 105 years, which would consist of analytical periods of 5 years of construction, 50 years of operations, 50 years of monitoring, and 10 years of closure, which would overlap the final 10 years of the monitoring analytical period.

Table D-4. Projected 2067 population distribution within 84 kilometers (52 miles) of repository site.

Direction	Distance (kilometers)[a]										
	8	16	24	32	40	48	56	64	72	80	Totals
South	0	0	39	1,000	1,685	402	0	2	0	0	3,128
South-southwest	0	0	0	1,107	245	0	0	2	0	0	1,354
Southwest	0	0	0	0	0	0	347	16	0	0	363
West-southwest	0	0	0	0	0	0	0	0	60	0	60
West	0	0	0	1,492	31	0	0	0	0	0	1,523
West-northwest	0	0	123	2,468	0	0	0	0	0	12	2,603
Northwest	0	0	0	69	0	0	0	0	85	0	154
North-northwest	0	0	0	0	0	0	0	0	0	0	0
North	0	0	0	0	0	0	0	0	0	0	0
North-northeast	0	0	0	0	0	0	0	0	0	0	0
Northeast	0	0	0	0	0	0	0	0	0	0	0
East-northeast	0	0	0	0	0	0	0	0	0	0	0
East	0	0	0	0	0	0	0	0	0	0	0
East-southeast	0	0	0	0	0	0	0	0	4,034	0	4,034
Southeast	0	0	0	0	0	0	90	8	16	516	630
South-southeast	0	0	0	0	74	427	69	172	21,281	81,612	103,635
Totals	0	0	162	6,136	2,035	829	506	200	25,476	82,140	117,484

a. To convert kilometers to miles, multiply by 0.62137.

D.3.2 NONINVOLVED WORKERS

The analysis assumed noninvolved workers on the surface would be at the site 2,000 hours a year (8 hours a day, 5 days a week, 50 weeks a year). Noninvolved workers would be construction, managerial, technical, supervisory, and administrative personnel who would not be directly involved in subsurface excavation and waste operations activities. In this analysis, noninvolved workers included onsite construction workers during the first several years of repository operations when construction activities would continue in parallel with ongoing operations. All workers, regardless of work responsibility, would receive exposure to releases of radon-222 and its decay products from the subsurface facilities. The maximally exposed noninvolved worker location for releases of radon and its decay products would be about 100 meters (330 feet) northeast of the South Portal development area for all analytical periods. DOE based the noninvolved worker population in the South Portal development area on the number of full-time equivalent worker years for subsurface workers. The number of noninvolved workers in the South Portal development area would be 15 percent of the subsurface workers. During the construction analytical period and the development of the first two emplacement panels during initial operations (Chapter 2, Section 2.1.2.2.1), ventilation air from repository excavation activities would exhaust from the South Portal and result in the highest potential exposure to radon and radon decay products. Once waste package emplacement began in Panel 2, DOE would convert the South Portal to an air intake, which would stop releases of radon gas from that location. For releases from the Wet Handling Facility and Aging Facility during normal operations, the maximally exposed noninvolved worker location would be in the surface geologic repository operations area and vicinity. For the period during operations when there would be surface and subsurface sources of radionuclides, the maximally exposed noninvolved worker location would be the South Portal development area because radon releases would contribute most of the total worker dose.

The analysis evaluated DOE workers at the Nevada Test Site as a potentially exposed noninvolved worker population. The analysis used the current Test Site population of 1,544 workers for dose calculations (DIRS 182717-Skougard 2007, all). The analysis assumed that all these workers would be at Mercury, Nevada, about 50 kilometers (31 miles) east-southeast of the proposed repository.

Figure D-2 shows the estimated numbers of workers (involved and noninvolved) as a function of time.

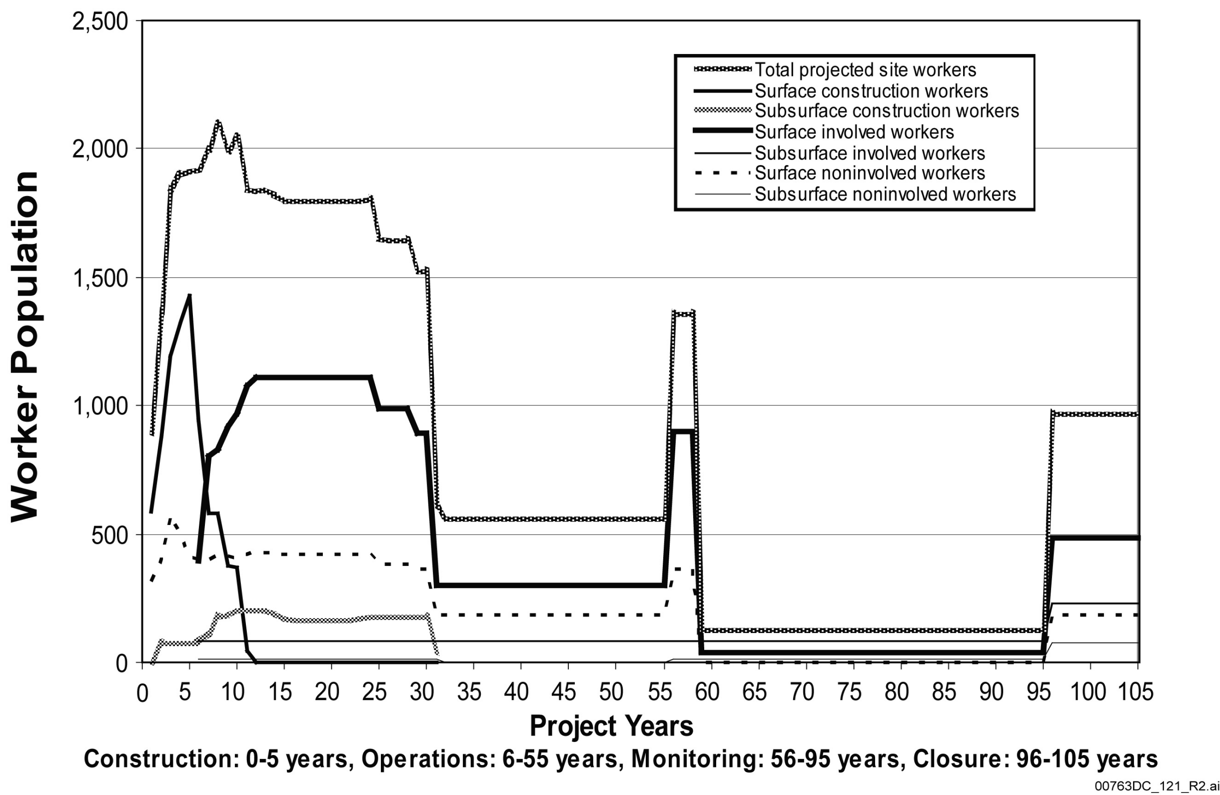

Figure D-2. Projected worker population for radiological impact assessment.

D.3.3 INVOLVED WORKERS

Involved workers would be craft and operations personnel who were directly involved in waste operations activities and subsurface development, which would include subsurface excavation; receipt, handling, packaging, aging, and emplacement of spent nuclear fuel and high-level radioactive waste; monitoring of the condition and performance of the waste packages; and closure. To assess radiological health impacts to involved workers, the analysis assumed they would receive 2,000 hours per year of occupational exposure at the repository. The method used to assess radiological doses to the maximally exposed involved workers and the worker population is described in Section D.4.2.

D.4 Radiological Doses

This section describes the potential radiological health impacts to workers and the general public from proposed repository activities. It includes descriptions of the calculations and results for estimation of impacts under normal conditions for the public and involved and noninvolved workers for each analytical period of the project (construction, operations, monitoring, and closure). Radiological impacts to workers would include those from naturally occurring and manmade radiation and from radioactive materials in the workplace. Radiological impacts to members of the public (offsite individual) would include those from potential exposure to airborne releases of naturally occurring radiation and manmade radionuclides.

This section lists and describes radiological impacts to workers and the public as doses to the maximally exposed members of the worker and public populations and population doses for all workers and the affected public population within 84 kilometers (52 miles) of the repository.

D.4.1 ESTIMATED PUBLIC AND NONINVOLVED WORKER DOSES

D.4.1.1 Estimated Doses from Atmospheric Releases

To calculate estimated dose to the maximally exposed offsite individual beyond the boundary of the analyzed land withdrawal area from manmade radionuclide releases, the analysis used the GENII computer program (DIRS 179907-Napier 2007, all) and biosphere model parameters developed for the entire Amargosa Valley (DIRS 177399-SNL 2007, all). GENII Version 2.05 calculates doses from exposure to radionuclides in the environment based on site-specific biosphere model parameters including various food consumption rates and periods and external and inhalation exposure times (DIRS 179907-Napier 2007, all). To estimate the maximum annual doses, the analysis assumed that the proposed repository would receive and process commercial spent nuclear fuel at the maximum annual receipt rate of 3,600 MTHM, which would be 20 percent more than the design throughput of 3,000 MTHM per year.

To calculate estimated collective dose to the public and the estimated dose from radon releases to the maximally exposed individual, the analysis used CAP88-PC, version 3 (DIRS 179923-Shroff 2006, all), a computer program that models atmospheric transport for assessment of dose and risk from radioactive air emissions. CAP88-PC is the EPA-approved computer program for demonstration of compliance for emissions from DOE facilities [40 CFR 61.93(a)]. EPA has validated CAP88-PC by comparing its predictions of annual average concentrations to actual environmental measurements at five DOE sites (DIRS 179923-Shroff 2006, Section 1.4). The program provides capabilities for radon release dispersion and exposure calculations that include calculation of radon decay product concentrations in working levels. It uses dose factors in accordance with Federal Guidance Report 13 (DIRS 175452-EPA 1999, all). EPA based the Report 13 factors on the methods in Publication 72 of the International Commission on Radiological Protection (DIRS 172935-ICRP 2001, all).

CAP88-PC requires meteorological data in the form of the joint frequency distribution of wind speed, direction, and atmospheric stability class. The analysis compiled these data from onsite meteorological measurements at Yucca Mountain from 2001 to 2005 at Air Quality and Meteorology Monitoring Site 1 (DIRS 183160-BSC 2007, all and Attachment III). Site 1 is a 60-meter (197-foot) tower about 1 kilometer (0.6 mile) south-southwest of the North Portal. The measurement heights are 10 meters (33 feet) and 60 meters (197 feet).

The analysis used the CAP88-PC program with the meteorological data along with the source terms in Section D.2 to calculate the unit dose factors listed in Table D-5. These individual and population unit dose factors are normalized for the various sources. For surface facility release, the table lists the factors per MTHM of processed fuel. Factors for radon releases are per unit curie of radon-222. Factors for

Table D-5. Unit dose factors for maximally exposed individuals and total population dose for normal operations releases.

Source/facility	Maximally exposed individuals[a] Offsite individual (millirem)	Noninvolved subsurface worker (millirem)	Noninvolved surface worker (millirem)	NTS worker (millirem)	Population dose within 84 kilometers (52 miles) (person-rem)
Subsurface facility per curie radon release	0.0016	0.0011	0.00097	0.000031	0.033
South Portal per curie radon release[b]		0.066			
Surface facility per MTHM SNF processed	0.000011[c]	0.000016[d]	0.000016[d]	0.000000025	0.00014
Subsurface facility per year operation (nonradon release)	0.0029[e]	0.010[f]	0.0099[f]	0.000028	0.033
Aging Facility per year operation	0.012[g]	0.013[d]	0.092[d]	0.00018	0.11

Notes: The analysis based doses on the CAP88-PC (DIRS 179923-Shroff 2006, all) calculation except where noted. Numbers are rounded to two significant figures.

a. Based on maximum total individual dose over the entire project duration.
b. South Portal release applicable only to construction analytical period.
c. Based on DIRS 185225-BSC 2008, Table 43.
d. Based on DIRS 185287-BSC 2008, Tables 12 and 13.
e. Based on DIRS 185225-BSC 2008, Table 44.
f. Based on DIRS 185287-BSC 2008, Table 14.
g. Based on DIRS 185225-BSC 2008, Table 45.

NTS = Nevada Test Site.
MTHM = Metric tons of heavy metal.
SNF = Spent nuclear fuel.

other releases from the subsurface facilities are per year of operation. The analysis used the factors in Table D-5 to calculate doses from every year of repository operation and during each analytical period.

The analysis calculated individual and population doses for every year of the project duration from the beginning of construction to the end of closure. To estimate the expected annual doses, the analysis assumed the proposed repository would receive and process spent nuclear fuel and high-level radioactive waste at the design throughput. Multiplying the unit dose factors in Table D-5 by the projected annual spent nuclear fuel processing rate for the repository yielded the expected annual individual and population doses. The analysis calculated cumulative or time-integrated doses by summing the yearly doses.

Figure D-3 shows the annual individual and population doses to the public and noninvolved workers as a function of time predicted for each year using the 105-year analysis period.

D.4.1.2 Estimated Doses to Workers from Direct Radiation

With the exception of subsurface involved workers, potential direct radiation exposures would originate only from surface facilities because massive layers of rock would shield workers from radiation sources such as waste packages inside subsurface facilities. Surface facilities with potential radiation sources that could contribute direct exposures to workers would include the transportation cask staging areas and the commercial spent nuclear fuel aging pads. All other surface facilities that handled radiological materials

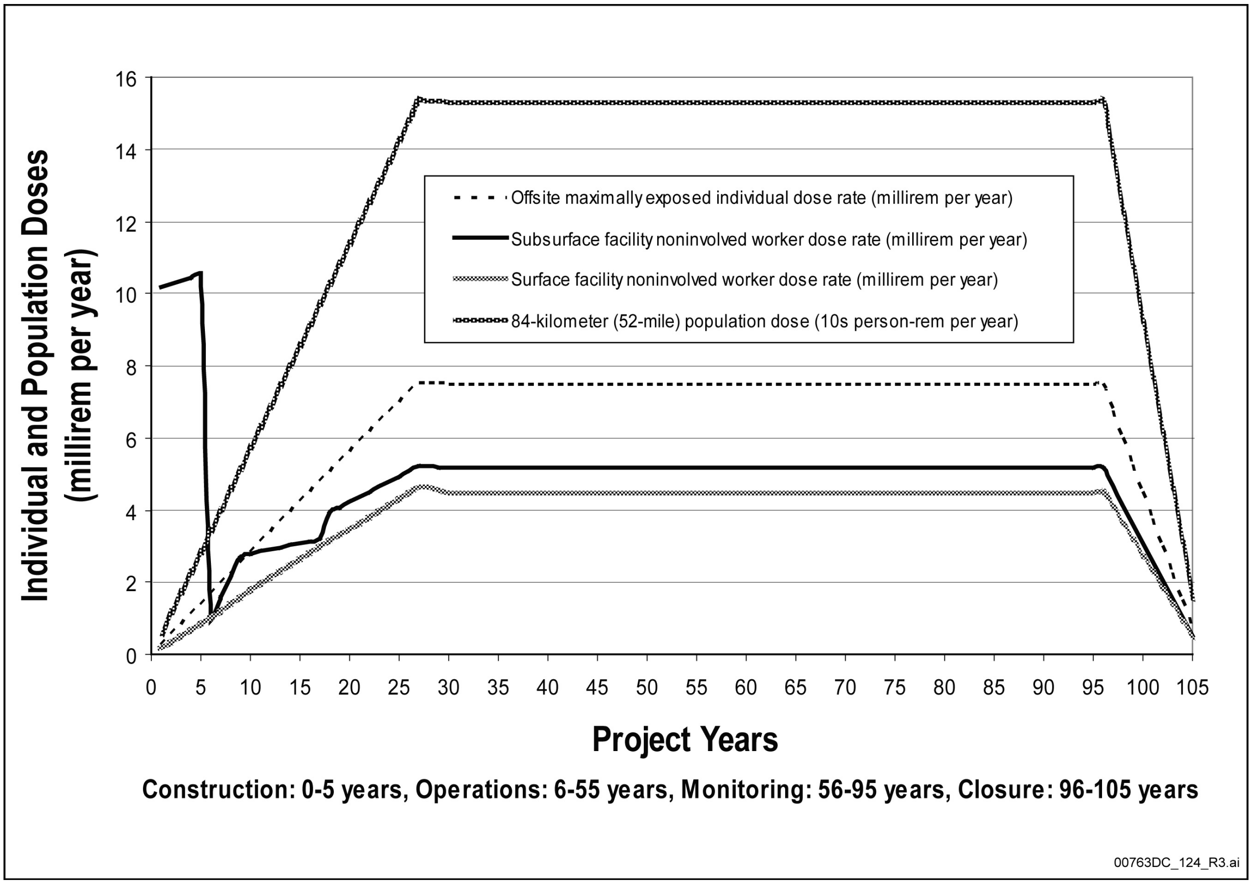

Figure D-3. Estimated individual and population doses from normal operations releases.

would provide concrete shielding for radiation sources, so dose rates at potentially occupied areas would be negligible.

The analysis used dose-rate-versus-distance information (DIRS 182886-BSC 2007, Tables 4, 5, 6, 7, and 8) and relative distances of the worker locations from various cask aging areas to calculate dose rates at worker locations from exposure to external radiation. It used dose rate-versus-distance information based on an aging overpack surface dose rate of 40 millirem per hour (DIRS 182886-BSC 2007, Section 3.2.8) and relative distances of the worker locations from each aging pad to estimate dose rates at worker locations from exposure to commercial spent nuclear fuel on the aging pads.

The total estimated dose rate at a worker location would be the sum of all doses from casks temporarily stored at designated staging and aging areas. For conservatism, the analysis did not consider radiation shielding from construction materials or temporary shielding that DOE would provide for construction and operations activities. The calculated maximum annual dose and total dose for the entire operations analytical period to a full-time noninvolved worker would be 10 millirem per year and 60 millirem, respectively. The total population dose to noninvolved workers over the entire operations period would be 20 person-rem. The analysis based the dose estimate over the operations period on the projection of annual commercial spent nuclear fuel processing rate and the capacity of the Aging Facility.

D.4.1.3 Estimated Total Public and Noninvolved Worker Doses from Normal Operations

Table D-6 summarizes estimates of radiation doses to members of the public and noninvolved workers for each analytical period from normal operations.

Table D-6. Estimated radiation doses to the public and noninvolved workers for each analytical period.[a]

Impact category	Construction	Operations	Monitoring	Closure
Maximum individual annual dose (millirem per year)				
Member of the public[b]	1.4	7.6	7.5	7.5
Noninvolved surface facility worker	0.83	15	4.5	4.5
Noninvolved subsurface facility worker	11	4.8	5.2	5.2
Maximum individual period total dose (millirem)				
Member of the public[b]	4.2	310	300	41
Noninvolved surface facility worker	2.5	250	180	25
Noninvolved subsurface facility worker	52	220	210	28
Population dose (person-rem)				
Exposed 84-kilometer (52-mile) population[c]	85	6,400	6,100	840
Noninvolved onsite population	4.7	190	26	18
Noninvolved Nevada Test Site population	0.12	9.2	8.9	1.2

Note: Numbers are rounded to two significant figures.

a. About 99.8 percent of the dose and impact to the offsite public would be from naturally occurring radon-222 and its decay products.

b. A hypothetical individual who would reside continuously at a location in the prevailing downwind direction from the repository in the unrestricted public access area that could receive the highest radiation exposure.

c. The projected population would include about 117,000 individuals within 84 kilometers of the repository.

D.4.2 ESTIMATED INVOLVED WORKER DOSES

Involved worker radiation exposure at proposed repository facilities from normal operations could result from cask, fuel, and waste package handling; routine maintenance of the facilities; and airborne releases. In the subsurface repository, additional exposure could result from exposure to naturally occurring ambient radiation fields and elevated concentrations of radon-222 and its decay products.

The primary sources of radiation exposure to involved workers would be:

- Internal and external exposure of workers to naturally occurring radionuclides that would include:
  - Internal exposure by inhalation of radon-222 and its decay products in the air (subsurface workers could receive exposure from elevated concentrations of radon-222 and its decay products in the air in the repository drifts; workers on the surface could receive exposure to radon-222 releases from the subsurface ventilation exhausts), and
  - Direct external exposure of workers in the repository drifts as a result of naturally occurring radionuclides in the rocks of the drift walls (primarily potassium-40 and radionuclides of the naturally occurring uranium and thorium decay series);
- Internal and external exposure of workers to potential releases to air of radionuclides during handling of spent nuclear fuel in the repository; and

- External exposure of workers to direct radiation from contained sources, such as transportation casks, aging overpacks, and loaded waste packages during handling and packaging at the surface facilities and after emplacement in the subsurface facilities.

D.4.2.1 Estimated Doses from Naturally Occurring Radionuclides

D.4.2.1.1 *Ambient External Radiation*

Workers in the subsurface facility could receive exposure to external radiation from naturally occurring radionuclides in the drift rock. The analysis used an average ambient external radiation dose rate of 5 millirem per year (Chapter 3, Section 3.1.8) for a worker underground exposure time of 2,000 hours per year to calculate worker doses from ambient external radiation in the subsurface repository.

D.4.2.1.2 *Inhalation of Radon-222 and its Decay Products*

The analysis used predicted radon-222 and decay product concentrations for the subsurface repository (DIRS 167021-BSC 2003, Table 5) to estimate potential dose rates for a subsurface worker from inhalation of radon-222 and its decay products. The predicted average concentrations in potentially occupied areas in the subsurface environment would be 5.8 picocuries per liter and 0.012 Working Level, respectively. The 0.012 Working Level concentration converts to the worker exposure unit of 0.14 Working Level Months per year based on 2,000 hours per year of exposure. To convert Working Level Months to rem, the analysis applied a conversion factor of 0.5 rem (500 millirem) per Working Level Month for inhalation of radon decay products (DIRS 103279-ICRP 1994, p. 24).

Table D-7 lists estimated doses to involved workers for each analytical period. The estimates include potential doses to the maximally exposed involved worker and the total dose for all involved workers from exposure to natural radiation sources.

Table D-7. Estimated radiation doses to involved workers from natural sources for each analytical period.[a]

Impact category	Construction	Operations	Monitoring	Closure
Maximum individual annual dose (millirem per year)				
Surface facility	0.83	4.5	4.5	4.5
Subsurface facility	120	120	120	120
Maximum individual period total dose (millirem)				
Surface facility	2.5	190	180	25
Subsurface facility	490	6,100	4,900	1,200
Population dose (person-rem)				
Total worker population	33	910	390	320

Note: Numbers are rounded to two significant figures.

a. Doses from exposure to radon-222, its decay products, and ambient radiation.

D.4.2.2 Estimated Doses from Airborne Releases

The analysis used the calculated annual average atmospheric dispersion factors (DIRS 183739-BSC 2007, Table 32), the predicted quantity of radionuclide releases (Table D-3), and the projected spent nuclear fuel processing rate at the proposed repository to estimate annual doses to repository workers from potential Wet Handling Facility and Aging Facility normal operational releases. The annual average dispersion factors represent the average dilution of airborne contamination from atmospheric mixing and turbulence; the analysis used the site-specific atmospheric conditions, the relative distance and configuration of the release point, and the receptor of interest to calculate the dispersion factors.

Involved worker doses from airborne releases would include releases of manmade radionuclides through the subsurface ventilation exhaust. These releases could occur as a result of neutron activation of the air and dust and resuspension of radioactive surface contamination on waste packages. They would be the only airborne releases of manmade radionuclides during the monitoring and closure analytical periods because the Wet Handling Facility and Aging Facility would no longer be operating.

Table D-8 lists estimated radiological doses to involved workers from potential normal operational releases for each analytical period. The estimated doses include potential doses to the maximally exposed involved worker and the total for all workers.

Table D-8. Estimated radiation doses to involved workers from manmade radionuclide releases during each analytical period.[a,b]

Impact category	Operations	Monitoring	Closure
Maximum individual annual dose (millirem per year)			
Surface facility	15[c]	0.0099	0.0099
Subsurface facility	0.097	0.036	0.036
Maximum individual period total dose (millirem)			
Surface facility	270	0.19	0.026
Subsurface facility	3.1	1.4	0.20
Total worker population dose (person-rem)	1.5	0.15	0.13

Note: Numbers are rounded to two significant figures.
a. Doses incurred from exposure to both surface and subsurface normal operations releases.
b. There would be no manmade radionuclide releases during the construction analytical period.
c. Doses based on a maximum annual receipt rate of 3,600 metric tons of heavy metal.

D.4.2.3 Estimated Doses from Direct Radiation

The analysis assessed annual doses to repository workers from exposure to direct radiation emitted from contained sources, such as transportation casks and waste packages, during normal operations for each of the following repository facilities:

- Receipt Facility,
- Initial Handling Facility,
- Wet Handling Facility,
- Canister Receipt and Closure Facilities,
- Subsurface facility,
- Aging Facility,
- Low-Level Waste Facility, and
- Cask Receipt Security Station.

With the exception of the Low-Level Waste Facility, dose assessments derive from the current facility general arrangement and projections of annual transportation cask, TAD canister, and waste package processing rates with the current simulated throughput model. The Low-Level Waste Facility would collect, package, and ship low-level radioactive waste to an approved disposal facility.

The dose assessments for this Repository SEIS evaluated the various worker groups and used time-motion inputs to determine estimated dose rates at various worker locations. For the surface facility dose assessments, the analysis assumed that all of the commercial spent nuclear fuel handled at the repository would have the radiological characteristics of design basis commercial spent nuclear fuel. This design basis fuel would be represented by pressurized-water reactor fuel with a burnup of 60,000 megawatt-days per MTHM, initial enrichment of 4 percent, and cooling or aging time of 10 years after removal from the reactor (DIRS 161120-BSC 2002, Section 5.5). For the radiation shielding analysis, the characteristics of design-basis fuel bound those of the representative spent nuclear fuel developed for repository normal operation airborne releases. The assessments considered all major activities, the types and numbers of involved workers in each activity, the duration of exposure, and the dose rate during that exposure period for each worker. The analysis calculated doses for a unit campaign—that is, for a typical received transportation cask and a delivered TAD canister or waste package. The estimated annual doses to the facility workers are the product of the unit campaign doses and the projected bounding number of campaigns during a year.

The calculated doses include the contributions from direct external radiation and airborne radionuclides. Calculation results indicate that the inhalation and submersion doses would represent a small fraction of the total worker doses. The analysis calculated total worker population doses from the total number of cask and waste package campaigns over the entire operations analytical period. Table D-9 lists the estimated surface worker doses during the operations period. There would be no direct external radiation exposure to surface workers during the construction, monitoring, and closure analytical periods. Table D-10 summarizes the estimated subsurface worker doses during the operations, monitoring, and closure periods. The estimated doses in Tables D-9 and D-10 include potential doses to the maximally exposed involved worker for each repository facility and the population total for all involved workers. The total estimated worker population doses for all surface and subsurface activities during the operations period would be 2,600 person-rem and 510 person-rem, respectively. The largest contributions to individual and population doses would be preparation of casks and the transfer of casks to waste processing and storage areas in surface facilities.

These conservative estimates of involved worker doses do not take credit for the application of administrative limits to reduce individual exposures. DOE would apply additional measures to ensure that radiation exposures to workers were as low as reasonably achievable. These dose reduction measures would include the application of refined shielding design in handling activities, rotation of crew members to other handling facilities, optimization of crew sizing, rotation of functional tasking in a crew, and applications of more remote operations and development of refined handling tools. Further reduction in worker doses would occur through continued application of experience-based improvements in handling operations through good radiation protection planning and practice and application of lessons learned (DIRS 184957-BSC 2008, p. 8).

D.4.3 ESTIMATED TOTAL RADIOLOGICAL DOSES FOR ENTIRE PROJECT

This section summarizes the total radiological doses to workers and members of the public from activities at the proposed Yucca Mountain Repository. The entire project would last 105 years and include a 5-year construction analytical period, 50-year operations analytical period, 50-year monitoring analytical period, and 10-year closure analytical period, which would overlap the last 10 years of the monitoring period.

Table D-9. Estimated radiation doses to involved surface workers from manmade external radiation during the operations analytical period.

Facility	Impact category[a,b]	Dose
Receipt Facility	Maximum annual individual dose (rem/year)	1.3
	Total individual dose (rem)	30
	Total population dose (person-rem)	840
Initial Handling Facility	Maximum annual individual dose (rem/year)	0.80
	Total individual dose (rem)	19
	Total population dose (person-rem)	110
Wet Handling Facility	Maximum annual individual dose (rem/year)	0.40
	Total individual dose (rem)	9.3
	Total population dose (person-rem)	300
Canister Receipt and Closure Facilities	Maximum annual individual dose (rem/year)	0.29
	Total individual dose (rem)	6.8
	Total population dose (person-rem)	630
Aging Facility	Maximum annual individual dose (rem/year)	0.30
	Total individual dose (rem)	7.0
	Total population dose (person-rem)	200
Low-Level Waste Facility	Maximum annual individual dose (rem/year)	0.70
	Total individual dose (rem)	16
	Total population dose (person-rem)	310
Cask Receipt Security Station	Maximum annual individual dose (rem/year)	0.40
	Total individual dose (rem)	9.3
	Total population dose (person-rem)	230
Total surface repository operations	Population dose (person-rem)	2,600

Source: DIRS 184957-BSC 2008, Table 1.0.
Note: Numbers are rounded to two significant figures; therefore, total might differ from sums.
a. Annual doses based on processing 500 casks per year, or about 3,000 MTHM of commercial spent nuclear fuel throughput per year.
b. Total doses based on processing a total waste throughput of 70,000 MTHM.
MTHM = Metric tons of heavy metal.

Table D-10. Estimated radiation doses to involved subsurface workers from manmade external radiation during each analytical period.[a,b]

Impact category	Operations	Monitoring	Closure[c]
Maximum annual individual dose (millirem per year)	210	200	39
Total individual dose (rem)	10	8	0.39
Total population dose (person-rem)	510	510	80

Source: DIRS 185337-BSC 2007, Sections 6.2 and 6.3.
Note: Numbers are rounded to two significant figures.
a. Doses incurred from loaded waste packages inside the subsurface drifts.
b. There would be no manmade external radiation sources during the construction analytical period.
c. Doses incurred from backfill operations.

Table D-11 summarizes estimates of radiological doses to the public for each analytical period and for the entire project duration. It lists estimated radiation doses for the maximally exposed member of the public and the potentially exposed population. About 99.8 percent of the potential doses would be from exposure to naturally occurring radon-222 and its decay products released in subsurface exhaust ventilation air. Estimated individual doses would be for the offsite maximally exposed member of the public who resided continuously for 70 years at the site boundary location in the prevailing downwind direction. The highest annual radiation dose would be 7.6 millirem, which is less than 4 percent of the annual average natural background radiation exposure of 340 millirem per year to members of the public

Table D-11. Estimated radiation doses to the public during each analytical period and entire project duration.[a]

Impact category	Construction	Operations	Monitoring[a]	Closure	Entire project[b]
Maximally exposed member of the public[c]					
Maximum annual dose (millirem per year)	1.4	7.6	7.5	7.5	7.6
Total dose (millirem)	4.2	310	300	41	530[d]
Population[e] dose (person-rem)	85	6,400	6,100	840	13,000

Note: Numbers are rounded to two significant figures; therefore, totals might differ from sums.

a. Doses are for the monitoring analytical period under active ventilation operating mode.

b. About 99.8 percent of the dose and impact would be from naturally occurring radon-222 and its decay products.

c. A hypothetical individual who would reside continuously at a location in the prevailing downwind direction from the repository in the unrestricted public access area that could receive the highest radiation exposure.

d. Based on a 70-year continuous exposure of the maximally exposed individual.

e. The projected population includes about 117,000 individuals within 84 kilometers (52 miles) of the repository.

(Chapter 3, Section 3.1.8.1). This 340-millirem-per-year dose includes a 200-millirem dose from ambient background levels of naturally occurring radon-222 and its decay products (Chapter 3, Section 3.1.8.2).

The estimated collective dose for the population within 84 kilometers (52 miles) of the repository for the entire project duration of 105 years would be 13,000 person-rem. This population dose can be compared with about 2.5 million person-rem the projected population in 2067 of about 117,000 persons within 84 kilometers of the repository would receive from natural background radon exposure.

Table D-12 lists estimates of radiological doses to workers for each analytical period and for the entire project. The estimated radiological doses include potential doses to involved workers, noninvolved workers, and the total for all workers. The table lists estimated doses for the maximally exposed involved worker and for the involved worker population; doses for the maximally exposed noninvolved worker and for the noninvolved worker population; and the estimated population doses for the combined population of workers. The estimated total worker population radiation dose for the entire project duration of 105

Table D-12. Estimated radiation doses to workers during each analytical period and entire project duration.

Worker group and impact category	Construction[a]	Operations	Monitoring[b]	Closure	Entire project
Maximum individual annual dose (rem per year)					
Surface facility involved worker	0.00083	1.3	0.0045	0.0045	1.3
Subsurface facility involved worker	0.12	0.33	0.33	0.16	0.33
Onsite noninvolved worker	0.011	0.015	0.0052	0.0052	0.015
NTS noninvolved worker	0.000026	0.00014	0.00014	0.00014	0.00014
Maximum individual period total dose (rem)					
Surface facility involved worker	0.0025	30	0.18	0.025	30
Subsurface facility involved worker	0.49	17	13	1.6	17
Onsite noninvolved worker	0.052	0.25	0.21	0.028	0.25
NTS noninvolved worker	0.000079	0.0059	0.0057	0.00078	0.0059
Population dose (person-rem)					
Surface facility involved worker	0.0	2,800	0.040	0.048	2,800
Subsurface facility involved worker	33	1,400	890	400	2,700
Onsite noninvolved worker	4.7	190	26	18	240
NTS noninvolved worker	0.12	9.2	8.9	1.2	19
Total worker population	38	4,400	930	420	5,800

Note: Numbers are rounded to two significant figures; therefore, totals might differ from sums.

a. Only subsurface workers have potential for measurable radiation dose from natural sources.

b. Doses are for the monitoring analytical period under active ventilation operating mode.

NTS = Nevada Test Site.

years would be 5,800 person-rem. About 76 percent of the dose would occur during the operations analytical period for the repository workforce. The principal source of exposure would be external radiation from handling of spent nuclear fuel in surface facilities and monitoring and maintenance activities in the subsurface facility. Exposure to the naturally occurring radioactive sources would account for 29 percent of the total worker dose. Inhalation of radon-222 and its decay products by subsurface workers would contribute 17 percent of the total dose, and ambient radiation exposure to subsurface workers would contribute 12 percent. To put the 5,800-worker person-rem occupational risk in perspective, the estimated workforce at 86,000 full-time equivalent worker years for the entire project duration of 105 years would receive 29,000 person-rem from natural background radiation exposure of 340 millirem per year (Chapter 3, Section 3.1.8.1). Therefore, the addition of 5,800 person-rem would represent a 20-percent increment.

D.5 Preclosure Radiological Human Health Impacts

To calculate the potential impacts to human health from the estimated radiation doses, the analysis multiplied the doses from Tables D-11 and D-12 by the updated dose-to-health-risk conversion factors (Section D.1.6). The estimated potential radiological health impacts cover the entire project duration of 105 years. This section discusses radiological health impacts for the maximally exposed workers and member of the public as increases in the probabilities of latent cancer fatality from the received radiation doses, and it provides health impacts for exposed populations as the estimated numbers of latent cancer fatalities that could occur within the exposed populations. For this Repository SEIS, the analysis used the conversion factor of 0.0006 latent cancer fatality per person-rem to convert worker and public doses to health effects.

D.5.1 ESTIMATED HEALTH IMPACTS TO THE GENERAL POPULATION

Table D-13 summarizes estimates of radiological health impacts to the public for each analytical period and the entire project duration. It lists estimated health effects for the offsite maximally exposed member of the public and the potentially exposed population. As indicated in Section D.4.3, almost all of the potential health impacts would be from exposure to naturally occurring radon-222 and its decay products released in subsurface exhaust ventilation air.

Table D-13. Estimated radiological health impacts to the public for each analytical period and entire project duration.[a]

Health impact	Construction	Operations	Monitoring[a]	Closure	Entire project[b]
Maximally exposed member of the public[c]					
Increase in probability of latent cancer fatality	0.0000025	0.00019	0.00018	0.000025	0.00032
Exposed 84-kilometer (52-mile) population[d]					
Number of latent cancer fatalities	0.051	3.8	3.7	0.51	8.0

Note: Numbers are rounded to two significant figures; therefore, totals might differ from sums.
a. Doses are for the monitoring analytical period under active ventilation operating mode.
b. About 99.8 percent of the dose and impact would be from naturally occurring radon-222 and decay products.
c. A hypothetical individual who would reside continuously at a location in the prevailing downwind direction from the repository in the unrestricted public access area that could receive the highest radiation exposure.
d. The projected population includes about 117,000 individuals within 84 kilometers of the repository.

The estimated increase in probability of a latent cancer fatality to the maximally exposed hypothetical individual who resided continuously for 70 years at the site boundary location in the prevailing downwind direction during the preclosure period would be about 0.0003. The estimated number of latent cancer fatalities would be 8 in a projected population in 2067 of about 117,000 persons within 84 kilometers (52 miles) of the repository. For comparison, the analysis examined the number of expected cancer deaths that would occur from other causes in the same population during the same period. The analysis calculated the expected number of cancer deaths that would not be related to the repository project on the basis of current statistics from the Centers for Disease Control and Prevention, which indicated that 24 percent of all deaths in the State of Nevada were attributable to cancer of some type and cause during 1998 (DIRS 153066-Murphy 2000, p. 8). Therefore, the increased risk to this projected population would be about 0.02 percent.

D.5.2 ESTIMATED HEALTH IMPACTS TO WORKERS

Table D-14 summarizes estimates of radiological health impacts to workers for each analytical period and for the entire project duration. It lists estimated radiological health impacts for the maximally exposed involved worker and the involved worker population, the maximally exposed noninvolved worker and the noninvolved worker population, and the combined population of workers.

Table D-14. Estimated radiological health impacts to workers for each analytical period and entire project duration.

Worker group/health impact	Construction	Operations	Monitoring[a]	Closure	Entire project
Increase in probability of latent cancer fatality for the maximally exposed worker[b]					
Involved	0.00029	0.018	0.0078	0.00097	0.018
Noninvolved	0.000031	0.00015	0.00012	0.000017	0.00015
Number of latent cancer fatalities in worker population					
Involved	0.020	2.5	0.54	0.24	3.3
Noninvolved	0.0028	0.12	0.016	0.011	0.14
Nevada Test Site noninvolved	0.000074	0.0055	0.0053	0.00073	0.012
Total	0.023	2.6	0.56	0.25	3.5

Note: Numbers are rounded to two significant figures; therefore, totals might differ from sums.

a. Health effects are for the monitoring analytical period under an active ventilation operating mode.

b. Worker health impacts are based on 2,000 hours per year exposure time over each analytical period up to a maximum of 50 years. Exposure locations are based on site layout of the repository.

The estimated increase in number of latent cancer fatalities that could occur in the repository workforce from the received radiation doses over the entire project would be 3.5. This can be compared with the 17 latent cancer fatalities that the same worker population would normally incur over the entire project duration of 105 years from exposure to natural background radiation of 340 millirem per year (Chapter 3, Section 3.1.8.1).

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Appendix E

Potential Repository Accident Scenarios and Sabotage: Analytical Methods and Results

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E. POTENTIAL REPOSITORY ACCIDENT SCENARIOS AND SABOTAGE: ANALYTICAL METHODS AND RESULTS

This appendix describes the methods and detailed results of the analysis the U.S. Department of Energy (DOE or the Department) performed for this *Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F-S1) (Repository SEIS) to assess the potential impacts from hypothetical accident and sabotage scenarios at the repository. The scenarios and methods apply only to repository accidents that could occur during operations, monitoring, and closure. This appendix describes the details of calculation methods for specific scenarios that the analysis determined to be credible. Appendix G describes the analytical methods and results for estimation of impacts from accidents that could occur during loading activities at the 72 commercial and 4 DOE sites and during transportation of materials to the repository.

DOE based the accident scenarios in the *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F; DIRS 155970-DOE 2002, all) (Yucca Mountain FEIS) on the information available at the time about the repository design. The analysis of the impacts relied on assumptions and analyses DOE selected to ensure that it did not underestimate the impacts from accident scenarios. Since the completion of the Yucca Mountain FEIS, the repository design and associated construction and operational plans have continued to evolve, and additional information and updated analytic tools relevant to estimating potential environmental impacts have become available. DOE would now use phased construction of multiple surface facilities, and most of the commercial spent nuclear fuel would arrive in transport, aging, and disposal (TAD) canisters. DOE has reevaluated the potential for repository accidents for this Repository SEIS. In addition, the Department has identified accident scenarios (1) to evaluate impacts to support the application for construction authorization and (2) to assess whether the repository would comply with regulatory limits on radiation exposure to workers and the public from accidental releases of radionuclides.

Section E.1 describes the general methodology for the accident analysis and Section E.2 describes the selection of accident scenarios for analysis. Sections E.3 and E.4 discuss source terms and consequences for the analyzed accident scenarios, respectively. Sections E.5 and E.6 discuss accidents in relation to monitoring and closure, and Inventory Modules 1 and 2, respectively. Section E.7 discusses the scenario DOE chose to represent a potential sabotage event.

E.1 General Methodology

This analysis incorporates, as appropriate, accident analyses DOE has prepared since completion of the Yucca Mountain FEIS to account for revised data and changes in analytical methods for consequence analyses. Section E.7 describes the scenario DOE chose to represent a hypothetical sabotage event and the potential consequences of that scenario.

Because of the large amount of radioactive material workers would handle at the proposed repository (Chapter 2, Section 2.1), the focus of the analysis was on accident scenarios that could cause the release of radioactive material to the environment. DOE analyzed selected accident scenarios to determine the amount of radioactive material an accident could release to the environment and to estimate the consequences of the release in terms of health effects to workers and the public. The accident scenarios DOE selected include a spectrum of both high-frequency, low-consequence accident scenarios and low-frequency, high-consequence accident scenarios in accordance with DOE *National Environmental Policy Act* (NEPA; 42 U.S.C. 4321 et seq.) guidance (DIRS 178579-DOE 2004, p. 27).

The analysis derived accident frequency estimates to establish the credibility of an accident scenario (that is, to determine whether an accident scenario is reasonably foreseeable). For these accident scenarios that DOE determined to be reasonably foreseeable, DOE estimated the potential consequences, which are presented without discounting for accident frequency (in other words, DOE did not multiply the consequences by the estimated frequencies to derive point estimates of risks). Estimates of accident frequency are inherently uncertain. Based on the available design information, DOE used the accident analysis approach this appendix describes to ensure it would not underestimate potential accident impacts.

For accidents that do not involve radioactive materials, the analysis determined that application of accident statistics from other DOE operations would provide a reasonable estimate of nonradiological accident impacts (Section E.2.2).

E.2 Potential Operations Accident Scenarios

The analysis identified potential repository accident scenarios for preclosure operations by using scenarios DOE has developed for the repository design in several reports that categorized event sequences (DIRS 180095-BSC 2008, all; DIRS 180096-BSC 2008, all; DIRS 180098-BSC 2008, all; DIRS 180099-BSC 2008, all; DIRS 180100-BSC 2008, all; DIRS 180101-BSC 2008, all; DIRS 183621-BSC 2008, all). Section E.2.1 describes the radiological accident scenarios, all of which would apply during operations activities. Section E.2.2 discusses the treatment of nonradiological accidents.

E.2.1 RADIOLOGICAL ACCIDENT SCENARIOS

Radiological accidents involve an initiating event that could lead to a release of radioactive material to the environment. The analysis considered accident scenarios separately for two types of initiating events: (1) internal initiating events that would originate in the repository and involve equipment failure, human error, or both, and (2) external initiating events that would originate outside the facility and affect the ability of the facility to maintain confinement of radioactive or hazardous material.

E.2.1.1 Internally Initiated Events

As noted in Section E.2, several reports provide the most recent repository accident scenario analysis for internal and external events that would involve receipt, handling, or emplacement of spent nuclear fuel and high-level radioactive waste. These documents address U.S. Nuclear Regulatory Commission (NRC) requirements in 10 CFR 63.112 and preclosure performance objectives in 10 CFR 63.111. The reports represent a comprehensive evaluation of repository operations to identify accident sequences that could lead to a radioactive release. DOE performed detailed analyses on the sequences using event trees and fault trees to estimate accident frequencies. As required by 10 CFR Part 63, the analysis used the frequency evaluation to identify (1) Category 1 events (sequences that would be likely to occur one or more times before permanent closure), (2) Category 2 events (sequences that would have at least a 1-in-10,000 chance of occurring before permanent closure), or (3) beyond-design-basis Category 2 events (which would have less than a 1-in-10,000 chance of occurring before permanent closure). The period before permanent closure includes a period up to 50 years for receipt, handling, or emplacement operations (DIRS 176678-DOE 2006, p. 4-6). For Category 1 events that could happen during these operations, the average annual frequency threshold would be approximately 1 in 50, or 0.02 per year. The total period of activity before permanent closure would be 100 years, so the average annual frequency threshold for events that could occur anytime before permanent closure would be 0.01 per year. Similarly, the Category 2 event threshold is 2.0×10^{-6} per year (1 in 10,000 divided by 50) for events that could occur only during receipt, handling, or emplacement operations. The event categorization analysis identified a number of beyond-Category-2 events that DOE eliminated from further consideration. However, DOE NEPA guidance recommends consideration of these events for evaluation if (1) they have an annual frequency above 1.0×10^{-7} per year, and (2) the consequences could be very large (DIRS 178579-DOE 2004, p. 28). As discussed in Section E.2.1.1.7, none of these beyond-Category-2 event sequences have the potential to produce consequences greater than the aircraft crash evaluated as a sabotage event in Section E.7 and, therefore, DOE did not evaluate them further in this Repository SEIS.

The evaluations that identified the internal accident scenarios (DIRS 180095-BSC 2008, all; DIRS 180096-BSC 2008, all; DIRS 180098-BSC 2008, all; DIRS 180099-BSC 2008, all; DIRS 180100-BSC 2008, all; DIRS 180101-BSC 2008, all) did not quantitatively evaluate criticality events. DOE has performed a separate risk-informed, performance-based Preclosure Criticality Safety Analysis of waste forms (DIRS 181643-BSC 2008, all). This analysis concluded that preclosure criticality would be prevented for normal operations and for Category 1 and Category 2 event sequences (DIRS 181643-BSC 2008, p. 119). Criticality would be prevented by the use of neutron-absorbing materials in waste containers, control of moderator materials in the waste handling buildings, limiting the number of waste forms in proximity, and boration of the Wet Handling Facility storage pool. Therefore, DOE did not evaluate consequences of criticality accidents further.

Table E-1 lists the accident scenarios that DOE included in the analysis. The table lists the bounding accident scenarios (resulting in the highest radiological releases). The analysis did not identify any Category 1 scenarios. In addition, DOE performed a qualitative evaluation of beyond-Category-2 accident scenarios (Section E.2.1.1.7).

In the Draft Repository SEIS, the list of internal events (DIRS 183188-DOE 2007, Appendix E, Section E.2.1.1) included a Category 2 event that would involve a drop and subsequent breach of a naval canister that contained spent nuclear fuel. Since the publication of the Draft Repository SEIS, DOE has determined, based on additional analysis, that this is a Beyond-Category 2 event (DIRS 180096-BSC 2008,all). The Department based the additional analysis on the frequency of initiating events that could pose a threat to the integrity of a Navy canister; the number of handling operations involving naval canisters; and the robustness of the analyzed naval canisters to survive a drop, other impact events, fire events, seismic events, and other external events (DIRS 180096-BSC 2008, all). Therefore, this event does not appear in Table E-1. In addition, DOE has determined that a truck fire that involved a transportation cask would be a new Category 2 event that the Department identified since it issued the Draft Repository SEIS (DIRS 180100-BSC 2008, Section 6.8). DOE has added this event to Table E-1 (Scenario No. 12), as discussed in Section E.2.1.1.5.

The Scenario Number column in Table E-1 provides a numerical identifier. The Location column lists the repository location designator where the accident scenario could occur. The Description column describes the scenario. The Material at Risk column identifies the radioactive material the scenario would involve. The final column lists the estimated annual frequency for the scenario.

Table E-1. Evaluated accident scenarios.

Scenario number	Location	Description	Material at risk	Mean number of occurrences over preclosure period (mean annual frequency)[a]
1	Low-Level Waste Handling Operations[b]	Breach of containers with HEPA filter, pool filter, wet-solid resins; breach of HEPA ductwork	HEPA filters, pool filter, resins	6×10^{-2} (1×10^{-3})
2	Initial Handling Facility, Canister Receipt and Closure Facilities	Breach of sealed HLW canisters in a sealed transportation cask	5 HLW canisters	$< 1 \times 10^{-4}$ ($< 2 \times 10^{-6}$)
3	Canister Receipt and Closure Facilities[b]	Breach of sealed HLW canisters in unsealed waste package	5 HLW canisters	$< 1 \times 10^{-4}$ ($< 2 \times 10^{-6}$)
4	Initial Handling Facility, Canister Receipt and Closure Facilities	Breach of sealed HLW canisters during transfer (one drops onto another)	2 HLW canisters	1×10^{-2} (1×10^{-3})
5	Wet Handling Facility	Breach of uncanistered commercial SNF in an unsealed truck transportation cask in air	4 PWR or 9 BWR fuel assemblies	1×10^{-1} (2×10^{-3})
6	Wet Handling Facility[b]	Breach of uncanistered commercial SNF in an unsealed transportation cask in pool	4 PWR or 9 BWR fuel assemblies	7×10^{-4} (1.4×10^{-5})
7	Wet Handling Facility	Breach of sealed DPC in air	36 PWR or 74 BWR fuel assemblies	9×10^{-3} (2×10^{-4})
8	Wet Handling Facility[b]	Breach of commercial SNF in unsealed DPC in pool	36 PWR or 74 BWR fuel assemblies	2×10^{-4} 4×10^{-6}
9	Canister Receipt and Closure Facilities	Breach of a sealed TAD canister within facility	21 PWR or 44 BWR fuel assemblies	2×10^{-3} (4×10^{-5})
10	Wet Handling Facility[b]	Breach of commercial SNF in unsealed TAD canister in pool	21 PWR or 44 BWR fuel assemblies	5×10^{-4} (1×10^{-5})
11	Wet Handling Facility	Breach of uncanistered commercial SNF assembly in pool (one drops onto another)	2 PWR or 2 BWR fuel assemblies	3×10^{-1} (6×10^{-3})
12	Wet Handling Facility	Breach of uncanistered commercial SNF in pool	1 PWR or 1 BWR fuel assembly	$< 1 \times 10^{-4}$ ($< 2 \times 10^{-6}$)
13	Low-Level Waste Facility	Fire involving low-level radioactive waste	Filters, spent resin, dry active waste, liquid waste	7×10^{-2} (1×10^{-3})

Table E-1. Evaluated accident scenarios (continued).

Scenario number	Location	Description	Material at risk	Mean number of occurrences over preclosure period (mean annual frequency)[a]
14	Receipt Area, Wet Handling Facility	Fire involving truck transportation cask	4 PWR fuel or 9 BWR assemblies	2×10^{-2} (4×10^{-4})

a. Annual frequency is estimated by dividing the expected number of occurrences over the preclosure period by the preclosure operating interval of 50 years. Some scenarios could occur at more than one location. The frequency given is the highest estimated for any location. For accidence scenarios potentially initiated by only Beyond-Category-2 event sequences, the expected occurrence value is less than the maximum frequency of a Beyond-Category-2 event over the preclosure period (that is 1×10^{-4}).

b. These scenarios are initiated by seismic events and are discussed in Section E.2.1.2.2.

BWR = Boiling-water reactor.
DPC = Dual-purpose canister.
HEPA = High-efficiency particulate air (filter).
HLW = High-level radioactive waste.
PWR = Pressurized-water reactor.
SNF = Spent nuclear fuel.
TAD = Transportation, aging, and disposal (canister).

The waste forms that DOE would receive at the repository include commercial and DOE spent nuclear fuel and high-level radioactive waste. None of the event sequences in Table E-1 involves DOE spent nuclear fuel. This is because the Department intends to implement a safety strategy that would preclude a breach during handling of DOE spent nuclear fuel canisters (DIRS 185225-BSC 2008, p. 31).

DOE selected fuel from pressurized-water reactors to evaluate consequences for accident scenarios that could involve commercial spent nuclear fuel because it would be the most common type of fuel in the proposed repository (DIRS 155970-DOE 2002, Appendix A, p. A-15) and because it would produce higher doses than boiling-water reactor fuel for equivalent accident scenarios (Section E.3.3).

E.2.1.1.1 *Initial Handling Facility*

The Initial Handling Facility would receive high-level radioactive waste and naval spent nuclear fuel in canisters and transfer them from transportation casks to waste packages. The Initial Handling Facility would receive, package, and support emplacement of waste. Canister transfer operations would occur in concrete enclosures or the shielded canister transfer machine.

The Initial Handling Facility would interface with the other facilities as follows:

- Receive casks with high-level radioactive waste and naval spent nuclear fuel on transporters from the rail or truck buffer areas,
- Receive empty waste packages, lids, and shield plugs from the warehouse for the processing of the canisters, and receive support equipment for each waste package.

DOE analyzed accident Scenarios 2 and 4 in Table E-1 that could happen at the Initial Handling Facility. The Department retained Scenario 2 from the Draft SEIS to be consistent with the application for construction authorization (DIRS 185225-BSC 2008, p. 56); Scenario 3 was retained from the Draft

Repository SEIS even though it was found to be a Beyond-Category-2 event (DIRS 180096-BSC 2008, all).

While the Initial Handling Facility would have a filtered exhaust system with high-efficiency particulate air filters to mitigate the consequences of a radioactive release from a canister drop, the nature of the releases from a breached high-level radioactive waste canister does not require the filtration system to be important to safety.

E.2.1.1.2 *Receipt Facility*

The functions of the Receipt Facility would be to (1) receive loaded transportation casks, (2) remove impact limiters from the casks, and (3) transfer the TAD or vertical dual-purpose canister from the transportation cask into an aging overpack for movement to the Aging Facility. Horizontal dual-purpose canisters could be placed on a transfer trailer and moved to the Aging Facility where they are pushed into an aging overpack. The TAD could also be placed in an aging overpack and sent to the Canister Receipt and Closure Facilities for placement into a waste package or moved to the Wet Handling Facility for remediation if needed. Because the Canister Receipt and Closure Facilities can also directly receive TAD canisters in transportation casks, the primary function of the Receipt Facility would be to transfer TAD and dual-purpose canisters from transportation casks to the Aging Facility.

The Receipt Facility would receive only rail casks directly. It would not handle uncanistered spent nuclear fuel, and would not open canisters inside the facility. There would be direct rail access to the Receipt Facility.

The facility would consist of multipurpose cells for cask receipt for shielded handling of TAD and dual-purpose canisters, as well as the aging overpacks that held the canisters. The facility would accommodate the cask transporter for movement of the loaded aging overpacks. Casks containing horizontal dual-purpose canisters would be transferred in the cask receipt cell from a rail car to a transfer trailer and moved to the aging pad via a transfer trailer where the horizontal dual-purpose canister would be pushed into the aging overpack.

The receipt of TAD and most dual-purpose canisters and the transfer of these canisters to aging overpacks would utilize the vertical transfer method described in *Receipt Facility Reliability and Event Sequence Categorization Analysis* (DIRS 180099-BSC 2008, all). DOE would transfer casks containing horizontal dual-purpose canisters to the aging pad where the dual-purpose canister was pushed into the aging overpack. In this case, the dual-purpose canisters would be handled with a horizontal transfer method.

The Receipt Facility would have a filtered exhaust system with high-efficiency particulate air filters to mitigate the consequences of a radioactive release from a canister drop.

In evaluating potential hazards of operations in the Receipt Facility, DOE did not identify any Category 2 accident scenarios with the potential to release radioactive material (DIRS 180099-BSC 2008, Section 6.8).

E.2.1.1.3 *Wet Handling Facility*

Typical Wet Handling Facility operations would include:

1. Receive transportation casks with commercial spent nuclear fuel assemblies from truck or rail buffer areas. The Wet Handling Facility would handle commercial spent nuclear fuel as individual assemblies and in dual-purpose and TAD canisters.

2. Receive empty TAD canisters from the Warehouse and Non-Nuclear Receipt Facility for transfer into the pool for loading.

3. Prepare transportation casks for unloading by inspecting the cask; removing impact limiters; opening, sampling, and venting the cask; cooling the spent nuclear fuel; and unbolting the cask lid.

4. Transfer the cask into a pool for lid removal and transfer of commercial spent nuclear fuel to an empty TAD canister or to a staging rack in the pool. When unloaded, the transportation cask lid(s) would be installed, closed, and bolted in reverse sequence, and the empty transportation cask would be inspected and surveyed for contamination before transport back to the truck or rail buffer area.

5. Manage commercial spent nuclear fuel and blend fuel assemblies to ensure that the loaded TAD canister does not exceed thermal power limits. DOE would transfer a loaded TAD canister that exceeded the waste package thermal power emplacement limits to an aging pad to allow the thermal power to cool to the point where it could load the canister in a waste package and emplace it. The pool would provide limited staging capacity for fuel assemblies.

6. Close and seal-weld the loaded TAD canister and transfer it in a shielded transfer cask to a TAD canister closure station for draining of water from the interior, drying of the interior, evacuation, and helium backfilling. After these steps, the closed TAD canister would be ready for transfer to a Canister Receipt and Closure Facility in an aging overpack for loading in a waste package or in an aging overpack to the Aging Facility.

7. Open dual-purpose canisters and transfer the fuel from inside the dual-purpose canister to a TAD canister or to the staging rack in the pool.

8. Transfer TAD canisters from shielded transfer casks to aging overpacks and transfer vertical dual-purpose canisters from aging overpacks to shielded transfer casks.

The Wet Handling Facility would handle commercial spent nuclear fuel in dual-purpose and TAD canisters. Transportation casks with uncanistered commercial spent nuclear fuel would move directly into the Wet Handling Facility on the railcars or trucks that transported them to the repository. Rail transportation casks with dual-purpose canisters would move from the railcar buffer area directly into the facility. In addition, vertical dual-purpose canisters would be brought to the facility in aging overpacks from the Receipt Facility or the Aging Facility and horizontal dual-purpose canisters would be brought to the facility in shielded transfer casks from the Aging Facility. The facility would have a single pool to transfer commercial spent nuclear fuel from transportation casks and dual-purpose canisters to staging racks for eventual transfer to TAD canisters. Preparation of transportation casks for unloading in the Wet Handling Facility could require cooling of the casks before their immersion in the pool. A limited quantity of commercial spent nuclear fuel could be temporarily staged in racks in the pool. Normal handling operations would occur underwater or in a shielded transfer cask to protect operators from radiological hazards. The facility design includes a high-efficiency particulate air filtration exhaust system to mitigate the consequences of canister or fuel assembly drop events.

DOE identified Scenarios 5 through 7 and 10 through 12 (Table E-1) as accident scenarios applicable to operations in the Wet Handling Facility (DIRS 180098-BSC 2008, Section 6.8). The Department retained Scenario 12 from the Draft Repository SEIS even though it was found to be a Beyond-Category-2 event (DIRS 180098-BSC 2008, all).

E.2.1.1.4 *Canister Receipt and Closure Facilities*

Typical Canister Receipt and Closure Facility operations would include:

1. Receive transportation casks with spent nuclear fuel and high-level radioactive waste in disposable canisters (TAD and DOE spent nuclear fuel canisters other than naval spent nuclear fuel canisters, and high-level radioactive waste canisters). In addition, the facility would receive aging overpacks with TAD canisters from the Wet Handling Facility and aging overpacks with TAD canisters from the Aging Facility.

2. Prepare transportation casks for unloading by inspecting the cask; removing impact limiters; opening, sampling, and venting the cask; and unbolting the cask lid.

3. Transfer the contents of the transportation casks and aging overpacks to waste packages.

4. Transfer TAD and vertical dual-purpose canisters from transportation casks to aging overpacks. Horizontal dual-purpose canisters would be placed on a transfer trailer for movement to the aging pad.

5. Install lids on the unloaded transportation casks. The casks would be inspected, decontaminated, and surveyed before transport back to the rail buffer area.

6. Install the inner waste package lid and weld it closed; inspect and test the inner lid weld; evacuate the waste package and backfill it with helium; close and seal-weld the backfill port on the inner lid; inspect and test the backfill port closure weld; install the outer waste package lid and weld it closed; inspect, nondestructively examine, test, and stress-relieve the outer lid weld.

7. Inspect the completed waste package for physical condition and external radioactive contamination.

8. Transfer the waste package to the transport and emplacement vehicle.

Each Canister Receipt and Closure Facility would house two shielded, remote canister transfer machines where DOE would transfer TAD canisters from aging overpacks to waste packages. The Department would construct as many as three Canister Receipt and Closure Facilities, each with two waste package closure cells, which would house vertical waste package loading and closing operations. Each facility would have the capability to process TAD spent nuclear fuel canisters or DOE high-level radioactive waste canisters. All transportation casks with high-level radioactive waste and DOE and commercial spent nuclear fuel would move on rail cars or truck trailer directly from the rail buffer area to a Canister Receipt and Closure Facility. The facility would also receive TAD canisters in aging overpacks from the

Receipt Facility and Aging Facility. An overhead crane would upend and unload the transportation casks from the conveyance. Canister transfers would occur in a vertical orientation using a shielded canister transfer machine. A staging area would be in line with each process line.

The Canister Receipt and Closure Facilities would have high-efficiency particulate air filtration exhaust systems to mitigate the consequences of a canister drop.

DOE identified Scenarios 3, 4, and 9 (Table E-1) as Category 2 accident scenarios with the potential to release radioactive material resulting from operations in a Canister Receipt and Closure Facility (DIRS 180095-BSC 2008, Section 6.8).

E.2.1.1.5 *Intra-Site Operations and Balance of Plant Facilities*

Intra-site operations would include site transportation activities associated with movement of transportation casks from the geologic repository operations area boundary to buffer areas and waste handling facilities. They would also include transfer of aging overpacks and horizontal casks between the Aging Facility and waste handling facilities and among waste handling facilities. Balance of plant facilities include activities at the Aging Facility and management of low-level radioactive waste, including loading at collection areas, transfer to the Low-Level Waste Facility, and unloading and storage of solid and liquid radioactive waste at the Low-Level Waste Facility. Other balance of plant facilities would be the Emergency Diesel Generator Facility and support systems for geologic repository operations area operations and other nonnuclear facilities (craft shop, equipment yard, and heavy equipment maintenance facility).

DOE would use standard vehicular transport, such as open flatbed trucks, to move low-level radioactive waste from the surface and subsurface nuclear facilities to the Low-Level Waste Handling Facility. Shielding would be provided as needed. The waste would be stored at the facility in 55-gallon drums, boxes, and bags. It would be transferred from onsite storage at the Low-Level Waste Handling Facility to an offsite vendor for processing, disposal, or both at an approved facility. The Low-Level Waste Handling Facility would contain areas for the sorting and storage of waste.

DOE would place TAD canisters into aging overpacks at the Wet Handling Facility, the Receipt Facility, and Canister Receipt and Closure Facilities. The aging overpacks would then be transferred to the Aging Facility to age the waste until it was ready for emplacement or repackaging or to a Canister Receipt Closure Facility for transfer into a waste package. For emplacement, the TAD canisters would be removed from the aging overpacks and placed in a waste package. Vertical dual-purpose canisters could be placed in aging overpacks at the Receipt Facility and Canister Receipt and Closure Facilities and transferred to the Aging Facility. Casks containing horizontal dual-purpose canisters could be placed on a transfer trailer in the Receipt Facility and moved to the aging pad where the dual-purpose canisters would be pushed into an aging overpack. The Aging Facility would contain two aging pads with 2,500 spaces for storage of as much as 21,000 metric tons of heavy metal of waste. Chapter 2 of this Repository SEIS provides a more detailed description of aging operations.

DOE identified Scenarios 1 and 14 (Table E-1) as applicable to intra-site operations and balance of plant facilities (DIRS 180100-BSC 2008, Section 6.8).

E.2.1.1.6 *Waste Emplacement and Subsurface Facility Systems*

Waste packages would move from the Initial Handling Facility or a Canister Receipt and Closure Facility to the emplacement drifts on a rail-based transport and emplacement vehicle. The waste package would be inside the shielded enclosure of the transport and emplacement vehicle, which would descend the North Ramp and proceed to the predetermined emplacement drift. A third-rail electrical system would power the transport and emplacement vehicle. The transport and emplacement vehicle would have a battery for secondary power. Waste emplacement operations would include drip shield emplacement. DOE did not identify any accident scenarios for waste emplacement operations (DIRS 180101-BSC 2008, Section 6.8). However, the Yucca Mountain FEIS identified a transporter runaway accident scenario as a potential event with an estimated frequency of 1.2×10^{-7} per year (DIRS 155970-DOE 2002, Appendix H, p. H-5, Event 19), which is less than the Category 2 threshold of 2×10^{-6} per year. Section E.2.1.1.7 discusses this accident scenario.

E.2.1.1.7 *Beyond-Category-2 Accident Scenarios*

As noted above, DOE evaluated accident scenarios with frequencies of 2×10^{-6} per year or higher for compliance with offsite dose requirements. However, DOE NEPA guidance (DIRS 178579-DOE 2004, p. 28) recommends evaluation of scenarios with frequencies of 1×10^{-6} to 1×10^{-7} per year if the consequences could be very large. DOE determined in the Yucca Mountain FEIS (DIRS 155970-DOE 2002, Appendix H, p. H-36) that one scenario could fall into this category: runaway and derailment of the vehicle that would transport waste packages to the emplacement drifts. In this scenario, the waste package would be ejected from the transport vehicle and breached by impact with the ground, which would release radioactive material. DOE has replaced the transporter the Yucca Mountain FEIS evaluated with the transport and emplacement vehicle. The Department determined that the probability of a runaway event involving the transport and emplacement vehicle would be 1.7×10^{-5} during the preclosure period (DIRS 180101-BSC 2008, Table 6.0-2, p. 99), or about 3×10^{-7} per year for the 50-year preclosure operating period. This probability is greater (meaning that event is less likely to occur) than the threshold guidance DOE provided for reasonably foreseeable events. DOE determined that the consequences of the transporter runaway and derailment described in the FEIS were not "very large" because the calculated maximally exposed offsite individual dose for unfavorable meteorological conditions was only 3.8×10^{-2} rem (DIRS 155970-DOE 2002, Volume IV, Appendix H, p. H-37). The consequences of the transport and emplacement vehicle derailment described in this Repository SEIS are expected to be smaller than those of the transporter derailment described in the FEIS and thus would be bounded by it. Therefore, DOE did not evaluate this event further for this Repository SEIS.

Other Beyond-Category-2 events could occur at the repository. However, DOE determined that none would be likely to cause very large offsite consequences because most could occur only in waste handling buildings that had high-efficiency particulate air filtration systems, which would limit radionuclide releases. Even if these filtration systems failed, the resulting release would be unlikely to cause very large consequences because of the limited amount of material involved in the event and the retention of radionuclides by the building enclosure. Some of the remaining events could occur in the subsurface areas where a significant fraction of particulate radionuclides could deposit on surfaces during transport to the atmosphere. For those few accidents that could occur on the surface outside waste handling buildings, none would be likely to result in radioactive releases that resulted in very large offsite consequences because of the limited amount of material involved and the protection offered by enclosures (such as casks and the waste package) surrounding the waste forms (DIRS 180100-BSC 2008.all).

E.2.1.2 Externally Initiated Events

Externally initiated events would result either from causes external to the repository (such as earthquakes and high winds) or from natural processes that occurred over a long period in the repository (for example, corrosion and erosion). In the Yucca Mountain FEIS, DOE performed an evaluation to identify which of these events could initiate accidents at the repository with the potential for release of radioactive material. Based on this evaluation, DOE concluded that the only external events with a credible potential to release radionuclides of concern would be an aircraft crash and a large (beyond-design-basis) seismic event. The evaluation of both of these externally initialed events has evolved since completion of the FEIS and is described individually below.

E.2.1.2.1 *Aircraft Crash*

For the repository design, a recent DOE analysis determined that an aircraft crash into repository surface facilities would have a frequency of 5.9×10^{-7} per year (DIRS 180112-BSC 2007, p. 75). While this probability is below the frequency threshold of 2.0×10^{-6} per year and DOE does not need to consider it in the licensing process (Section E.2.1.1), it is above the DOE NEPA recommended threshold of 1×10^{-7} per year (DIRS 178579-DOE 2004, p. 28) if the consequences could be very large. Therefore, DOE performed a further evaluation of this scenario for this Repository SEIS.

The DOE aircraft crash probability assessment (DIRS 180112-BSC 2007, all) contained several conservative assumptions that tended to produce an upper-bound estimate. For this Repository SEIS, DOE undertook a more realistic evaluation. The conservative assumptions in the DOE assessment were:

- The TAD canister storage modules at the Aging Facility would be vulnerable to aircraft crash impacts.
- The entire footprint of each waste handling building would be vulnerable in case of an impact. However, only a fraction of the building floor areas would contain spent nuclear fuel and high-level radioactive waste during operations.
- The building walls would be vulnerable during the crash. However, the walls would be thick, reinforced concrete and could resist penetration during the crash.

The analysis for this Repository SEIS considered each of these assumptions separately, as follows:

- Aging Facility. The Aging Facility would consist of concrete pads on which DOE would place TAD and dual-purpose canister aging overpacks. The specification for the aging overpacks (DIRS 181403-DOE 2007, Section 3.3.2) states the module design would withstand the largest of the most likely aircraft impact, which would be an F-15 fighter aircraft with an impact speed of 152 meters (500 feet) per second. Therefore, DOE removed the storage modules as a target area from the aircraft crash frequency evaluation for this Repository SEIS.
- Building Footprint. The analysis reduced the building footprints to include only those areas that would handle spent nuclear fuel and high-level radioactive waste based on design drawings of areas shown to be vulnerable (DIRS 180278-BSC 2007, all; DIRS 180989-BSC 2007, all; DIRS 181268-BSC 2007, all; DIRS 184100-BSC 2007, all.). Table E-2 lists the dimension changes.

Table E-2. Surface waste handling building dimensions [meters (feet)] for aircraft crash frequency analysis.

Building	DOE frequency analysis[a]		Repository SEIS frequency analysis	
	Length	Width	Length	Width
Initial Handling Facility	91 (310)	51 (186)	67 (220)	11 (36)
Canister Receipt and Closure Facility	128 (420)	97 (318)	100 (330)	30 (98)
Receipt Facility	99 (320)	87 (290)	61 (200)	21 (69)
Wet Handling Facility	120 (390)	91 (299)	82 (270)	30 (98)

a. Source: DIRS 180112-BSC 2007, p. 62.

- Concrete walls. The concrete walls of the buildings would be 1.2 meters (4 feet) thick (180278-BSC 2007, all; DIRS 180989-BSC 2007, all; DIRS 181268-BSC 2007, all; DIRS 184100-BSC 2007, all.). The *DOE Standard, Accident Analysis for Aircraft Crash into Hazardous Facilities* (DIRS 101810-DOE 1996, p. 68, Equation 6-2) evaluates the potential for aircraft parts to penetrate concrete and recommends the following concrete penetration formula (derived in English units):

$$t_p = (U/V)^{0.25}(MV^2/Df_c)^{0.5} \quad \text{(Equation E-1)}$$

where

t_p = perforation thickness, or the concrete panel thickness that is just great enough to allow a missile to pass through the panel without any exit velocity (inches)
U = reference velocity (200 feet per second) (DIRS 101810-DOE 1996, p. 68)
V = missile impact velocity (aircraft impact velocity) (feet per second)
M = mass of the missile or the weight (pounds) divided by gravitational acceleration (32 feet per second)
D = missile diameter (inches)
f_c = ultimate compressive strength of the concrete (pounds per square foot)

Small military aircraft from Nellis Air Force Base dominate the probability for aircraft crash (DIRS 180112-BSC 2007, Section 6.5.3), and F-15 and F-16 jet fighters make up about 80 percent of the total flights. The aircraft parts with the highest chance of concrete penetration would be the jet engines and engine shafts (DIRS 101810-DOE 1996, p. 58). The characteristics of these engine parts that are relevant to Equation E-1 are an engine weight of about 4,200 pounds, an engine diameter of about 39 inches, an engine shaft weight of about 55 pounds, and an engine shaft diameter of about 3.0 inches. The ultimate compressive strength of reinforced concrete is 720,000 pounds per square foot (DIRS 101910-Poe 1998, p. 1-4). The assumed impact velocity would be 500 feet per second based on *DOE Standard, Accident Analysis for Aircraft Crash into Hazardous Facilities* (DIRS 101810-DOE 1996, p. C-7), which states that impact velocities would typically be less than 500 feet per second. Using the given values for the parameters in Equation E-1 shows that the engine would produce greater penetration than the engine shaft. For a velocity of 500 feet per second, the F-15 or F-16 jet engine would penetrate about 33 inches of concrete, less than the 4-foot wall thickness of the waste handling buildings.

The analysis for this Repository SEIS recalculated the probability of an aircraft crash into waste being handled at the repository using the methods stated in Frequency Analysis of Aircraft Hazards for License Application (DIRS 180112-BSC 2007, all) and modifying the input to account for the three analysis changes described above. The result was an estimated aircraft crash frequency of 1.5 × 10-8 per year

(DIRS 185405-Ashley 2008, all), which is below the DOE-recommended threshold for consideration (DIRS 178579-DOE 2004, p. 29).

Because operations at Nellis Air Force Base include aircraft that carry live ordnance, the analysis considered the possibility of an aircraft crash with ordnance or of jettisoned ordnance striking a waste handling building. However, as the DOE aircraft crash analysis noted (DIRS 180112-BSC 2007, p. 72), carrying ordnance over the flight-restricted airspace around the repository would be prohibited. Therefore, DOE considers this hazard as negligible or nonexistent (DIRS 180112-BSC 2007, p. 72).

Consistent with the Yucca Mountain FEIS, DOE analyzed a scenario in which a jet aircraft would impact and penetrate a Canister Receipt and Closure Facility that contained the maximum inventory of vulnerable commercial spent nuclear fuel. Section E.7 discusses this scenario as a potential sabotage event.

E.2.1.2.2 *Seismic Phenomena*

In the Yucca Mountain FEIS, DOE evaluated a beyond-design-basis earthquake that it assumed would cause the waste handling building to collapse. The Department based the analysis on the selection of a seismic design basis that specified that structures, systems, and components important to safety (including the waste handling building) should be able to withstand the horizontal motion from an earthquake with a return frequency of once in 10,000 years (DIRS 103237-CRWMS M&O 1998, p. VII-1). DOE has performed additional evaluations of the seismic hazard for the repository and revised the seismic design requirements for the facilities. DOE has committed to seismic design criteria and standards that would minimize potential consequences of seismic events. The Department intends to demonstrate capability for the major structures against earthquake ground motions that are considerably larger than the design-basis ground motion (DIRS 181572-DOE 2007, p. 3-9). Therefore, for this Repository SEIS, DOE did not evaluate the consequences of a waste handling building collapse due to a seismic event. In any event, the collapse of a waste handling building would be unlikely to produce consequences as great as the waste handling building collapse that DOE evaluated in the Yucca Mountain FEIS. This is because, unlike the bare fuel assemblies stored in air in the waste handling building that DOE assumed for the FEIS, most of the spent nuclear fuel in the waste handling buildings would be in casks or canisters, or stored underwater, and would not be vulnerable to extensive damage from building collapse. However, DOE has identified (DIRS 183621-BSC 2008, Sections 6.7 and 6.8) five Category 2 seismic events that could occur that could result in a release of radioactive material. In some cases, these accidents would be similar to or would bound other accidents that involved the same waste form from internal (nonseismic) initiators. Table E-3 lists the five accidents.

E.2.2 NONRADIOLOGICAL ACCIDENT SCENARIOS

The potential for a significant release of chemicals or toxic materials during postulated off-normal events at the proposed repository would be very unlikely because the repository would not accept hazardous waste as defined by the *Resource Conservation and Recovery Act of 1976* (42 U.S.C. 6901 et seq.) and 40 CFR Part 261, “Protection of Environment: Identification and Listing of Hazardous Waste.”

Hazardous and toxic substances would be present in limited quantities at the repository as part of operational requirements. Such substances would include liquid chemicals such as sulfuric acid,

Table E-3. Seismic-initiated Category 2 accidents.

Number	Accident	Location	Waste form	Mean number of occurrences over preclosure period (annual frequency)
1.	LLWF collapse and failure of HEPA filters and ductwork in other facilities	LLWF, other facilities	Low-level waste, HEPA filters, ductwork residue	8×10^{-3} (2×10^{-4})
2.	Seismic failure of the Canister Transfer Machine breaching a HLW canister during processing to a waste package	CRCF	5 HLW canisters	1×10^{-4} (2×10^{-6})
3.	Breach of uncanistered CSNF in an unsealed truck transportation cask in pool	WHF	4 PWR CSNF assemblies in pool	2×10^{-4} (4×10^{-6})
4.	Breach of CSNF in unsealed DPC in pool	WHF	36 PWR CSNF assemblies in pool	2×10^{-4}(4×10^{-6})
5.	Breach of CSNF in unsealed TAD canister in pool	WHF	21 PWR CSNF assemblies in pool	2×10^{-4} (4×10^{-6})

Source: DIRS 183621- BSC 2008, Tables 6.6-8 and 6.7-6.
BWR = Boiling-water reactor.
CSNF = Commercial spent nuclear fuel.
CRCF = Canister Receipt and Closure Facility.
DPC = Dual-purpose canister.
HEPA = High-efficiency particulate air (filter).
HLW = High-level radioactive waste.
LLWF = Low-Level Waste Facility.
PWR = Pressurized-water reactor.
TAD = Transportation, aging, and disposal (canister).
WHF = Wet Handling Facility.

hydrocarbons (including fuels, oils, and lubricants), and solid chemicals. These substances are in common use at other DOE sites. DOE evaluated the potential for impacts to workers from the handling of hazardous and toxic materials as part of the industrial health and safety analysis in Chapter 4, Section 4.1.7.1 of this Repository SEIS. That analysis estimated the impacts to workers from industrial hazards using DOE accident experience at other sites, which include impacts from hazardous materials and toxic substances as part of typical DOE operations.

Impacts to members of the public would be unlikely. Because the hazardous materials would be mostly liquid and solid rather than gaseous, a release would not transport the materials off the repository site. The potential for hazardous chemicals to reach surface water would be limited to spills or leaks that occurred just before a rare precipitation or snowmelt event large enough to generate runoff. DOE would use engineered measures to minimize the potential for spills or releases of hazardous chemicals throughout the project. Therefore, solid and liquid hazardous waste at the site would present a very small potential for accidental releases and exposures of workers or the public.

E.3 Source Terms for Repository Accident Scenarios

DOE estimated source terms for each accident scenario the analysis retained (Table E-1). The source term is an estimate of the amount of radioactive material an accident could release, which partially determines the estimated radiological impacts from accident scenarios. The source term includes several factors: the materials at risk (the total inventory of radioactive materials the scenario could involve) and the quantity of the release of those materials, the elevation of the release, the chemical and physical forms of the released materials, and the energy (if any) of the plume that would carry the radionuclides to the environment. These factors would vary according to the state of the material at the time and the extent and type of damage that would initiate the release. In addition, the analysis of the source terms considered measures that would reduce the amount of the release to the environment, such as filtration systems and local deposition of radionuclides.

For accident releases that passed through high-efficiency particulate air filters, DOE assumed a leak path factor of 1×10^{-4} for a two-stage filter system for particulates (DIRS 185225-BSC 2008, p. 76). The two-stage filter systems in the Initial Handling Facility, Wet Handling Facility, Canister Receipt and Closure Facilities, and Receipt Facility could reduce airborne particulates by a factor of 10,000.

E.3.1 NAVAL SPENT NUCLEAR FUEL

The Draft Repository SEIS determined that a drop and breach of a naval canister would be a Category 2 event sequence. However, as noted in Section E.2.1.1, DOE has now determined that this accident would be a Beyond-Category-2 event. Furthermore, DOE determined that the consequences of a breach of a naval canister, as evaluated in this Repository SEIS (Section E.4.2), would not be very large. Therefore, DOE did not consider it further.

E.3.2 HIGH-LEVEL RADIOACTIVE WASTE

High-level radioactive waste in vitrified form would arrive at the repository in sealed canisters inside transportation casks from the Savannah River Site, the Hanford Site, the West Valley Demonstration Project, and the Idaho National Laboratory. The analysis used Savannah River Site high-level radioactive waste to represent the materials at risk because it would produce the highest dose consequences (DIRS 185225-BSC 2008, p. 92, Table 27). Table E-4 lists the materials at risk per canister.

The analysis established the airborne release fraction of the materials at risk to calculate doses to workers and members of the public based on the method described in *Release Fractions for Spent Nuclear Fuel and High-Level Waste* (DIRS 180307-BSC 2007, Section 4.3.4). The high-level radioactive waste release fraction would consist of pulverized particles from an impact and breach of a high-level radioactive waste canister. The release fraction *PULF* is a function of the drop height of the high-level radioactive waste canister:

$$PULF = 2.0 \times 10^{-4} \text{ cubic centimeters per joule} \times E/V \qquad \text{(Equation E-2)}$$

where

PULF = fraction of crud release pulverized to respirable size (less than 10 micrometers in diameter) from a drop scenario

E/*V* = impact energy density in high-level radioactive waste

= 1×10^{-7} joule-square second per gram-square centimeter × $p \times g \times h$

where

p = density of the high-level radioactive waste, 2.75 gram per cubic centimeter

g = gravitational constant, 980.7 centimeters per square second

h = drop height in centimeters.

Table E-4. Inventory for Savannah River Site high-level radioactive waste (curies per canister).

Radionuclide	Inventory per canister	Radionuclide	Inventory per canister
Antimony-125	9.2	Plutonium-238	9.1×10^{2}
Americium-241	3.4×10^{2}	Plutonium-239	1.7×10^{1}
Americium-242m	7.4×10^{-2}	Plutonium-240	8.8
Americium-243	1.4	Plutonium-241	5.2×10^{2}
Barium-137m	4.2×10^{4}	Plutonium-242	2.1×10^{-2}
Cesium-134	6.5	Radon-226	4.6×10^{-8}
Cesium-135	2.2×10^{-1}	Radon-228	9.9×10^{-4}
Cesium-137	4.4×10^{4}	Promethium-147	1.5×10^{2}
Cobalt-60	4.9×10^{1}	Ruthenium-106	4.4×10^{-3}
Curium-242	6.1×10^{-2}	Samarium-147	5.1×10^{-8}
Curium-243	3.3×10^{-1}	Samarium-151	1.5×10^{2}
Curium-244	3.0×10^{2}	Selenium-79	5.3×10^{-1}
Curium-245	2.4×10^{-2}	Strontium-90	2.7×10^{4}
Curium-246	2.9×10^{-2}	Technetium-99	9.2
Curium-247	2.2×10^{-2}	Thorium-229	1.4×10^{-4}
Europium-154	1.9×10^{2}	Thorium-230	1.4×10^{-5}
Europium-155	1.5×10^{-1}	Thorium-232	1.4×10^{-3}
Iodine-129	3.2×10^{-4}	Tin-126	7.8×10^{-1}
Lead-210	6.0×10^{-9}	Uranium-232	2.7×10^{-4}
Neptunium-237	3.0×10^{-2}	Uranium-233	5.6×10^{-2}
Nickel-59	8.4×10^{-1}	Uranium-234	7.2×10^{-2}
Nickel-63	7.5×10^{1}	Uranium-235	6.6×10^{-4}
Niobium-93m	2.3×10^{-1}	Uranium-236	3.7×10^{-3}
Protactinium-231	1.4×10^{-7}	Uranium-238	4.7×10^{-2}
Palladium-107	1.3×10^{-3}	Yttrium-90	2.7×10^{4}
		Zirconium-93	3.9×10^{-1}

Source: DIRS 185225-BSC 2008, p. 92, Table 2.

For the high-level radioactive waste drop (Scenario 3 from Table E-1), the drop height would be 1,200 centimeters (40 feet) (DIRS 180307-BSC 2007, Section 6.7). Using a drop height of 1,200 centimeters results in a respirable fraction of

$$PULF = (2 \times 10^{-4}) \times (1.0 \times 10^{-7}) \times 2.75 \times 980.7 \times 1{,}200 = 6.5 \times 10^{-5} \qquad \text{(Equation E-3)}$$

DOE rounded the value in Equation E-3 up to 7.0×10^{-5}.

For the three accident scenarios that would involve high-level radioactive waste (Scenarios 2 and 3, Table E-1), the analysis applied a leak path factor (DIRS 185225-BSC 2008, p. 76, Section 6.1.4.2). This factor would account for deposition of particles in the leakage path out of the canisters or cask. For Scenario 2, the analysis applied a leak path factor of 0.01 to account for the leak path out of the high-level radioactive waste canister (0.1) and then out of the transportation cask (0.1). For Scenarios 3 (Table E-1) and Scenario 2 (Table E-3), the analysis used a leak path factor of 0.1 to account for the canister leak path. Therefore, for particulate releases, the respirable airborne release fractions for scenarios that involved high-level radioactive waste would be:

$$\text{Scenario 2 and 3, Table E-1} = 5 \text{ canisters} \times 0.01 \times 7 \times 10^{-5} = 3.5 \times 10^{-6}$$

$$\text{Scenario 4, Table E-1} = 2 \text{ canisters} \times 0.1 \times 7 \times 10^{-5} \quad = 1.4 \times 10^{-5}$$

The analysis applied these values to the materials at risk radionuclide values from Table E-4 and used the results to calculate the consequences from the high-level radioactive waste drop scenario.

E.3.3 COMMERCIAL SPENT NUCLEAR FUEL

Scenarios 5 to 12 and 14 (Table E-1) would involve releases from commercial spent nuclear fuel assemblies when the assemblies were damaged during an accident. The releases would consist of fuel particles, radioactive gas, and crud. For the analysis in this Repository SEIS, DOE chose to use the maximum fuel characteristics. This choice helps ensure that the calculated consequences would encompass those of commercial spent nuclear fuel received at the repository and that the results would be conservative and not underestimated. Table E-5 lists maximum fuel characteristics.

Table E-5. Maximum commercial boiling- and pressurized-water-reactor spent nuclear fuel characteristics.

Commercial SNF assembly	Initial enrichment (%)	Burnup (GWd/MTU)	Decay time (years)
Maximum PWR	5.0	80	5
Maximum BWR	5.0	75	5

Source: DIRS 185225-BSC 2008, Table 6, p. 67.
BWR = Boiling-water reactor.
GWd = Gigawatt-day.
MTU = Metric ton of uranium.
SNF = Spent nuclear fuel.
PWR = Pressurized-water reactor.

Previous analyses determined that the consequences of accidents that involved pressurized-water reactor fuel assemblies (DIRS 155970-DOE 2002, Appendix H, p. H-35) would be higher than those that involved boiling-water reactor assemblies. For the maximum fuel, the preclosure consequence analysis (DIRS 185225-BSC 2008, Section 6.6) validates this conclusion.

E.3.3.1 Fuel Release

As noted in the Yucca Mountain FEIS (DIRS 155970-DOE 2002, Appendix H, p. H-24), commercial spent nuclear fuel contains nearly 400 radionuclides. Not all of these, however, would be important in terms of a potential to cause adverse health effects, and many would have decayed to minor quantities by the time the material arrived at the repository. For this Repository SEIS, DOE performed an assessment and identified 50 radionuclides as part of the inventory that would contribute to offsite consequences from a release (DIRS 180185-BSC 2007, Attachment II). Table E-6 lists the inventory for the consequences analysis for pressurized-water reactor fuel based on the maximum fuel characteristics in Table E-5. DOE selected maximum fuel characteristics to bound the impacts from accidents involving commercial spent nuclear fuel. Section E.4.3 describes the effect of using maximum fuel rather than representative fuel (described in Section E.7).

To calculate the consequences from a commercial spent nuclear fuel drop accident scenario, it is necessary to derive an airborne respirable release fraction to apply to the inventory. For accidents in air, the release fractions would have two components—burst release fraction and oxidation release fraction. The burst release fraction would be that fraction that was released immediately when the commercial spent nuclear fuel rod ruptured as a result of the drop. This fraction would consist of the releasable material in the fuel pin gap plus additional particles that were produced by fragmentation of the fuel

Table E-6. Inventory for commercial pressurized-water-reactor spent nuclear fuel having maximum characteristics.

Radionuclide	Inventory	Radionuclide	Inventory
Americium-241	8.8×10^{2}	Niobium-93m	3.9×10^{-1}
Americium-242	1.0×10^{1}	Niobium-94	1.0×10^{-4}
Americium-242m	1.0×10^{1}	Palladium-107	1.6×10^{-1}
Americium-243	6.0×10^{1}	Plutonium-238	6.8×10^{3}
Antimony-125	1.9×10^{3}	Plutonium-239	1.8×10^{2}
Barium-137m	9.9×10^{4}	Plutonium-240	4.0×10^{2}
Cadmium-113m	3.8×10^{1}	Plutonium-241	8.0×10^{4}
Carbon-14	5.4×10^{-1}	Plutonium-242	3.3
Cesium-134	4.1×10^{4}	Promethium-147	2.3×10^{4}
Cesium-135	6.3×10^{-1}	Protactinium-231	4.2×10^{-5}
Cesium-137	1.1×10^{5}	Ruthenium-106	1.3×10^{4}
Chlorine-36	1.1×10^{-2}	Samarium-151	3.2×10^{2}
Cobalt-60[a]	3.3×10^{1}	Selenium-79	7.4×10^{-2}
Curium-242	3.6×10^{1}	Strontium-90	6.5×10^{4}
Curium-243	4.2×10^{1}	Technetium-99	1.3×10^{1}
Curium-244	1.4×10^{4}	Thorium-230	3.3×10^{-5}
Curium-245	1.8	Tin-126	6.8×10^{-1}
Curium-246	1.2	Uranium-232	6.0×10^{-2}
Europium-154	6.2×10^{3}	Uranium-233	2.4×10^{-5}
Europium-155	1.8×10^{3}	Uranium-234	5.2×10^{-1}
Hydrogen-3	5.0×10^{2}	Uranium-235	3.3×10^{-3}
Iodine-129	3.6×10^{-2}	Uranium-236	2.2×10^{-1}
Iron-55[a]	7.5×10^{2}	Uranium-238	1.4×10^{-1}
Krypton-85	5.8×10^{3}	Yttrium-90	6.5×10^{4}
Neptunium-237	4.0×10^{-2}	Zirconium-93	1.3

a. Buildup of activated components (crud) contained on fuel assembly surfaces.

Source: DIRS 185225-BSC 2008, p. 67, Table 7.

pellets from the mechanical impact of the drop. The oxidation release fraction would occur when the hot fuel pellets were exposed to air and became oxidized, producing a powder (DIRS 180307-BSC 2007, Section 4.3.5). This release fraction would be produced over a longer period (up to 30 days). Table E-7 lists the release fractions for these components (DIRS 185225-BSC 2008, p. 74, Table 10). Some releases could involve locations where high-efficiency particulate air filtration of the material would be available before release to the atmosphere. The table indicates the airborne release fraction for cases with and without high-efficiency particulate air filtration.

The analysis applied the release fractions from Table E-7 to the radionuclide inventories in Table E-6 to calculate the respirable airborne release fractions for those accident scenarios that involved commercial spent nuclear fuel in an air environment (Scenarios 5, 7, 9, and 14 in Table E-1).

For accident scenarios that would occur in the pool of the Wet Handling Facility (Scenarios 6, 8, 10, 11 and 12 in Table E-1), the analysis assumed release of only gaseous radionuclides because the water above the commercial spent nuclear fuel assemblies would trap the particulates (DIRS 179965-BSC 2007, Section 7.4), which would not be available for release. Consistent with the preclosure consequence analysis (DIRS 185225-BSC 2008, p. 77), the analysis for this Repository SEIS assumed release fractions

Table E-7. Release fractions for commercial spent nuclear fuel drop accident scenarios.

Radionuclide	Burst release: RARF without HEPA[a]	Burst release: RARF with HEPA[b]	Oxidation release-RARF with HEPA[b]	Accident scenarios (Table E-1)[c]
Hydrogen-3	3.0×10^{-1}	3.0×10^{-1}	7.0×10^{-1}	4 to 10
Krypton-85	3.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	4 to 10
Iodine-129	3.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	4 to 10
Cesium	2.0×10^{-3}	2.0×10^{-7}	2.0×10^{-7}	4, 6
Strontium[e]	3.0×10^{-5}	3.0×10^{-9}	2.0×10^{-7}	4, 6
Ruthenium	2.0×10^{-3}	2.0×10^{-7}	2.0×10^{-7}	4, 6
Crud[d]	1.5×10^{-2}	1.5×10^{-6}	0	4, 6
Fuel fines[e]	3.0×10^{-5}	3.0×10^{-9}	2.0×10^{-7}	4, 6

a. Source: DIRS 185225-BSC 2008, p. 74, Table 10.
b. Factor of 1.0×10^{-4} applied per DIRS 185225-BSC 2008, p. 76, Section 6.1.4.1.
c. These scenarios would occur where HEPA filtration was operating.
d. See Section E.3.3.2 for crud component.
HEPA = High-efficiency particulate air (filter).
RARF = Respirable airborne release fraction.

For the breach of a sealed truck transportation cask due to a fire (accident scenario 14 from Table E-1), the source term would include the radioactive crud components (cobalt-60 and iron-55) (Section E.3.2.2) from four pressurized-water-reactor assemblies (release fraction of 0.015 from Table E-7). In addition, the analysis assumed for this event that 1 percent of the fuel rods would be damaged and would release radioactive components available for release in the fuel rods (DIRS 185225-BSC 2008, p. 149). The release fractions are from Table E-7 and the radionuclides involved are from Table E-6. Thus, the source term for the radionuclides in the fuel rods would be 1 percent of the fuel rods in four pressurized-water-reactor assemblies (a factor of 0.04 applied to the Table E-6 inventory).

E.3.3.2 Crud

During nuclear power reactor operation, crud (corrosion material) builds up on the outside of the fuel rod assembly surfaces and becomes radioactive from neutron activation. An accident could dislodge crud from those surfaces. After decaying for 5 years, the nuclide species that have significant activity in the crud for commercial spent nuclear fuel are iron-55 and cobalt-60. Table E-8 lists the crud activity per assembly after 5 years of decay (DIRS 185225-BSC 2008, Table 7). The analysis assumed that the fraction of crud release in a drop accident scenario would be 0.015 (DIRS 185225-BSC 2008, p. 74, Table 10), all of which would be respirable.

Table E-8. Pressurized-water-reactor commercial spent nuclear fuel crud activities (curies per assembly).

	Inventory			
	At 5 years		Respirable amount (5-year-old fuel)	
Radionuclide	PWR	BWR	PWR	BWR
Iron-55	7.5×10^2	3.5×10^2	11	5.3
Cobalt-60	33	1.1×10^2	0.49	1.6

PWR = Pressurized-water reactor.
BWR = Boiling-water reactor.

E.3.4 LOW-LEVEL RADIOACTIVE WASTE FIRE

Several operations at the proposed repository would produce low-level radioactive waste, which the Low-Level Waste Facility would receive for shipment off the site. The accident scenario the analysis identified for this facility (Scenario 13, Table E-1) would be a fire that involved combustion of the combustible portion of the low level waste stored at the Low-Level Waste Facility. Table E-9 lists the distribution of radionuclides released from the fire event as developed in *Preclosure Consequence Analyses* (DIRS 185225-BSC 2008, p. 96, Table 30).

Table E-9. Respirable airborne release for low-level radioactive waste fire.

Radionuclide	Respirable airborne release (curies)
Cesium-134	0.20
Cesium-137	0.22
Cobalt-58	0.20
Cobalt-60	0.50
Manganese-54	6.8×10^{-2}

Source: DIRS 185225-BSC 2008, p. 96, Table 30.

E.3.5 SEISMIC EVENT

As noted in Section E.2.1.2.2, DOE identified five Category 2 events initiated by a seismic event. The source terms for the seismic-initiated events are as follows:

- Event 1 (Event 1 from Table E-1 and event 1 from Table E-3). This event would involve releases from damage to high-efficiency particulate air filters, wet-solid resin, and Wet Handling Facility pool filters. The seismic initiator for this event would result in the bounding source term because it would affect several waste forms simultaneously and would cause Low-Level Waste Facility collapse and a release from the stored low-level waste (DIRS 185225-BSC 2008, p. 99). This event would involve failure of the high-efficiency particulate air filters and associated ducting and dampers as well as failure of the confinement function for the solid and liquid low-level radioactive waste. DOE based airborne release fractions for this event on values for free-fall spills (DIRS 185225-BSC 2008, p. 75, Section 6.1.3.3) from the DOE handbook on release fractions (DIRS 103756-DOE 1994, all). The Department used free-fall spill release fractions for the releases because the collapse of structures and components or falling debris onto materials would be equivalent to a crush or impact event or a free fall of the material onto an unyielding surface. The development of the release fractions considered multiple seismic release effects including shock vibration, structure collapse, and debris turbulence; details are in *Preclosure Consequence Analyses* (DIRS 185225-BSC 2008, p. 75, Section 6.1.3.3). The release fractions for estimating accumulation of particulate radionuclides on high-efficiency particulate air filters and associated ducting and dampers are 2.0×10^{-4} for cesium and ruthenium, 1.5×10^{-2} for the crud components (cobalt and iron), and 3.0×10^{-5} for all remaining particulate radionuclides. Because barium-137m would be in equilibrium with cesium-137 on the filters, the release for the seismic event is set equal to that of cesium-137. DOE determined that processing boiling-water reactor commercial spent nuclear fuel would produce a higher accumulation on the filters, resulting in a high dose from the accident (DIRS 185225-BSC 2008, Table 50, p. 139) Therefore, DOE based the estimate of the amount of accumulated radiological material available for release on the basis of (1) boiling-water reactor commercial spent nuclear fuel received at an average rate of 1,500 fuel assemblies per month (based on 3,600 metric tons per year with each boiling-water reactor fuel assembly equivalent to 0.2 metric ton), (2) 10 percent of these (150 per month) would be handled as uncanistered fuel assemblies in the Wet Handling Facility (and therefore would be available to release radionuclides during normal operations), and (3) 1 percent of the fuel pins would have damaged cladding (resulting in a release that accumulated on the Wet Handling Facility high-efficiency particulate air filters) (DIRS 185225-BSC 2008, p. 96). An airborne release fraction of 1.0×10^{-2} and a respirable fraction of 1.0 was applied to the accumulated inventory based on releases from unenclosed filter media during a seismic event sequence from DIRS 185225-BSC (2008, p. 75). The fuel assumed for this event would be the representative boiling-water reactor fuel assembly developed for normal operations releases. Assuming boiling-water-reactor fuel assemblies for this event is conservative because radionuclide releases to the high-efficiency particulate air filters from this fuel would produce a higher dose (DIRS 185225-BSC 2008, p. 139). Table E-10 lists the source term for this event. The table lists the radionuclide inventory for the representative fuel assembly in the second column. The fourth column lists the filter buildup rate, which was calculated by the product of the curies per spent fuel assembly in the second column multiplied by 150 fuel assemblies per month, the airborne release fraction in the third column, and a factor of 0.01 for the damaged fuel cladding fraction for all radionuclides except cobalt and iron, which are crud contributions released from the cladding of all the fuel assembly surfaces. The fifth column lists the buildup after 18 months, and the sixth column is the amount released from the filters (1 percent of the 18-month buildup quantity). In addition, the analysis assumed the seismic event would release radionuclides from the Low-Level Waste Facility from high-integrity containers, drums, boxes, and tanks containing liquid low-level radioactive waste. *Preclosure Consequence Analyses* (DIRS 185225-BSC 2008, Section 6.1.3.3) presents details of this release estimate. The Low-Level Waste Facility respirable airborne release includes five radionuclides; the seventh column in Table E-10 (DIRS 185225-BSC 2008, p. 102, Table 35) lists their activity. This activity is added to the corresponding high-efficiency particulate air filter release to provide the total respirable airborne release (last column).

- Event 2. This event involves drop and breach of five high-level radioactive waste canisters in an unsealed waste package from a seismic event. The source term would be the same as that developed in Section E.3.1 for high-level radioactive waste.

- Event 3. This event would involve breach of four uncanistered pressurized-water-reactor spent nuclear fuel assemblies in an unsealed transportation cask from a seismic event. The source term for this event would be the same as that for Event 5 from Table E-1, which is developed in Section E.3.3.

- Event 4. This event would involve breach of 36 pressurized-water reactor spent nuclear fuel assemblies in an unsealed dual-purpose canister in the Wet Handling Facility pool from a seismic event. The source term for this event would be the same as that for Event 3 above, except the number of fuel assemblies increased from 4 to 36.

Table E-10. Source term (curies) for bounding seismic event.

Radionuclide	Representative BWR (curies/SFA)	Fuel ARF	HEPA filter buildup rate (curies/ month)	HEPA filter buildup-18 months	HEPA filter seismic release (curies)	LLW seismic release (curies)	Total seismic release (curies)
Americium-241	3.7×10^{2}	3.0×10^{-5}	1.7×10^{-2}	3.0×10^{-1}	3.0×10^{-3}	0	3.0×10^{-3}
Americium-242	2.9	3.0×10^{-5}	1.3×10^{-4}	2.3×10^{-3}	2.3×10^{-5}	0	2.3×10^{-5}
Americium-242m	2.9	3.0×10^{-5}	1.3×10^{-4}	2.3×10^{-3}	2.3×10^{-5}	0	2.5×10^{-5}
Americium-243	8.6	3.0×10^{-5}	3.9×10^{-4}	7.0×10^{-3}	7.0×10^{-5}	0	7.0×10^{-5}
Antimony-125	1.2×10^{2}	3.0×10^{-5}	5.4×10^{-3}	9.7×10^{-2}	9.7×10^{-4}	0	9.7×10^{-5}
Barium-137m	2.3×10^{4}	2.0×10^{-4}	6.9	18	1.2×10^{2}	0	1.2×10^{2}
Cadmium-113m	5.2	3.0×10^{-5}	2.3×10^{-4}	4.2×10^{-3}	4.2×10^{-5}	0	4.2×10^{-5}
Carbon-14	0.21	0.3	0	0	0	0	0
Cerium-144	17	3.0×10^{-5}	7.7×10^{-4}	1.4×10^{-2}	1.4×10^{-4}	0	1.4×10^{-4}
Cesium-134	1.3×10^{3}	2.0×10^{-4}	0.39	7.0	7.0×10^{-2}	1.1	8.1
Cesium-135	0.18	2.0×10^{-4}	5.4×10^{-5}	9.7×10^{-4}	9.7×10^{-6}	0	9.7×10^{-6}
Cesium-137	2.4×10^{4}	2.0×10^{-4}	7.2	1.3×10^{2}	1.3	1.2	2.5
Chlorine-36	3.5×10^{-3}	0.3	0	0	0	0	0
Cobalt-58	0	0	0	0	0	1.1	1.1
Cobalt-60	57	1.5×10^{-2}	1.3	2.3×10^{3}	23	2.7	26
Curium-242	2.4	3.0×10^{-5}	1.1×10^{-4}	1.9×10^{-3}	1.9×10^{-5}	0	1.9×10^{-5}
Curium-243	5.5	3.0×10^{-5}	2.5×10^{-4}	4.5×10^{-3}	4.5×10^{-5}	0	4.5×10^{-5}
Curium-244	9.2×10^{2}	3.0×10^{-5}	4.1×10^{-2}	0.75	7.5×10^{-3}	0	7.5×10^{-3}
Curium-245	9.1×10^{-2}	3.0×10^{-5}	4.1×10^{-6}	7.4×10^{-5}	7.4×10^{-7}	0	7.4×10^{-7}
Curium-246	4.3×10^{-2}	3.0×10^{-5}	1.9×10^{-6}	3.5×10^{-5}	3.5×10^{-7}	0	3.5×10^{-7}
Europium-154	7.7×10^{2}	3.0×10^{-5}	3.5×10^{-2}	0.62	6.2×10^{-3}	0	6.2×10^{-3}
Hydrogen-3	1.1×10^{2}	0.3	0	0	0	0	0
Iodine-129	9.2×10^{-3}	0.3	0	0	0	0	0
Iron-55	9.8×10^{1}	1.5×10^{-2}	2.2×10^{2}	4.0×10^{3}	0.40	0	0.40
Krypton-85	1.2×10^{3}	0.3	0	0	0	0	0
Manganese-54	0	0	0	0	0	0.37	0.37
Neptunium-237	8.7×10^{-2}	3.0×10^{-5}	3.9×10^{-6}	7.0×10^{-5}	7.0×10^{-7}	0	7.0×10^{-7}
Neptunium-239	8.6	3.0×10^{-5}	3.9×10^{-4}	7.0×10^{-3}	7.0×10^{-5}	0	7.0×10^{-5}
Niobium-93m	0.16	3.0×10^{-5}	7.2×10^{-6}	1.3×10^{-4}	1.3×10^{-6}	0	1.3×10^{-6}
Europium-155	1.9×10^{2}	3.0×10^{-5}	8.6×10^{-3}	0.15	1.5×10^{-3}	0	1.5×10^{-3}
Niobium-94	2.6×10^{-5}	3.0×10^{-5}	1.2×10^{-9}	2.2×10^{-8}	2.2×10^{-10}	0	2.2×10^{-10}
Palladium-107	3.5×10^{-2}	3.0×10^{-5}	1.6×10^{-6}	2.8×10^{-5}	2.8×10^{-7}	0	2.8×10^{-7}
Plutonium-238	1.0×10^{3}	3.0×10^{-5}	4.5×10^{-2}	0.81	8.1×10^{-3}	0	8.1×10^{-3}
Plutonium-239	54	3.0×10^{-5}	2.4×10^{-3}	4.4×10^{-2}	4.4×10^{-4}	0	4.4×10^{-4}
Plutonium-240	1.3×10^{2}	3.0×10^{-5}	5.9×10^{-3}	0.11	1.1×10^{-3}	0	1.1×10^{-3}
Plutonium-241	1.6×10^{4}	3.0×10^{-5}	0.72	13	0.13	0	0.13
Plutonium-242	0.71	3.0×10^{-5}	3.2×10^{-5}	5.8×10^{-4}	5.8×10^{-6}	0	5.8×10^{-6}
Praseodymium-144	17	3.0×10^{-5}	7.7×10^{-4}	1.4×10^{-2}	1.4×10^{-4}	0	1.4×10^{-4}
Promethium-147	2.1×10^{3}	3.0×10^{-5}	9.5×10^{-2}	1.7	1.7×10^{-2}	0	1.7×10^{-2}
Protactinium-231	1.9×10^{-5}	3.0×10^{-5}	8.6×10^{-10}	1.5×10^{-8}	1.5×10^{-10}	0	1.5×10^{-10}
Ruthenium-106	91	2.0×10^{-4}	2.7×10^{-2}	0.49	4.9×10^{-3}	0	4.9×10^{-3}
Samarium-151	67	3.0×10^{-5}	3.0×10^{-3}	5.4×10^{-2}	5.4×10^{-4}	0	5.4×10^{-4}
Selenium-79	2.0×10^{-2}	3.0×10^{-5}	9.0×10^{-7}	1.6×10^{-5}	1.6×10^{-7}	0	1.6×10^{-7}
Strontium-90	1.7×10^{4}	3.0×10^{-5}	0.77	14	0.14	0	0.14
Technetium-99	3.9	3.0×10^{-5}	1.8×10^{-4}	3.2×10^{-3}	3.2×10^{-5}	0	3.2×10^{-5}
Thorium-230	3.1×10^{-5}	3.0×10^{-5}	1.4×10^{-9}	2.5×10^{-8}	2.5×10^{-10}	0	2.5×10^{-10}
Tin-126	0.16	3.0×10^{-5}	7.2×10^{-6}	1.3×10^{-4}	1.3×10^{-6}	0	1.3×10^{-6}
Uranium-232	8.7×10^{-3}	3.0×10^{-5}	3.9×10^{-7}	7.0×10^{-6}	7.0×10^{-8}	0	7.0×10^{-8}
Uranium-233	0	3.0×10^{-5}	0	0	0	0	0

Table E-10. Source term (curies) for bounding seismic event (continued.

Radionuclide	Representative BWR (curies/SFA)	Fuel ARF	HEPA filter buildup rate (curies/ month)	HEPA filter buildup-18 months	HEPA filter seismic release (curies)	LLW seismic release (curies)	Total seismic release (curies)
Uranium-234	0.24	3.0×10^{-5}	1.1×10^{-5}	1.9×10^{-4}	1.9×10^{-6}	0	1.9×10^{-6}
Uranium-235	2.1×10^{-3}	3.0×10^{-5}	9.5×10^{-7}	1.7×10^{-6}	1.7×10^{-8}	0	1.7×10^{-8}
Uranium-236	7.5×10^{-2}	3.0×10^{-5}	3.4×10^{-6}	6.1×10^{-5}	6.1×10^{-7}	0	6.1×10^{-7}
Uranium-238	6.2×10^{-2}	3.0×10^{-5}	2.8×10^{-6}	5.0×10^{-5}	5.0×10^{-7}	0	5.0×10^{-7}
Yttrium-90	1.7×10^{4}	3.0×10^{-5}	0.77	14	0.14	0	0.14
Zirconium-93	0.35	3.0×10^{-5}	1.6×10^{-5}	2.8×10^{-4}	2.8×10^{-6}	0	2.8×10^{-6}

ARF = Airborne release fraction.
BWR = Boiling-water reactor.
HEPA = High-efficiency particulate air (filter).
LLW = Low-level radioactive waste.
SFA = Spent (nuclear) fuel assembly.

E.4 Accident Scenario Consequences

E.4.1 GENERAL METHODOLOGY

The analysis calculated radiological accident scenario consequences as individual doses (rem), collective doses (person-rem), and latent cancer fatalities. It considered the following individuals: (1) the maximally exposed offsite individual, who is a hypothetical member of the public at the point on the analyzed land withdrawal area boundary who would receive the largest dose from the assumed accident scenario, which is either about 18.5 kilometers (11 miles) southeast of the repository site or 7.8 kilometers (4.8 miles) east of the site, (2) the noninvolved worker, or the hypothetical worker near the accident, who would be 60 meters (200 feet) from the release point, and (3) members of the public who resided within about 84 kilometers (52 miles) of the proposed repository in 2067 (Chapter 3, Figure 3-16). The 60-meter distance for the noninvolved worker would be less than the 100 meters (330 feet) DOE used in the Yucca Mountain FEIS because the repository design would place exclusion fences 60 meters from the facilities. This analysis did not calculate doses to involved workers for the following reasons: (1) for releases in waste handling buildings (Scenarios 2 through 12, Table E-1), operators would be in enclosed operating areas that would isolate them from a release; (2) for Scenarios 13 and 14 from Table E-1 (fires involving low-level radioactive waste and a truck with a transportation cask), the fire would loft the release into the atmosphere such that workers close to the release would not receive meaningful exposure; and (3) for Scenario 1 from Table E-1 (seismic event), workers inside the Low-Level Waste Handling Facility would probably be injured or killed as a result of the event, and the dose to the noninvolved worker at 60 meters (200 feet) would be representative of the dose to involved workers outside the facility. Appendix D, Section D.1 discusses the health effects of radiation doses.

The analysis used the GENII computer program (DIRS 179907-Napier 2007, all) and the radionuclide source terms for the identified accident scenarios to calculate consequences to individuals and populations. The GENII program, developed by the U.S. Environmental Protection Agency (EPA) at Pacific Northwest National Laboratory, has been widely used to compute radiological impacts from accident scenarios that involve releases of radionuclides. The analysis used this program to calculate doses for offsite members of the public, the maximally exposed offsite individual, and the noninvolved worker. The GENII program calculates radiological doses based on input meteorological conditions. The analysis used 95th-percentile and 50th-percentile Yucca Mountain sector-specific weather conditions for 2001 to 2005; 16 radial sectors represented areas affected by wind direction from the repository. DOE used the methodology in *General Public Atmospheric Dispersion Factors* (DIRS 177510-BSC 2007, all) to calculate atmospheric dispersion factors (dilution of the plume as a function of weather and distance from the release point) for site boundary doses and collective population doses.

The GENII program evaluates doses from various pathways including direct radiation from the radioactive plume produced by the accident, inhalation of radioactive material in the plume, direct exposure from radionuclides deposited on soil (groundshine), ingestion of food products that become contaminated with radionuclides from the plume, and exposure from radionuclides that are resuspended from the ground. The dose calculations included all these pathways for the southeast site boundary and 84-kilometer (52-mile) population doses. For the noninvolved worker, the analysis assumed the worker would be exposed for 8.5 hours (DIRS 185225-BSC 2008, p. 82), so only direct exposure, inhalation from the plume, and groundshine for 8.5 hours were factors. *Preclosure Consequence Analysis* (DIRS 185225-BSC 2008, Section 6.4) provides details on the input data for the analysis. For the maximum site boundary dose, calculations included a hypothetical individual 18.5 kilometers (11 miles) southeast and 7.8 kilometers (4.8 miles) east of the repository. These two locations would produce the highest site boundary dose based on sector-specific meteorology. For the individual assumed to be at the southeast boundary, the analysis evaluated an exposure period of 8.5 hours per day to account for the fact that this individual would be a worker on the Nevada Test Site (no members of the public reside at this location). For this individual, the analysis did not consider ingestion doses because no crops grow at this location.

For facilities with high-efficiency particulate air filtration systems, the analysis in this Repository SEIS credits the filtration that would be provided during an accident. In some cases (Initial Handling Facility), the results provide the consequences of the same accident without credit for filtration. These results indicate that some filtration systems may not have to meet regulatory standards; however, because they are in the facility design, DOE has included their availability in the assessment of accident consequences.

For exposure to inhaled and ingested radioactive material, the analysis assumed (in accordance with EPA guidance) that doses would accumulate in the body for a total of 50 years after the accident (DIRS 101069-Eckerman et al. 1988, p. 7). For external exposures (from ground contamination and contaminated food consumption), the analysis assumed an exposure period of 30 days (DIRS 182588-NRC 2007, p. 4). It also assumed that the accident occurred during the fall of the year, so the 30-day exposure period included harvesting and consumption of contaminated food crops.

The analysis used the projected population around the repository in 2067 (Chapter 3, Figure 3-15). The exposed population would be individuals living within about 84 kilometers (52 miles) of the repository, including pockets of people who would reside just beyond the 84-kilometer distance. DOE selected the south-southeast sector to compute population doses because this sector would contain the highest population out to 84 kilometers (Chapter 3, Figure 3-16) and the predominant wind direction is very near to this direction (Chapter 3, Figure 3-3). The dose calculation used the specific dispersion factor (dilution of the plume with distance) for this sector (DIRS 177510-BSC 2007, all). The population dose calculations included impacts from the consumption of food that radionuclide releases contaminated. The contaminated food consumption analysis used site-specific data on food production and consumption for the region around the Yucca Mountain site (DIRS 177751-BSC 2007, Section 8.4).

DOE has not evaluated in detail the potential cleanup costs for the accident scenarios, but the Yucca Mountain FEIS did consider cleanup costs for transportation accidents that involved material en route to the repository (DIRS 155970-DOE 2002, Appendix J, Section J.1.4.2.5). Such costs are highly uncertain and would depend on the types of soils and remediation actions and the extent of cleanup, which would be based on the requirements at the time of the accident. As noted in the FEIS, the costs could range from about $1 million to $10 billion for severe, maximum reasonably foreseeable transportation accidents. For the repository accident scenarios, costs should to be below the lower end of this range because the releases would be very small and the land near the repository would be federally controlled, undeveloped, and uninhabited. In any event, liability for and recovery of costs of such accidents would be covered under provisions of the *Price-Anderson Act* (Section 170 of the *Atomic Energy Act*, as amended; 42 U.S.C. 2011 et seq.), which currently provides for costs as high as $10.26 billion, as described in Appendix H of this Repository SEIS.

E.4.2 ACCIDENT SCENARIO CONSEQUENCE RESULTS

To calculate the potential consequences for the accident scenarios (Tables E-1 and E-3), the analysis did not take credit for mitigation measures (evacuation and interdiction of contaminated foods). This assumption ensured that the estimated consequences would be conservative. Section E.4.3 discusses the effect of this assumption. Tables E-11 and E-12 list the results of the consequence calculations (DIRS 185403-Schulz 2008, all). Table E-11 provides consequence results for unfavorable (95th-percentile) weather conditions. Unfavorable weather conditions (those that could result in a high dose) would occur no more than 5 percent of the time. Table E-12 provides consequence results for annual average weather (50th-percentile). These conditions would result in average doses. The tables list doses in millirem for individuals and in person-rem (collective dose to all exposed persons) for the 84-kilometer (52-mile) population around the site.

For selected individuals and populations, the tables list estimated probability and number of latent cancer fatalities for the maximally exposed offsite individual, the public, and noninvolved workers over the lifetimes of the exposed individuals as a result of the calculated doses using the conversion factors in Section E.4.1. These estimates do not consider the accident frequency. The accident scenario with the highest population impact for the unfavorable weather conditions (seismic event involving failure of high-efficiency particulate air filtration system and low-level radioactive waste confinement) would result in an estimated 0.19 latent cancer fatality for this same population.

In addition, Table E-11 lists radiological dose information for accidents in the Initial Handling Facility that do not credit the filtration system. As indicated above, DOE has provided these results only to illustrate that these filtration systems might not be necessary to meet regulatory standards; however, because they are an integral part of the facility design, this Repository SEIS credits the filters in the analysis of impacts. The estimated annual frequencies of these events are consistent with the availability of the filters.

E.4.3 EFFECT OF CONSERVATIVE ASSUMPTIONS

As noted above, DOE made several conservative assumptions in the accident analyses for this Final Repository SEIS. These assumptions account for uncertainties and help ensure that impacts would not be underestimated. This section evaluates the effect of two of the more significant assumptions.

Table E-11. Estimated radiological consequences of repository operations accident scenarios for unfavorable (95th-percentile) sector-specific meteorological conditions.

	Expected occurrences over the preclosure period (annual frequency)[a]		Maximally exposed offsite individual[b]		Population		Noninvolved worker	
Accident scenario	Internal events	Seismic events	Dose (rem)	LCF_i[c]	Dose (person-rem)	LCF_p[c]	Dose (rem)	LCF_i[c]
1. Seismic event resulting in LLWF collapse and failure of HEPA filters and ductwork in other facilities	(not applicable)	8×10^{-3} (2×10^{-4})	3.5×10^{-2}	2.1×10^{-5}	3.1×10^{2}	1.9×10^{-1}	3.5×10^{0}	2.1×10^{-3}
2. Breach of sealed HLW canisters in a sealed transportation cask	$< 1 \times 10^{-4}$ $(< 2 \times 10^{-6})$	$< 1 \times 10^{-4}$ $(< 2 \times 10^{-6})$	2.6×10^{-5} (2.6×10^{-3})[d]	1.6×10^{-8}	2.1×10^{-1} (2.1×10^{1})[d]	1.3×10^{-4}	3.5×10^{-3} (3.5×10^{-1})[d]	2.1×10^{-6}
3. Breach of sealed HLW canister in an unsealed waste package	$< 1 \times 10^{-4}$ $(< 2 \times 10^{-6})$	1×10^{-4} (2×10^{-6})	2.6×10^{-4} (2.6×10^{-2})[d]	1.6×10^{-7}	2.1×10^{0} (2.6×10^{-2})[d]	1.3×10^{-3}	3.5×10^{-2} (2.6×10^{-2})[d]	2.1×10^{-5}
4. Breach of sealed HLW canister during transfer (one drops onto another)	1×10^{-2} (2×10^{-4})	$< 1 \times 10^{-4}$ $(< 2 \times 10^{-6})$	1.0×10^{-4} (1.0×10^{-2})[d]	6.0×10^{-8}	8.5×10^{-1} (8.5×10^{1})[d]	5.1×10^{-4}	1.4×10^{-2} (1.4×10^{0})[d]	8.4×10^{-6}
5. Breach of uncanistered commercial SNF in a sealed truck transportation cask in air	1×10^{-1} (2×10^{-3})	not applicable[e]	1.0×10^{-3}	6.0×10^{-7}	2.7×10^{-5}	1.6×10^{-2}	8.3×10^{-2}	5.0×10^{-5}
6. Breach of uncanistered commercial SNF in an unsealed truck transportation cask in pool	7×10^{-4} (1×10^{-5})	2×10^{-4} (4×10^{-6})	9.4×10^{-4}	5.6×10^{-7}	2.6×10^{1}	1.6×10^{-2}	5.2×10^{-2}	3.1×10^{-5}
7. Breach of a sealed DPC in air	9×10^{-3} (2×10^{-6})	not applicable[e]	9.1×10^{-3}	5.5×10^{-6}	2.5×10^{2}	1.5×10^{-1}	5.5×10^{-2}	3.3×10^{-3}
8. Breach of commercial SNF in unsealed DPC in pool	$< 1 \times 10^{-4}$ $(< 2 \times 10^{-6})$	2×10^{-4} (4×10^{-6})	8.4×10^{-3}	5.0×10^{-6}	2.3×10^{2}	1.4×10^{-1}	7.4×10^{-1}	4.4×10^{-4}
9. Breach of a sealed TAD canister in pool	2×10^{-3} (4×10^{-5})	not applicable[e]	5.3×10^{-3}	3.2×10^{-6}	1.4×10^{2}	8.4×10^{-2}	4.3×10^{-1}	2.6×10^{-4}

Table E-11. Estimated radiological consequences of repository operations accident scenarios for unfavorable (95th-percentile) sector-specific meteorological conditions (continued).

Accident scenario	Expected occurrences over the preclosure period (annual frequency)		Maximally exposed offsite individual[a]		Population		Noninvolved worker	
	Internal events	Seismic events	Dose (rem)	LCF_i[b]	Dose (person-rem)	LCF_p[b]	Dose (rem)	LCF_i[b]
10. Breach of commercial SNF n unsealed TAD canister in pool	5×10^{-4} (1×10^{-5})	not applicable[e]	4.9×10^{-3}	2.8×10^{-6}	1.3×10^{2}	7.8×10^{-2}	2.9×10^{-1}	1.7×10^{-4}
11. Breach of uncanistered commercial SNF assembly in pool (one drops onto another)	3×10^{-1} (6×10^{-3})	not applicable[e]	4.7×10^{-4}	2.8×10^{-7}	1.3×10^{1}	7.8×10^{-3}	2.7×10^{-2}	1.6×10^{-5}
12. Breach of uncanistered commercial SNF assembly in pool	$< 1 \times 10^{-4}$ ($< 2 \times 10^{-6}$)	not applicable[e]	2.3×10^{-4}	1.4×10^{-7}	6.4×10^{0}	3.8×10^{-3}	1.4×10^{-2}	8.4×10^{-6}
13. Fire involving LLWF inventory	7×10^{-2} (1×10^{-3})	not applicable[e]	9.0×10^{-4}	5.4×10^{-7}	8.4×10^{0}	5.0×10^{-3}	8.1×10^{-2}	4.9×10^{-5}
14. Breach of a sealed truck transportation cask due to fire	2×10^{-2} (4×10^{-4})	not applicable[e]	4.4×10^{-3}	2.6×10^{-6}	4.2×10^{1}	2.5×10^{-2}	1.3×10^{0}	7.8×10^{-4}

a. For accident scenarios potentially initiated by more than one Category 2 event sequence, the expected occurrence value is the maximum frequency of those Category 2 event sequences. For accident scenarios potentially initiated by only Beyond-Category-2 event sequences, the expected occurrence value is less than the maximum frequency of a Beyond-Category-2 event over the preclosure period (that is, $< 1 \times 10^{-4}$).

b. Assumed to be at the analyzed land withdrawal boundary either in the east sector [7.8 kilometers (4.8 miles)] or in the southeast sector [18.5 kilometers (11 miles)], whichever produces the highest site boundary dose. For Scenarios 3 through 10, DOE calculated the highest dose for the southeast sector. For all other accident scenarios, DOE calculated the highest dose for the east sector.

c. LCF_i is the estimated likelihood of a latent cancer fatality for an individual who receives the calculated dose (rem). LCF_p is the estimated number of cancers in the exposed population from the collective population dose (person-rem). DOE based these values on a conversion of dose to LCFs as discussed in Section E.4.1.

d. Unfiltered doses presented to illustrate that filtration systems might not be required to meet regulatory standards for these accident scenarios.

e. The seismic event sequence quantification and categorization analysis (DIRS 183261-BSC 2008, Sections 6.7 and 6.8) did not identify any seismic initiators for these scenarios.

DPC = Dual-purpose canister.
HEPA = High-efficiency particulate air (filter).
HLW = High-level radioactive waste.
LCF = Latent cancer fatality.
LLWF = Low-Level Waste Facility.
SNF = Spent nuclear fuel.
TAD = Transportation, aging, and disposal (canister).

Table E-12. Estimated radiological consequences of repository operations accident scenarios for annual average (50th-percentile) sector-specific meteorological conditions.

	Expected occurrences over the preclosure period (annual frequency)[a]		Maximally exposed offsite individual[b]		Population		Noninvolved worker	
Accident scenario	Internal events	Seismic events	Dose (rem)	LCF_i[c]	Dose (person-rem)	LCF_p[c]	Dose (rem)	LCF_i[c]
1. Seismic event resulting in LLWF collapse and failure of HEPA filters and ductwork in other facilities	(not applicable)	8×10^{-3} (2×10^{-4})	6.4×10^{-4}	3.8×10^{-7}	2.5×10^{0}	1.5×10^{-3}	5.8×10^{-1}	3.5×10^{-4}
2. Breach of sealed HLW canisters in a sealed transportation cask	$< 1 \times 10^{-4}$ ($< 2 \times 10^{-6}$)	$< 1 \times 10^{-4}$ ($< 2 \times 10^{-6}$)	4.4×10^{-7}	2.6×10^{-10}	1.5×10^{-3}	9.0×10^{-7}	5.8×10^{-4}	3.5×10^{-7}
3. Breach of sealed HLW canister in an unsealed waste package	$< 1 \times 10^{-4}$ ($< 2 \times 10^{-6}$)	1×10^{-4} (2×10^{-6})	4.4×10^{-6}	2.6×10^{-9}	1.5×10^{-2}	9.0×10^{-6}	5.8×10^{-3}	3.5×10^{-6}
4. Breach of sealed HLW canister during transfer (one drops onto another)	1×10^{-2} (2×10^{-4})	$< 1 \times 10^{-4}$ ($< 2 \times 10^{-6}$)	1.8×10^{-6}	1.1×10^{-9}	5.9×10^{-3}	3.5×10^{-6}	2.3×10^{-3}	1.4×10^{-6}
5. Breach of uncanistered commercial SNF in a sealed truck transportation cask in air	1×10^{-1} (2×10^{-3})	not applicable[d]	2.6×10^{-5}	1.6×10^{-8}	2.7×10^{-1}	1.6×10^{-4}	2.3×10^{-2}	1.4×10^{-5}
6. Breach of uncanistered commercial SNF in an unsealed truck transportation cask in pool	7×10^{-4} (1×10^{-6})	2×10^{-4} (4×10^{-6})	1.2×10^{-5}	7.2×10^{-9}	1.5×10^{-1}	9.0×10^{-5}	9.0×10^{-3}	5.4×10^{-6}
7. Breach of a sealed DPC in air	9×10^{-3} (2×10^{-6})	not applicable[d]	2.4×10^{-4}	1.4×10^{-7}	2.5×10^{0}	1.5×10^{-3}	2.1×10^{-1}	1.3×10^{-4}
8. Breach of commercial SNF in unsealed DPC in pool	$< 1 \times 10^{-4}$ ($< 2 \times 10^{-6}$)	2×10^{-4} (4×10^{-6})	1.1×10^{-4}	6.6×10^{-8}	1.4×10^{0}	8.4×10^{-4}	8.1×10^{-2}	4.9×10^{-5}
9. Breach of a sealed TAD canister in air in facility	2×10^{-3} (4×10^{-5})	not applicable	1.4×10^{-4}	8.4×10^{-8}	1.4×10^{0}	8.4×10^{-4}	1.2×10^{-1}	7.2×10^{-5}

Table E-12. Estimated radiological consequences of repository operations accident scenarios for annual average (50th-percentile) sector-specific meteorological conditions (continued).

Accident scenario	Expected occurrences over the preclosure period (annual frequency)		Maximally exposed offsite individual[a]		Population		Noninvolved worker	
	Internal events	Seismic events	Dose (rem)	LCF_i[b]	Dose (person-rem)	LCF_p[b]	Dose (rem)	LCF_i[b]
10. Breach of commercial SNF n unsealed TAD canister in pool	5×10^{-4} (1×10^{-5})	2×10^{-4} (4×10^{-6})	6.2×10^{-5}	3.7×10^{-8}	7.9×10^{-1}	4.7×10^{-4}	4.7×10^{-2}	2.8×10^{-5}
11. Breach of uncanistered commercial SNF assembly in pool (one drops onto another)	3×10^{-1} (6×10^{-3})	not applicable[d]	5.9×10^{-6}	3.5×10^{-9}	7.5×10^{-2}	4.5×10^{-5}	4.5×10^{-3}	2.7×10^{-6}
12. Breach of uncanistered commercial SNF in pool	$< 1 \times 10^{-4}$ ($< 2 \times 10^{-6}$)	not applicable[d]	2.9×10^{-6}	1.7×10^{-9}	3.8×10^{-2}	2.3×10^{-5}	2.2×10^{-3}	1.3×10^{-6}
13. Fire involving LLWF inventory	3×10^{-1} (6×10^{-3})	not applicable[d]	1.7×10^{-5}	1.0×10^{-8}	7.3×10^{-2}	4.4×10^{-5}	1.3×10^{-2}	7.8×10^{-6}
14. Breach of a sealed truck transportation cask due to a fire	2×10^{-2} (4×10^{-4})	not applicable[d]	5.4×10^{-4}	3.2×10^{-7}	3.4×10^{0}	2.0×10^{-3}	7.1×10^{-1}	4.3×10^{-4}

a. For accident scenarios potentially initiated by more than one Category 2 event sequence, the expected occurrence value is the maximum probability of those Category 2 sequences. For accident scenarios potentially initiated by only Beyond-Category-2 event sequences, the expected occurrence value is less that the maximum frequency of a Beyond-Category-2 event over the preclosure period (that is, $< 1 \times 10^{-4}$).
b. Assumed to be at the analyzed land withdrawal boundary in the east sector, which would produce the highest site boundary dose at a distance of 7.8 kilometers (4.8 miles).
c. LCF_i is the estimated likelihood of a latent cancer fatality for an individual who receives the calculated dose (rem). LCF_p is the estimated number of cancers in the exposed population from the collective population dose (person-rem). These values were computed based on a conversion of dose to LCFs as discussed in Section E.4.1.
d. The seismic event sequence quantification and categorization analysis (DIRS 183261-BSC 2008, all) did not identify any seismic initiators for these scenarios.

DPC = Dual-purpose canister.
HEPA = high-efficiency particulate air (filter).
HLW = High-level radioactive waste.
LCF = Latent cancer fatality.
LLW = Low Level Waste Facility.
SNF = Spent nuclear fuel.
TAD = Transportation, aging, and disposal (canister).

E.4.3.1 Effect of Assuming Maximum Fuel

For all the accident analyses involving drops of commercial spent nuclear fuel, DOE used pressurized-water-reactor fuel containing the maximum inventory of radionuclides for any commercial spent nuclear fuel DOE could receive at the repository (maximum pressurized-water-reactor fuel). For doses from commercial spent nuclear fuel releases during normal operations, DOE used "representative" fuel (DIRS 185225-BSC 2008, p. 66), which represents the approximate annual average fuel DOE would receive in terms of burnup and cooling time. *Preclosure Consequence Analysis* (DIRS 185225-BSC 2008, Section 6.6.1) evaluates the effect of assuming pressurized-water-reactor maximum fuel versus representative fuel. As listed in Tables 38 and 39 of that analysis, the general environment (public site boundary) doses for 95-percent weather conditions for filtered releases for maximum pressurized-water-reactor fuel would be about twice those for representative fuel. For unfiltered releases, the maximum fuel would produce a dose about 32 times higher than the representative fuel for the same conditions.

E.4.3.2 Effect of Assuming No Mitigation

As noted, all accident consequence results assume no mitigation efforts to minimize doses. Such efforts could include evacuation and interdiction of contaminated food products. If effective mitigation efforts are assumed, the dose pathways would include only direct radiation from plume immersion and inhalation; the groundshine, resuspension, and ingestion doses would be eliminated. For filtered releases involving pressurized-water-reactor fuel for 95-percent weather conditions, the early (burst release) dose from plume immersion represents about 36 percent of the total dose (DIRS 185225-BSC 2008, Section 6.6.1, Appendix III, Table III-2). Therefore, assuming effective mitigation would reduce doses for these accidents by about two-thirds. For the seismic events evaluated, the plume immersion and inhalation doses would be over 85 percent of the total (DIRS 185225-BSC 2008, Appendix III, Table III-2). For events involving high-level radioactive waste, the plume inhalation dose alone represents 96 percent of the total (DIRS 185225-BSC 2008, Appendix III, Table III-2).

E.5 Monitoring and Closure Accident Scenarios

During monitoring and closure activities, DOE would not move the waste packages, with the possible exception of removal of a waste package from an emplacement drift for examination or drift maintenance. Because the analysis identified no accident scenarios unique to monitoring or closure, DOE conducted no additional analyses in this Repository SEIS.

E.6 Inventory Modules 1 and 2 Accident Scenarios

Inventory Modules 1 and 2 with cases A and B for each are alternative inventory options that this Repository SEIS considers for potential cumulative impacts in Chapter 8. These modules would involve additional waste material for emplacement in the repository. They would involve the same types of waste and handling activities as those for the Proposed Action, but the quantity would increase, as would the period of emplacement operations. As described in Chapter 8, Section 8.1.2.1, the Inventory Module 1B scenario would include a higher number of commercial high-level radioactive waste canisters than the Proposed Action; however, the accident consequences involving commercial high-level radioactive waste would be similar to those involving DOE high-level radioactive waste canisters. In addition, there would be a corresponding reduction in the number of commercial spent nuclear fuel handling operations in the Module 1B scenario. The analysis assumed the receipt and emplacement rates would remain the same as those for the Proposed Action. Therefore, the estimated consequences of the accident scenarios for operations would encompass the potential consequences of an accident in relation to Inventory Modules 1 and 2 because the same set of operations would be involved; therefore, DOE conducted no additional analyses in this Repository SEIS.

E.7 Representative Sabotage Scenario

In response to the terrorist attacks of September 11, 2001, and to intelligence information obtained since then, the United States Government has initiated nationwide measures to reduce the threat of sabotage. These measures include security enhancements to prevent terrorists from gaining control of commercial aircraft, such as (1) more stringent screening of airline passengers and baggage by the Transportation Security Administration, (2) increased presence of federal air marshals on many flights, (3) improved training of flight crews, and (4) hardening of aircraft cockpits. The measures have imposed additional measures on foreign passenger carriers and domestic and foreign cargo carriers, as well as charter aircraft.

Over the long term (after closure), deep geologic disposal of spent nuclear fuel and high-level radioactive waste would provide optimal security by emplacing the material in a geologic formation that would provide protection from inadvertent and advertent human intrusion, including potential terrorist activities. The use of robust metal waste packages to contain the spent nuclear fuel and high-level radioactive waste more than 200 meters (660 feet) below the surface would offer significant impediments to an attempt to retrieve or otherwise disturb the emplaced materials.

In the short term (before closure), the proposed repository at Yucca Mountain would offer certain unique features from a safeguards perspective: a remote location, restricted access afforded by federal land ownership and proximity to the Nevada Test Site, restricted airspace above the site, and access to a highly effective rapid-response security force.

NRC regulations (10 CFR 63.21 and 10 CFR 73.51) specify a repository performance objective that provides "high assurance that activities involving spent nuclear fuel and high-level radioactive waste do not constitute an unreasonable risk to public health and safety." The regulations require the storage of spent nuclear fuel and high-level radioactive waste in a protected area such that:

- Access to the material would require passage through or penetration of two physical barriers. The outer barrier must have isolation zones on each side to facilitate observation and threat assessment, be continually monitored, and be protected by an active alarm system.
- Adequate illumination must be provided for observation and threat assessment.
- The area must be monitored by random patrol.
- Access must be controlled by a lock system, and personnel identification must be used to limit access to authorized persons.

NRC regulations would require a trained, equipped, and qualified security force to conduct surveillance, assessment, access control, and communications to ensure adequate response to a security threat. The

NRC requires liaison with response forces to permit timely response to unauthorized entry or activities. The NRC also requires (10 CFR Part 63, by reference to 10 CFR Part 72) that comprehensive receipt, periodic inventory, and disposal records be kept for spent nuclear fuel and high-level radioactive waste in storage. A duplicate set of these records must be kept at a separate location sufficiently remote from the original records that a single event would not destroy both sets of records.

Whether acts of sabotage or terrorism would occur, and the exact nature and location of the events, or the magnitude of the consequences of such acts if they were to occur is inherently uncertain—the possibilities are infinite. Nevertheless, in response to public comments and to evaluate a scenario that would approximate the consequences of a major sabotage event, DOE analyzed a hypothetical scenario in which a large commercial jet aircraft would crash into and penetrate the repository facility with the largest inventory of radioactive material vulnerable to damage from such an event. Table E-13 lists the potentially affected amounts of radiological materials in major surface buildings. The Aging Facility could contain a large amount of commercial spent nuclear fuel, but DOE did not consider this location to be vulnerable to the aircraft crash scenario because (1) the aging overpacks on the Aging Facility pads would be 5.5 meters (18 feet) apart (DIRS 184100-BSC 2007, all), such that an aircraft crash into the pad could not damage more than a few of the overpacks, and (2) the storage canisters would be enclosed in thick concrete overpacks that would provide protection from penetration by aircraft parts (DIRS 155970-DOE 2002, Appendix H, p. H-37 and Chapter 7, p. 7-30). Further, as noted in Section E.2.1.2.1, DOE would design the TAD aging overpacks to withstand an impact from a jet fighter aircraft crash.

As listed in the table, the Wet Handling Facility would contain the most material. However, most of the fuel assemblies would be underwater in the below-ground storage pool. Similar to the conclusion in the Yucca Mountain FEIS (DIRS 155970-DOE 2002, Appendix H, p. H-38), fuel in this pool would not be vulnerable to an aircraft crash because the pool water would limit the potential for a fire to affect the fuel directly and would limit releases from damaged fuel assemblies. The next largest number of fuel assemblies from Table E-13 would be 168 fuel assemblies in eight TAD canisters in a Canister Receipt and Closure Facility. As the table indicates, nine canisters of DOE spent nuclear fuel could be in a Canister Receipt and Closure Facility at the same time. However, the analysis did not consider the DOE spent nuclear fuel inventory for the sabotage consequence calculation because these canisters would remain sealed while in the Canister Receipt and Closure Facilities. Further, DOE would design spent nuclear fuel canisters to preclude a breach if dropped during handling operations (Section E.2.1.1). The canisters would be robust steel containers that provided protection for DOE spent nuclear fuel during the aircraft crash event. Further, the radionuclide inventory in spent nuclear fuel canisters would be significantly less than that for a representative pressurized-water-reactor assembly. This can be seen by comparing the DOE spent nuclear fuel inventories in Appendix A, Table A-21 of the FEIS (DIRS 155970-DOE 2002, Appendix A, Table A-21) with the pressurized-water-reactor representative assembly inventory in Table 6 of the *Preclosure Consequence Analysis* report (DIRS 185225-BSC 2008, all) for the radionuclides important to offsite consequences (DIRS 185225-BSC 2008, Table III-1, p. III-8). Therefore, the breach and radionuclide release from a TAD canister containing 21 pressurized-water-reactor representative fuel assemblies would bound the consequences of a breach of nine DOE spent nuclear fuel canisters. The analysis assumed representative, rather than maximum, fuel assemblies for the sabotage event to provide a more realistic estimate of impacts. Of the eight TAD canisters in the Canister Receipt and Closure Facility, two would be in sealed waste packages, one would be in an aging overpack, and one would be in a sealed transportation cask. These canisters would be protected from aircraft damage. Of the four remaining TAD canisters, only two would be vulnerable to damage from the aircraft

Table E-13. Materials at risk for aircraft crash scenario.

Waste Form	Quantity			
	Receipt Facility	Initial Handling Facility	Wet Handling Facility	Cask Receipt and Closure Facility
DPC in AO	1	0	0	0
DPC in open transportation cask	1	0	0	0
DPC in sealed transportation cask	1	0	0	0
HLW or Naval SNF canister in sealed transportation cask	0	1	0	0
HLW or Naval canister in WP	0	5 HLW, 1 Naval	0	0
Transportation cask with uncanistered SNF	0	0	1	0
DPC in STC	0	0	1	0
DPC or TAD in AO or STC	0	0	1	0
DPC in STC or transportation cask with uncanistered SNF	0	0	1	0
TAD in STC	0	0	2	0
Transportation cask	0	0	1	0
DPC in STC (pool)	0	0	1	0
TAD in STC (pool)	0	0	1	0
SNF assemblies (pool)	0	0	213	0
Sealed WP with TAD	0	0	0	2
TAD	0	0	0	4
DOE SNF or HLW canisters (staging area 2)	0	0	0	4
DOE SNF or HLW canisters (staging area 4)	0	0	0	6
Sealed transportation cask with TAD	0	0	0	1
AO with TAD	0	0	0	1

Source: DIRS 185404-Dunn 2008, all.
AO = Aging overpack
DPC = Dual-purpose canister.
HLW = High-level radioactive waste.
SNF = Spent nuclear fuel.
STC = Shielded transfer cask.
TAD = Transportation, aging, and disposal (canister).
WP = Waste package.

crash because the four canisters would be in two locations (canister staging area 1 with an adjacent waste package positioning room and canister staging area 3 with an adjacent cask unloading room) (DIRS 181268-BSC 2007, all). These locations would be separated by distance as well as by two thick, reinforced concrete walls (DIRS 181268-BSC 2007, all). Therefore, DOE selected two TAD canisters containing 42 pressurized-water-reactor representative fuel assemblies as the source term for the aircraft crash sabotage event.

For the representative scenario, DOE assumed the aircraft would penetrate the roof of the building and the aircraft parts and debris from the roof impact would breach the two TAD canisters and rupture 100 percent of the fuel rods in the canisters. DOE also assumed the fuel aboard the aircraft would catch fire and heat and oxidize all the commercial spent nuclear fuel assembly pellets in the 42 fuel assemblies into powder form. The radionuclide release from the scenario would result from two sources: (1) mechanical damage to the fuel assemblies that would rupture the Zircaloy cladding, release activity in the gap, and pulverize a portion of the fuel pellets into particles (some of which would be small enough for transport to the nearest receptor and inhalation) and (2) the large fire from the jet fuel. DOE conservatively assumed that the fire would convert all the fuel in the two TAD canisters (42 assemblies from pressurized-water-reactor spent nuclear fuel) from uranium dioxide to uranium trioxide and produce a powder that contained radionuclides. Because all the fuel pellet material in the 42 assemblies would become powder, the particulates from the mechanical damage would not contribute further to the source term. The analysis assumed that 12 percent of the uranium trioxide particles would become airborne and 1 percent of the airborne particles would be respirable (small enough for downwind receptors to inhale into the lungs) (DIRS 155970-DOE 2002, Appendix H, p. H-38) Therefore, the analysis assumed that the fuel pellet respirable particulate source term would be 0.12 percent of the radionuclides in the 42 fuel assemblies. DOE assumed that the release would occur at ground level. This is conservative because the fire from the aircraft fuel would tend to loft the plume containing the radionuclides. This would result in increased plume dispersion and lower downwind radionuclide concentrations. For the radionuclides in gas form (chlorine, hydrogen, iodine, krypton, and carbon), the respirable fraction is 1.0. The analysis assumed the radionuclide inventory in the assemblies would be the representative fuel (DIRS 180185-BSC 2007, all), which would have a burnup of 50 gigawatt-days per metric ton of uranium and a cooling time of 10 years. It would not be realistic to assume that the fuel in the Canister Receipt and Closure Facility for this scenario would be the same as the maximum fuel (Section E.3.3) for the accident scenarios. The representative fuel represents a conservative estimate of the characteristics of the large number of commercial spent nuclear fuel assemblies that would be in a Canister Receipt and Closure Facility at any time during the year (DIRS 180185-BSC 2007, all). The crud source term would include 209 curies of iron-55 and 16.9 curies of cobalt-60 per assembly (DIRS 180185-BSC 2007, all). Consistent with the Yucca Mountain FEIS analysis (DIRS 155970-DOE 2002, Appendix H, p. H-38), the accident would release all the iron and cobalt because the Zircaloy cladding would burn. The respirable airborne release fraction for the radionuclides in the crud would be 0.05 (DIRS 103711-Davis et al. 1998, all). Table E-14 lists the source term for the aircraft crash scenario.

The analysis used the GENII computer program to calculate the consequences from the crash with the assumptions in Section E.4.1; however, for this case, due to the large release and potential for large doses, it assumed mitigation would occur. Mitigation measures would include evacuation of the affected population after 24 hours and interdiction of contaminated crops so consumption of contaminated food would not occur. Table E-15 lists the results of the consequence evaluation for the scenario for annual average weather conditions. The Repository SEIS analysis assumed that the wind would blow to the south-southeast and expose the entire population in this sector (104,000 persons).

Table E-14. Source term (curies) for the aircraft crash scenario.

Radionuclide	Per PWR assembly	Per 42 assemblies	Respirable airborne release
Americium-241	1.2×10^{3}	5.0×10^{4}	6.0×10^{1}
Americium-242	7.2×10^{0}	3.0×10^{2}	3.6×10^{-1}
Americium-242m	7.2×10^{0}	3.0×10^{2}	3.6×10^{-1}
Americium-243	3.5×10^{1}	1.5×10^{3}	1.8×10^{0}
Barium-137m	6.1×10^{4}	2.6×10^{6}	3.1×10^{3}
Carbon-14	4.0×10^{-1}	1.7×10^{1}	1.7×10^{1}
Cadmium-113m	2.1×10^{1}	9.0×10^{2}	1.8×10^{0}
Chlorine-36	8.0×10^{-3}	3.4×10^{-1}	3.4×10^{-1}
Curium-242	5.9×10^{0}	2.5×10^{2}	3.0×10^{-1}
Curium-243	2.2×10^{1}	9.2×10^{2}	1.1×10^{0}
Curium-244	5.3×10^{3}	2.2×10^{5}	2.7×10^{2}
Curium-245	7.3×10^{-1}	3.1×10^{1}	3.7×10^{-2}
Curium-246	3.7×10^{-1}	1.6×10^{1}	1.9×10^{-2}
Cobalt-60	1.7×10^{1}	7.1×10^{2}	8.4×10^{-1}
Cesium-134	4.9×10^{3}	2.1×10^{5}	2.5×10^{2}
Cesium-135	3.4×10^{-1}	1.4×10^{1}	1.7×10^{-2}
Cesium-137	6.4×10^{4}	2.7×10^{6}	3.2×10^{3}
Europium-154	2.7×10^{3}	1.1×10^{5}	1.4×10^{2}
Europium-155	5.8×10^{2}	2.4×10^{4}	2.9×10^{1}
Iron-55	2.1×10^{2}	8.8×10^{3}	1.1×10^{1}
Hydrogen-3	2.8×10^{2}	1.2×10^{4}	1.2×10^{4}
Iodine-129	3.0×10^{-2}	1.3×10^{0}	1.3×10^{0}
Krypton-85	3.1×10^{3}	1.3×10^{5}	1.3×10^{5}
Niobium-93m	2.3×10^{1}	9.7×10^{2}	1.1×10^{0}
Niobium-94	8.1×10^{-1}	3.4×10^{1}	4.1×10^{-2}
Nickel-59	1.7×10^{0}	7.1×10^{1}	8.5×10^{-2}
Nickel-63	2.4×10^{2}	1.0×10^{4}	1.2×10^{1}
Neptunium-237	2.6×10^{-1}	1.1×10^{1}	1.3×10^{-2}
Protactinium-231	1.6×10^{-5}	6.7×10^{-4}	8.0×10^{-7}
Palladium-107	1.1×10^{-1}	4.6×10^{0}	5.5×10^{-3}
Promethium-147	5.5×10^{3}	2.3×10^{5}	2.8×10^{2}
Plutonium-238	3.6×10^{3}	1.5×10^{5}	1.8×10^{2}
Plutonium-239	1.6×10^{2}	6.7×10^{3}	7.8×10^{-1}
Plutonium-240	3.3×10^{2}	1.4×10^{4}	1.7×10^{1}
Plutonium-241	5.1×10^{4}	2.1×10^{6}	2.6×10^{3}
Plutonium-242	2.2×10^{0}	9.2×10^{1}	1.1×10^{-1}
Ruthenium-106	3.6×10^{2}	1.5×10^{4}	1.8×10^{1}
Antimony-125	4.7×10^{2}	2.0×10^{4}	2.4×10^{1}
Selenium-79	5.0×10^{-2}	2.1×10^{0}	2.5×10^{-3}
Samarium-151	2.3×10^{2}	9.7×10^{3}	1.1×10^{1}
Tin-126	4.6×10^{-1}	1.9×10^{1}	2.3×10^{-2}
Strontium-90	4.1×10^{4}	1.7×10^{6}	2.0×10^{3}
Technetium-99	9.6×10^{0}	4.0×10^{2}	4.9×10^{-1}
Thorium-230	5.5×10^{-5}	2.3×10^{-3}	2.8×10^{-6}
Uranium-232	3.3×10^{-2}	1.4×10^{0}	1.7×10^{-3}
Uranium-233	2.3×10^{-5}	9.7×10^{-4}	1.1×10^{-6}
Uranium-234	4.7×10^{-1}	2.0×10^{1}	2.3×10^{-2}
Uranium-235	3.8×10^{-3}	1.6×10^{-1}	1.9×10^{-4}
Uranium-236	1.6×10^{-1}	6.7×10^{0}	8.0×10^{-3}
Uranium-238	1.3×10^{-1}	5.5×10^{0}	6.6×10^{-3}
Yttrium-90	4.1×10^{4}	1.7×10^{6}	2.0×10^{3}
Zirconium-93	9.4×10^{-1}	3.9×10^{1}	4.7×10^{-2}

PWR = Pressurized-water reactor.

Table E-15. Estimated doses and latent cancer fatality estimates for aircraft crash scenario.

Receptor	Dose	Latent cancer fatalities
Maximally exposed offsite individual	3.0 rem	1.8×10^{-3}(a)
84-kilometer (52-mile) population	9.9×10^{3} person-rem	5.9[b]

Note: These results are somewhat lower than the Draft Repository SEIS results because the Draft SEIS results were mistakenly calculated using maximum pressurized-water reactor fuel rather than representative pressurized-water reactor fuel.
Source: DIRS 185403-Schultz 2008, all.
a. Estimated likelihood of a latent cancer fatality for an individual who receives the calculated dose.
b. Estimated number of cancers in the exposed population from the collective population dose.

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Appendix F

Environmental Impacts of
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F. ENVIRONMENTAL IMPACTS OF POSTCLOSURE REPOSITORY PERFORMANCE

This appendix provides detailed information on the calculation of the environmental impacts of the postclosure period of repository performance. Chapter 5 of this *Final Supplemental Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F-S1) (Repository SEIS) summarizes these impacts for the Proposed Action. This appendix summarizes, incorporates by reference, and updates Appendix I of the *Final Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F; DIRS 155970-DOE 2002, pp. I-1 to I-94) (Yucca Mountain FEIS). Since completion of the FEIS, DOE has modified the Total System Performance Assessment (TSPA) model it uses to assess long-term repository performance to account for regulatory, design, data, model, and analysis changes since 2002. For this Repository SEIS, DOE based the analysis on *Total System Performance Assessment Model/Analysis for the License Application* (DIRS 183478-SNL 2008, all) (TSPA-LA).

Section F.1 introduces the bases for analysis of postclosure performance. Section F.2 provides an overview of the use of computational models the U.S. Department of Energy (DOE or the Department) developed for the TSPA-LA model. Section F.3 identifies and quantifies the inventory of waste constituents of concern for analysis of postclosure performance. Section F.4 provides detailed results for radioactive material impacts, and Section F.5 provides the results for waterborne chemically toxic material impacts.

F.1 Introduction

The model that DOE used to evaluate postclosure impacts of radioactive materials in the groundwater simulates the release and transport of radionuclides away from the proposed repository into the unsaturated zone, through the unsaturated zone, and ultimately through the saturated zone to the accessible environment. Analysis of postclosure performance depended on the underlying process models necessary to provide thermal-hydrologic conditions, near-field geochemical conditions, degradation characteristics of the Engineered Barrier System, and unsaturated and saturated zone flow fields as a function of time. The use of these underlying process models involved multiple sequential steps before modeling of the overall system could begin.

Figure F-1 shows the general flow of information among data sources, process models, and the TSPA-LA model. The figure identifies several process-level computer models (for example, the site- and drift-scale thermal hydrology model and the saturated zone flow and transport model). The process models are large, complex computer programs that DOE used in detailed studies to provide information to the TSPA-LA model. These process models are based on fundamental laboratory and field data DOE introduced into the modeling. The subsystem and abstracted models section of the figure encompasses those portions of the TSPA-LA model that the probabilistic simulation software, GoldSim, models (for example, the unsaturated zone flow fields and the biosphere dose conversion factors). These models are generally much simpler than the process models. They represent the results of the more detailed process

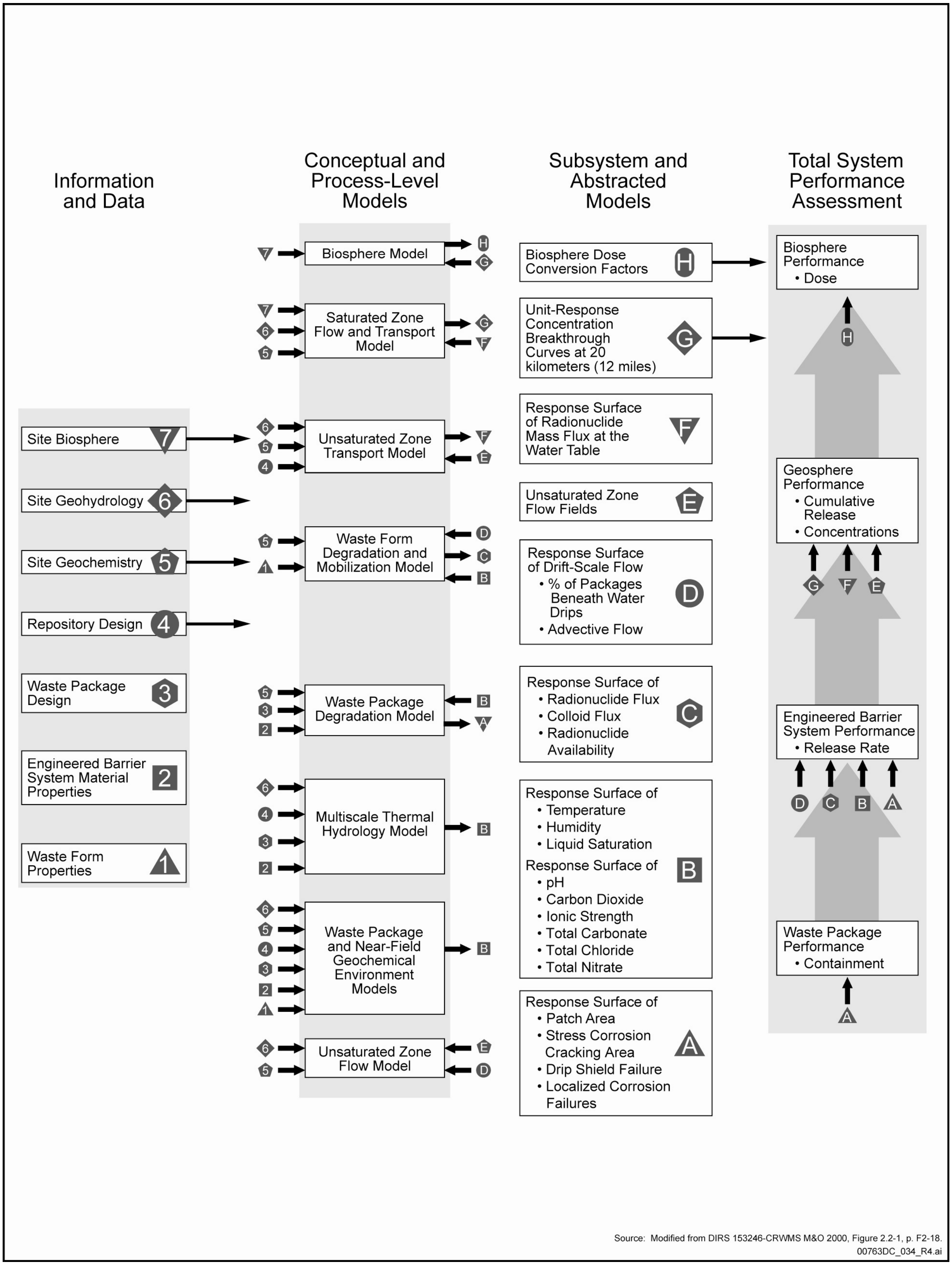

Figure F-1. Information flow in the TSPA-LA model.

modeling studies. They often are simple functions or tables of numbers. This process is called abstraction. It is necessary for some of these subsystem models to be complex, even extensive, computer programs. The result that DOE sought from modeling postclosure performance was a characterization of radiological dose to humans in relation to time (at the top of the TSPA section of Figure F-1). The model accomplished this by an assessment of behavior at intermediate points and "handing off" the results to the next subsystem in the primary release path.

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 maintaining the relevant aspects of features, processes, and events that could affect postclosure performance.

F.2 Total System Performance Assessment Methods and Models

DOE conducted analyses for this Repository SEIS to evaluate potential postclosure impacts to human health from the release of radioactive materials from the proposed repository. The TSPA-LA model started with the model in the Yucca Mountain FEIS and includes several enhancements. Table 5-1 in Chapter 5 summarizes these enhancements.

A TSPA is a comprehensive systems analysis in which models of appropriate levels of complexity represent all important features, events, and processes to estimate the behavior of the system under analysis and to compare this behavior with specified performance standards. In the case of the Yucca Mountain Repository system, a TSPA must capture the important components of both the engineered and the natural barriers. In addition, it must evaluate the overall uncertainty in the projection of waste containment and isolation, and the risks such uncertainties cause in the individual component models and corresponding parameters.

The components of the Yucca Mountain Repository system would include six major elements that the TSPA model has evaluated:

- Water flow from the ground surface through the unsaturated tuffs above and below the repository horizon, which would include water that dripped into the waste emplacement drifts;

- Thermal and chemical environments in the Engineered Barrier System, effects of disruptive events on that system, and perturbations to the surrounding natural system due to waste emplacement;

- The degradation of the engineered components that would contain the radioactive wastes;

- The release of radionuclides from the Engineered Barrier System;

- The migration of these radionuclides through the engineered and natural barriers to the biosphere and their potential uptake by people, which could lead to a radiation dose consequence; and

- The analysis includes models for disruptive events such as igneous activity, seismicity, and a hypothetical human intrusion (drilling).

This Repository SEIS analysis represents a snapshot in time of postclosure performance, and ongoing work will refine that snapshot.

The analysis for this Repository SEIS used a probabilistic framework for calculations that combined the most likely ranges of behavior for the component models, processes, and related parameters. In some cases, the analysis used bounding conservative values if the available data did not support development of a realistic range. This appendix presents the results as projections over time of annual radiological dose to an individual for the first 10,000 and the post-10,000-year period (up to 1 million years after repository closure). As noted in Section F.1, the TSPA-LA model provides a framework for incorporation of information from process models and abstraction models into an integrated representation of the repository system. This integration occurred in a Monte Carlo simulation-based method to create multiple random combinations of the likely ranges of the parameter values for the process models. The model computed the probabilistic performance of the entire waste disposal system in terms of radiological doses to the RMEI at a distance of approximately 18 kilometers (11 miles) south of the repository (the predominant direction of groundwater flow).

MONTE CARLO METHOD: UNCERTAINTY

Monte Carlo is an analytical method that uses random sampling of parameter values available for input into numerical models as a means to approximate the uncertainty in the process being modeled. A Monte Carlo simulation consists of many individual runs of the complete calculation, which uses different values for the parameters of interest sampled from a probability distribution. A different outcome for each calculation and each run of the calculation is called a realization.

F.2.1 FEATURES, EVENTS, AND PROCESSES

The first step in a TSPA is to determine the representations of possible future states of the proposed repository (scenarios and scenario classes). A scenario is a well-defined, connected sequence of events and processes that describes a possible future state of the repository system. A scenario class is a set of related scenarios that share sufficient similarities that can usefully be aggregated for the purposes of screening or analysis. The objective of scenario analysis for the TSPA is to define a set of scenario classes that can be quantitatively analyzed while maintaining comprehensive coverage of the range of possible future states of the repository system.

The first step in the development of scenario classes is to make an exhaustive list of the features, events, and processes that could apply to the repository system. Development of the initial list used a number of resources:

- Lists from other organizations on an international scale (such as the Nuclear Energy Agency or the Organization for Economic Cooperation and Development),

- Lists from earlier stages of site characterization, and

- Lists from experts from the Yucca Mountain Project and outside consultants.

The analysis subjected the starting list to a comprehensive screening process. It used the following criteria to screen out features, events, and processes:

- Inapplicability to the specific site (for example, the starting list included processes that occur only in salt, which is not present at Yucca Mountain),
- Very low probability of occurrence (for example, meteorite impact),
- Very low consequence to the closed repository (for example, an airplane crash), and
- Exclusion by regulatory direction (for example, deliberate human intrusion).

The analysis combined the remaining features, events, and processes in scenario classes that incorporate sequences of events and processes in the presence of features. The four main scenario classes are:

- Nominal Scenario Class (generally undisturbed performance)
- Early Failure Scenario Class (failure of drip shields or waste packages caused by manufacturing defects)
- Igneous Scenario Class (events and processes initiated by eruption through the repository or intrusion of igneous material into the repository)
- Seismic Scenario Class (events and processes initiated by ground motion or fault displacement)

In addition, the analysis evaluated a stylized inadvertent Human Intrusion Scenario.

When DOE formed these scenario classes from the features, events, and processes that remained after screening, its focus was on the 10,000-year compliance period. The proposed U.S. Environmental Protection Agency (EPA) and U.S. Nuclear Regulatory Commission (NRC) standards specify that features, events, and processes excluded from the TSPA for the 10,000-year period after disposal may be excluded from the TSPA for the additional compliance period of geologic stability after 10,000 years, with the exception of features, events, and processes that relate to specific effects of seismicity, igneous activity, general corrosion, and climate change. The proposed standards also specify a value to be used to represent climate change after 10,000 years. Therefore, this Repository SEIS analysis and projections of repository performance include the combined effects of seismicity (F.2.11), igneous activity (F.2.10), general corrosion (Section F.2.4), and the prescribed representation of climate change (Section F.2.2). In the Yucca Mountain FEIS, general corrosion and climate change were included. Igneous activity was not included directly in the combined calculation of repository performance, but was analyzed separately to estimate potential impacts from igneous activity alone. The FEIS analysis did include seismic activity and its effects on repository performance; however, processes representing seismic damage to waste packages were screened out for the 10,000-year period after disposal. The FEIS analysis for the post-10,000-year period extended the screening of seismic damage to waste packages throughout that time. This was an analytical assumption based on using the best data and models available for the FEIS. No quantitative analysis was performed to determine when a waste package might degrade to the point where it could be damaged by a seismic event.

The mechanical response of Engineered Barrier System components to seismic hazards was included in the TSPA-LA analysis of potential seismic events for the 10,000-year period after disposal and the period of geologic stability, and is thus included in this Repository SEIS. The addressed seismic hazards included vibratory ground motion, fault displacement, and drift collapse due to ground motion. The major Engineered Barrier System components DOE considered in this analysis were the drip shield and the waste package because failure of these components could form advective and diffusive pathways that could result in the direct release of radionuclides from the Engineered Barrier System into the unsaturated zone. The drift invert and emplacement pallet were included in the structural response analyses for the Engineered Barrier System; however, it was not necessary to develop damage models for these components because they could not form new pathways for transport and release of radionuclides after seismic events. The waste package internal components and the waste form were also considered in structural response analyses. However, in this SEIS, credit was not taken for the fuel rod cladding as a barrier to radionuclide release, so it was not necessary to include cladding damage due to a seismic event.

The following discussions provide a description of each seismic-related feature, event, and process that was included in this Repository SEIS followed by a brief description of how that feature, event, and process was included in the TSPA-LA model.

F.2.1.1 Seismic Ground Motion Damages Engineered Barrier System Components (FEP No. 1.2.03.02.0A)

Seismic activity that caused repeated vibration of the Engineered Barrier System components (drip shield, waste package, pallet, and invert) could result in disruption of the drip shields and waste packages through vibration damage or through contact between Engineered Barrier System components. Such damage mechanisms could lead to degraded performance.

Structural calculations were used to simulate the response of the drip shield and waste package to vibratory ground motion. These calculations utilized a three-dimensional, dynamic structural analysis model that incorporated the details of the Engineered Barrier System design. Ground motion time histories input into the calculations represented postclosure hazard levels at the emplacement depth. The potential for structural damage and for separation of the drip shields was examined. The potential damage to the waste package due to ground motion-induced interactions of the waste packages, the pallet, and the drip shield was examined. Using these analyses, surface area damage was determined for input to the damage abstractions for the drip shield and waste package. Results of these studies were used in creating damage abstractions that were implemented in the TSPA-LA model for the Seismic Scenario Class.

F.2.1.2 Seismic-Induced Drift Collapse Alters In-Drift Thermohydrology (FEP No. 1.2.03.02.0D)

Seismic activity could produce jointed-rock motion and/or changes in rock stress leading to enhanced drift collapse and/or rubble infill throughout part or all of the drifts. Drift collapse could affect flow pathways and condensation within the Engineered Barrier System, mechanisms for water contact with Engineered Barrier System components, and thermal properties within the Engineered Barrier System.

The potential for drift collapse and/or rubble infill associated with vibratory ground motion was assessed using detailed two- and three-dimensional tunnel stability models. Ground motion time histories input into the calculations represent postclosure hazard levels at the emplacement depth. Emplacement drift profiles and the porosity of rubble material in the drift following a seismic event were used as input to a series of thermal-hydrologic simulations for representative in-drift conditions. These simulations were used to develop thermal-hydrologic abstractions that were implemented in the TSPA-LA to account for the effect of drift collapse on thermal-hydrologic conditions in the drift for the Seismic Scenario Class.

F.2.1.3 Seismic-Induced Drift Collapse Damages Engineered Barrier System Components (FEP No. 1.2.03.02.0C)

Seismic activity could produce jointed-rock motion and/or changes in rock stress leading to enhanced drift collapse that could impact drip shields, waste packages, or other Engineered Barrier System components. Possible effects include both dynamic and static loading.

Structural calculations were used to simulate the response of the drip shield and waste package to vibratory ground motion and drift collapse. These calculations were used to quantify drip shield damage in terms of fragility curves on the peak ground velocity value for a given seismic event and the thickness of the drip shield components at the time of the seismic event. The effects of drift collapse on waste packages were quantified in terms of damaged areas or puncture areas based on the peak ground velocity value for a given seismic event and the thickness of the waste package outer corrosion barrier at the time of the seismic event. The fragility curves and damaged areas were used to develop drip shield and waste package damage abstractions that were implemented in the TSPA-LA model.

F.2.1.4 Fault Displacement Damages Engineered Barrier System Components (FEP No. 1.2.02.03.0A)

Movement of a fault that intersects drifts within the repository could cause the Engineered Barrier System components to experience related movement or displacement. Repository performance could be degraded by such occurrences as tilting of components, component-to-component contact, or drip shield separation. Fault displacement could cause a failure as significant as shearing of drip shields and waste packages by virtue of the relative offset across the fault, or as extreme as exhumation of the waste to the surface.

An analysis was performed that examined how fault displacement could contribute to mechanical disruption of the Engineered Barrier System. In that analysis, estimates of very low probability fault displacement were compared with the dimensions of the Engineered Barrier System features. Potential damage to the Engineered Barrier System was conservatively estimated, and the results were used to create drip shield and waste package damage abstractions that were implemented in the TSPA-LA. The output of these abstractions is the number of drip shields and waste packages that fail by fault displacement and the combined surface area from the waste packages that fail from fault displacement; affected drip shields were assumed to completely fail.

F.2.2 UNSATURATED ZONE FLOW

F.2.2.1 Climate Model

Changes in climate over time provide a range of conditions that determine how much water could fall on and infiltrate the surface of Yucca Mountain. Based on current scientific estimates, the current climate is the driest that the Yucca Mountain vicinity is ever likely to experience (DIRS 169734-BSC 2004, Sections 6.4 and 6.5). This Repository SEIS analysis assumed that all future climates would be similar to or wetter than current conditions. The climate model provided an estimate of future climates based on information about past climate patterns (DIRS 170002-BSC 2004, all). This is generally accepted as a valid approach because climate is cyclical and largely dependent on repeating patterns of the Earth's orbit and spin. The model represented future climate shifts as a series of instant changes. During the first 10,000 years, there would be three changes, in order of increasing wetness, from present-day (0 to 600 years) to monsoon (600 to 2,000 years) and then to glacial-transition climate (2,000 to 10,000 years). In its proposed changes to 10 CFR 63.342(c), the NRC directed DOE to represent climate change after 10,000 years (the post-10,000-year climate) with a constant value determined from a log-uniform probability distribution for deep percolation rates from 13 to 64 millimeters (0.5 to 2.5 inches) per year.

Precipitation that did not return to the atmosphere by evaporation or plant transpiration could enter the unsaturated zone flow system. A number of factors that relate to climate, such as an increase or decrease in vegetation on the ground surface, total precipitation, air temperature, and runoff, could affect water infiltration. The infiltration model for the Yucca Mountain FEIS was completely revised for this Repository SEIS. The purpose of the revision was to increase confidence in the results by improving the traceability, transparency, and reproducibility of the model development; the selection and qualification of inputs for calculations; and the determination of net infiltration maps and fluxes. The revised infiltration model used data from studies of surface infiltration in the Yucca Mountain region (DIRS 182145-SNL 2008, all). The model applied a water mass-balance approach to the near-surface layer that is influenced by evapotranspiration. It used a representation of downward water flow whereby water moves from the top soil layer downward by sequentially filling each layer to "field capacity" before draining to the layer below. Water was removed from the "root zone" by evapotranspiration, which was represented using an empirical model based on reference evapotranspiration, transpiration coefficients, and moisture content in the root zone. Water was redistributed as surface runoff when the soil could not accept all the available water at the surface. Precipitation was stochastically simulated on a daily time step based on observed weather records.

The results of the climate model affected infiltration rates. For each climate (present-day, monsoon, glacial transition, and post-10,000-year), there was a set of four infiltration rates (10th-, 30th-, 50th-, and 90th-percentile values) to represent uncertainty in infiltration rate. The corresponding weighting factors of 61.91, 15.68, 16.45, and 5.96 were used to describe the probability of occurrence for each of the four infiltration scenarios; therefore, the sum of the four weighting factors is 1. The same weighting factors were used in all four climate states of present-day, monsoon, glacial transition, and post-10,000 years (DIRS 183478-SNL 2008, Section 6.3 and Table 6.3.1-2).

Comparisons between unsaturated zone flow model simulations using the four infiltration scenarios and measured subsurface values of chloride and temperature data in combination with a likelihood uncertainty estimation methodology were used to determine the weighting factors; higher weights were given to infiltration maps that best match chloride and temperature data (DIRS 184614-SNL 2008, all). The infiltration rates and weighting factors form a discrete distribution that is sampled in the probabilistic modeling. The four infiltration cases represent epistemic uncertainty in the net infiltration rates. The TSPA-LA model sampled these infiltration cases once per realization (Table F-1) consistent with their weighting factors so that, for example, the 10th-percentile value was selected in approximately 62 percent of the realizations. Because of the once-per-realization sampling, the infiltration cases are completely correlated across the four climate states modeled for the simulation period (for example, during a realization in which the 50th-percentile infiltration case was sampled, that case would be used for each of the four climate states to select the appropriate unsaturated zone flow fields). This correlation of the infiltration uncertainty across the climate transitions ensures that the full effects of the infiltration uncertainty are not dampened out of the TSPA-LA model performance results.

The four post-10,000-year net infiltration rates Table F-1 lists correspond to four infiltration maps that were developed to satisfy the log-uniform probability distribution for deep percolation rates from 13 to 64 millimeters (0.5 to 2.5 inches) per year, as the NRC directed. These four infiltration maps were developed by selecting, from the available 12 infiltration maps implemented for the first 10,000-year period after closure, the map that has an average infiltration rate through the repository footprint that most closely matches the required value (from the log-uniform probability distribution) for the post-10,000-year period (DIRS 184614-SNL 2007, all). Then all infiltration rates for that map were scaled such that the four target values for the average infiltration through the repository footprint were obtained to meet the NRC requirement. The resulting percolation fluxes through the repository footprint for the four post-10,000-year period average infiltration rates were, respectively, 21.58, 40.78, 52.07, and 61.86 millimeters per year (0.85, 1.61, 2.05, and 2.44 inches per year).

Table F-1. Average net infiltration rates (millimeters per year)[a] over the unsaturated zone flow and transport model domain for the present-day, monsoon, glacial-transition, and post-10,000-year climate states.

Climate	Percentile			
	10th	30th	50th	90th
Present-day	3.03	7.96	12.28	26.78
Monsoon	6.74	12.89	15.37	73.26
Glacial-transition	11.03	20.45	25.99	46.68
Post-10,000-year	16.89	28.99	34.67	48.84
Weighting factor	61.91	15.68	16.45	5.960

Source: DIRS 184614-SNL 2007, all.
a. To convert millimeters to inches, multiply by 0.03937.

F.2.2.2 Mountain-Scale Unsaturated Zone Model

Water generally moves downward in the rock matrix and in rock fractures. The rock mass at Yucca Mountain consists of volcanic rock with varying degrees of fracturing due to contraction during cooling of the original, nearly molten rock and because of extensive faulting in the area (DIRS 169734-BSC 2004, Section 3.5.8). Water flowing in the fractures moves much more rapidly than water moving through the rock matrix (DIRS 184614-SNL 2007, Section 6.6.2.3). At some locations, water can collect in locally saturated zones (perched water) or can be laterally diverted because of differing rock properties at rock layer interfaces (DIRS 184614-SNL 2007, Section 6.2.2.2).

The mountain-scale unsaturated zone flow model used constant flow during each climate state and generated three-dimensional flow fields for each of the four different infiltration boundary conditions (10th-, 30th-, 50th-, and 90th-percentile values) for each climate state and set of rock properties for each infiltration rate (DIRS 184614-SNL 2007, all). This is an isothermal model; thermal effects can be neglected because flow would be strongly perturbed only by heat near the emplacement drifts and during early times (DIRS 184614-SNL 2007, all). The thermal hydrology models discussed below deal with the influence of heat near the drifts. The flow fields from the mountain-scale unsaturated zone flow model are the abstractions the TSPA-LA model used while the system model was running. The TSPA-LA model simply switched to the flow field for the sampled infiltration rate and climate state.

After the repository cooled, water would return to the repository walls. However, because of a capillary barrier effect at the drift wall, only a small fraction of this returned water would drip into the emplacement drifts. The remaining water would be diverted around the emplacement drifts. The low rate at which water flows through Yucca Mountain, which is in a semiarid area, would restrict the number of seeps and the amount of water available to drip. Drips would occur only if the hydrologic properties of the rock mass caused the water to concentrate enough to feed a seep. Over time, the number and locations of seeps would tend to increase, corresponding to increasing infiltration due to changing climate conditions. The seepage flow model calculated the amount of seepage that could occur based from information from the unsaturated zone flow model (DIRS 181244-SNL 2007, all). The conceptual model for seepage has determined, based on direct field observations, that openings in unsaturated rock act as capillary barriers and divert water around them (DIRS 181244-SNL 2007, all). For seepage to occur in the conceptual model, the rock pores at the drift wall would have to be locally saturated. Drift walls could become locally saturated by either disturbance to the flow field caused by the drift opening or variability in the permeability field that created channeled flow and local ponding. Of these two potential causes, the variability effect is more important. Drift-scale flow calculations made with uniform hydrologic properties suggested that seepage would not occur at expected percolation fluxes. However, calculations that included permeability variations do estimate seepage, with the amount dependent on the hydrologic properties and the incoming percolation flux (DIRS 181244-SNL 2007, all). DOE based the seepage abstraction on extensive modeling calibrated by measurements from tests in the Exploratory Studies Facility (DIRS 181244-SNL 2007, all). The seepage abstraction included probability distributions for the fraction of waste packages that could encounter seepage and the seep flow rate; it accounted for parameter uncertainty, spatial variability, and other effects such as focusing, episodicity, rock bolts, drift degradation, and coupled processes (DIRS 181244-SNL 2007, all). All of these parameters were input as uncertainty distributions and sampled in the probabilistic TSPA-LA simulations.

F.2.3 ENGINEERED BARRIER SYSTEM ENVIRONMENTS

Engineered Barrier System environments refer to the thermal-hydrologic and chemical environments in the emplacement drifts. These environments would control processes that affect the engineered components of the system (such as the drip shields, waste packages, and waste forms). The environmental characteristics of importance are the degradation of the drift (which would include rockfall into the drift from seismic ground motion), temperature, relative humidity, liquid saturation, pH, liquid composition, and gas composition. Thermal effects on flow and chemistry outside the drifts would be important because they would affect the amount and composition of water and gas that entered the drifts. The Engineered Barrier System environments would be important to postclosure repository performance because they would help determine degradation rates of waste packages, degradation of waste forms in breached waste packages, quantities and species of mobilized radionuclides, transport of radionuclides from breached waste packages through the drift into the unsaturated zone, and movement of seepage water through the drift into the unsaturated zone.

Emplacement drifts could degrade with time as a result of seismic ground motion. These effects could lead to partial or complete drift collapse, with rock material filling the enlarged drifts and changing their shape and size. These effects could alter the thermal hydrology in the drifts and damage the engineered barriers. Depending on the intensity of these effects, impacts to thermal hydrology and damage to the engineered barriers and drifts could be small with local rockfall from the ceiling of otherwise intact drift openings or, in extreme cases, could result in substantial impacts to thermal hydrology and damage to the engineered barriers and partial or complete drift collapse, with rubble rock material filling the enlarged drifts (DIRS 176828-SNL 2007, all).

The TSPA-LA model performed most engineered system calculations for a limited number of waste package locations. In the model, each of these locations is representative of a group of waste packages with similar environmental characteristics. The model calculated radionuclide releases, for example, for representative codisposal and commercial spent nuclear fuel waste packages in each group and then scaled up by the number of failed waste packages of each type in each group. The waste package groups (referred to as percolation subregions) are not based on physical location but rather on percolation-flux patterns (that is, divided into categories of specific ranges of percolation flux) (DIRS 184433-SNL 2008, all). The analysis defined five percolation subregions according to percolation-flux distributions.

The heat generated by the decay of nuclear materials in the proposed repository would cause the temperature of the surrounding rock and waste packages to rise from the time of emplacement until a few hundred years after repository closure (DIRS 184433-SNL 2008, all). The water and gas in the heated rock, referred to in this Repository SEIS as the thermal pulse, would be driven away from the repository during this period. The thermal output of the materials would decrease with time; eventually, the rock would return to its original temperature, and the water and gas would flow back toward the repository. DOE used the multiscale thermal hydrology model to study the processes that would govern the temperature, relative humidity, liquid saturation, liquid flow rate, liquid evaporation rate, and thermal effects on seepage. Drift-scale modeling included coupling of drift-scale processes with mountain-scale processes to account for effects such as faster cooling of waste packages near the edge of the repository in comparison with packages near the center. DOE developed a multiscale modeling and abstraction method to couple drift-scale processes with mountain-scale processes (DIRS 184433-SNL 2008, all). The analysis abstracted the results of detailed thermal-hydrologic modeling studies as response surfaces of temperature, humidity, and liquid saturation.

The source term for transport of radionuclides from the proposed repository through the unsaturated zone and saturated zone would be the radionuclide flux from inside the drifts to the unsaturated zone rock. The in-drift Engineered Barrier System chemical environment would influence that flux. DOE used the physical and chemical environment model (DIRS 177412-SNL 2007, all) to study the changing composition of gas, water, colloids, and solids in the emplacement drifts under the perturbed conditions of the repository. The analysis integrated several models to provide detailed results and interpretations. The thermal loading of the system would cause the major composition changes. Emplaced materials could be an additional source of colloids that could affect the transport of radionuclides in the aqueous system. The Engineered Barrier System chemical environment models produced detailed results that DOE abstracted for the following key processes:

- Chemistry of seepage water flowing into the drift. The composition of water that entered the repository drifts would have a primary influence on the types of brines that could form as evaporation occurred in the drifts. The composition of that water is closely coupled with the thermal-hydrologic processes in the host rock near the drifts. During the thermal period, water would boil and evaporate. Vapor would move away from the heated drifts, while condensed liquid water would simultaneously percolate down and replace the evaporated water. This process, which is referred to as reflux, would continue as long as the host rock was hot enough to support it. Percolating reflux waters would contain dissolved chemical species such as sodium, chlorides, calcium, and carbonates. If evaporation occurred, dissolved chemical species would precipitate as minerals and salts. After the primary thermal period passes, and after soluble precipitates and salts redissolved, the composition of seepage water that entered the drifts would approximate the composition of the preemplacement ambient percolation in the host rock.

- Composition of the gas phase in the emplacement drifts. The gas composition would influence the evolution of the chemical environment in the drifts. The gas composition would initially be similar to the composition of atmospheric air. However, during the thermal period, reactive components (oxygen and carbon dioxide) of the gas phase would be diluted by steam and strongly modified by water evaporation and interaction with carbon dioxide in water and carbonate minerals. One important aspect that would affect the system would be the exsolution of carbon dioxide from the liquid phase as the temperature rose. This exsolution in the boiling zone in the rock would result in a localized increase in pH, which would decrease in the condensation zone where the vapor (enriched in carbon dioxide) was transported and condensed.

- Evolution of the chemical environment in the Engineered Barrier System. Seepage waters would enter drifts, either by dripping from the drift crown or by imbibition (the absorption of fluid by a solid body without resultant chemical change in either) into the invert. Once in the drifts, the chemical compositions of the seepage waters could change due to evaporation, mineral precipitation, or both. The composition of seepage water in the emplacement drift would change according to the sequence of minerals that precipitated from that solution as a function of the composition of seepage water in the drift, thermal conditions, relative humidity, and gas composition during evaporation. The chemistry of the water in the drift would affect the mobility of radionuclides in the Engineered Barrier System and the likelihood of initiation of localized corrosion if this water contacted waste packages.

DOE developed abstractions for the above chemical processes (DIRS 177412-SNL 2007, all) and integrated them in the TSPA-LA model as chemistry look-up tables.

Drift seepage is the flow of liquid water into emplacement drifts. Water that seeped into drifts could contact waste packages, mobilize radionuclides, and result in advective transport of radionuclides through waste packages breached by general corrosion and localized corrosion processes. The unsaturated rock layers that overlie and host the repository would form a natural barrier that reduced the amount of water that entered drifts by natural subsurface processes. For example, the capillary barrier would limit drift seepage at the drift crown (roof), which would decrease or even eliminate water flow from the unsaturated fractured rock into the drift. During the first few hundred years after waste emplacement, when above-boiling rock temperatures would develop from the decay heat of the radioactive waste, vaporization of percolation water would further limit seepage. Estimating the effectiveness of these natural barrier capabilities and the amount of seepage into drifts is an important aspect of assessing the performance of the repository. The TSPA-LA seepage abstraction model is based on a synthesis of detailed modeling studies (DIRS 181244-SNL 2007, all) and field testing (DIRS 177394-SNL 2007, all) that DOE abstracted as look-up tables for seepage into nondegraded and collapsed drifts as a function of capillary strength and tangential permeability of the fracture network near the drift wall.

Condensation water that dripped from drift walls would be another potential source of seepage water in the drift. The source of condensation water would be the invert and the drift wall. Natural convection would transport water vapor axially from hotter to cooler regions where the vapor could condense. The axial movement of the water vapor, the saturated vapor pressure at the drift wall and invert surface, and the change in temperature along the drifts would be the main factors that would drive the occurrence of condensation (DIRS 181648-SNL 2007, all).

Evaporation and mixing with condensation water and circulating gas, particularly during the thermal pulse, would strongly influence the chemistry of seepage water when it entered the drift. At later times, as the thermal pulse dissipated and condensation fluxes decreased, the chemistry of the seepage water would not change substantially from that when the water entered the drift.

The primary water input to the Engineered Barrier System would be the total flow rate from two sources: (1) the seepage volumetric flow rate into the drifts from the drift seepage abstraction model and (2) the condensation volumetric flow rate on the drift walls from the in-drift natural convection and condensation model. A secondary source of inflow to the Engineered Barrier System would be imbibition into the invert from the surrounding unsaturated rock matrix, from the *Multiscale Thermohydrologic Model* (DIRS 184433-SNL 2008, all).

The flow of water through the Engineered Barrier System could have eight pathways (DIRS 177407-SNL 2007, all):

- Seepage and drift wall condensation. This would be the water inflow from the crown of the drift. It would include drift seepage and any condensation on the section of the drift wall above the drip shield.

- Flow through the drip shields. DOE based the flow rate through the drip shields on the presence of breaches due to general corrosion (DIRS 180778-SNL 2007, all) or possible displacement of drip shields due to a seismic event (DIRS 176828-SNL 2007, all).

- Diversion around the drip shields. The portion of the dripping water that did not flow through the drip shield would flow directly to the invert.

- Flow through the waste packages. Three general types of openings in the waste packages could exist due to corrosion: (1) stress corrosion cracks from residual stress or seismic ground motion, (2) breaches from general corrosion, and (3) breaches from localized corrosion. DOE based the flow rate through the waste packages on the presence of breaches due to general and localized corrosion. Stress corrosion cracking could occur, but the analysis did not include the advective flow of water through stress corrosion cracks because (1) capillary behavior would allow water to reside indefinitely in the crack without flow; (2) surface tension would oppose hydraulic pressure at the outlet; and (3) stress corrosion cracks would be tight, rough, and tortuous, which would limit the transient response to dripping water (DIRS 177407-SNL 2007, all).

- Diversion around the waste package. The portion of the dripping water that did not flow into the waste packages would bypass the waste forms and flow directly to the invert.

- Flow into the invert. DOE has modeled all water flow from the waste packages as flowing into the invert, independent of the location of a breach on the waste package. In addition, the dripping water that diverted around the drip shields and waste packages would flow into the invert. The analysis did not include the presence of the emplacement pallets in the abstraction of Engineered Barrier System flow, so the water flow was modeled without resistance from the pallets.

- Imbibition flow to the invert. Water could be imbibed from the host rock matrix into the invert. The Engineered Barrier System thermal-hydrologic environment submodel provides the rate of water imbibition into the invert.
- Flow from the invert to the unsaturated zone. A portion of the advective flux from the invert equal to the total dripping flux would flow directly into unsaturated zone fractures. The portion of the advective flux from the invert equal to the imbibition flux to the invert would flow into the unsaturated zone matrix.

These pathways are time-dependent in the sense that waste package breaches would vary with time and local conditions in the repository. The analysis did not include the effect of evaporation on seepage water flow through the Engineered Barrier System, which would tend to overestimate Engineered Barrier System flow.

F.2.4 WASTE PACKAGE AND DRIP SHIELD DEGRADATION

A two-layer waste package would enclose the radioactive waste that DOE emplaced in the proposed repository. The layers would be of two different materials that would fail at different rates and from different mechanisms as they were exposed to repository conditions. The outer layer would be a high-nickel alloy (Alloy 22) and the inner layer would be a stainless-steel alloy. In addition, commercial spent nuclear fuel waste packages would contain a stainless-steel transportation, aging, and disposal (TAD) canister. It should be noted that the TSPA-LA model is conservative in that it does not take credit for corrosion of the inner layer of the waste package nor for the TAD canister, which would limit water influx after the outer layer of the waste package was breached.

To divert dripping water from the waste package and thereby extend waste package life, DOE would place a Titanium Grade 7 drip shield over the waste packages just before repository closure. The drip shield would divert water that entered the drift from above and thereby prevent seep water from contact with the waste package. The analysis used the drip shield and waste package degradation models to simulate the degradation of these components (DIRS 180778-SNL 2007, all; DIRS 178519-SNL 2007, all). General corrosion was the only drip shield degradation mechanism DOE considered under nominal conditions because analyses showed that if other degradation mechanisms (stress corrosion cracking, localized corrosion, and microbially influenced corrosion) occurred the consequences to drip shield performance would be insignificant (DIRS 180778-SNL 2007, Section 6.10).

Three main types of waste package degradation were considered under nominal conditions—general corrosion, stress corrosion cracking, and seepage-induced localized corrosion. An additional corrosion process—microbially influenced corrosion—was considered to provide enhanced general corrosion on the waste package. The analysis screened out mechanical failure of the drip shield and waste package by rockfall under nominal conditions due to low consequence. However, it included mechanical failure of the drip shield and waste package by rockfall and fault displacement in the Seismic Scenario Class. Failure mechanisms that the analysis considered included collapse of the drip shield, stress corrosion cracking of the waste package, and rupture of the drip shield and waste package.

For nominal degradation processes, output from the drip shield and waste package degradation models included time-dependent quantitative assessments of drip shield and waste package degradation and failure. Results included the time to failure by general corrosion for the drip shield and the time to initial failure by general corrosion for the waste package, time to first breach of the waste package by stress corrosion crack failure, and the degree of drip shield and waste package failure as a function of time. In this Repository SEIS, drip shield failure by general corrosion would occur between approximately 270,000 years and 340,000 years, with the failure time different for each epistemic realization (DIRS 183478-SNL 2008, Figures 7.7.3-2[a] and 8.1-4 and Section 8.2.1). In addition, because there was no spatial variability in drip shield corrosion rates, all drip shields in the repository would fail at the same time in a given realization. The time of the first breach of the waste package would correspond to the start of waste form degradation in the breached package. The time of first breach ranged from approximately 170,000 years to beyond 1 million years, with the breaches caused by stress corrosion cracking in the weld of the outer closure lid (DIRS 183478-SNL 2008, Figures 8.3-5[a] and 8.3-6[a]). General corrosion failures would start at around 400,000 years and about 9 percent of the waste packages would experience a general corrosion breach within 1 million years (DIRS 183478-SNL 2008, Figure 8.3-6[a]). Diffusion would be the only transport mechanism acting to release radionuclides from a waste package when cracks were the only penetration through the waste package. The diffusive area for a single stress corrosion crack based on the geometry of an ellipsoidal crack would be 6.7×10^{-6} square meters (7.2×10^{-5} square feet) (DIRS 183478-SNL 2008, Section 8.3.3.2.1[a]). On average, approximately 60 percent of the commercial spent nuclear fuel waste packages and 54 percent of the codisposal waste packages would experience a first breach by stress corrosion cracking by 1 million years (DIRS 183478-SNL 2008, Figure 8.3-6[a]). The average number of cracks per breached waste package at 1 million years would be about five (DIRS 183478-SNL 2008, Section 8.3.3.2.1[a]). Advection and diffusion would be the transport mechanisms acting to release radionuclides from a waste package when general corrosion breaches formed. On average, only about 9 percent of the commercial spent nuclear fuel and codisposal waste packages would experience a general corrosion breach within 1 million years (DIRS 183478-SNL 2008, Figure 8.1.3-6[a]). The average number of general corrosion breaches at 1 million years would be about four (DIRS 183478-SNL 2008, Section 8.3.3.2.1[a]). General corrosion breaches were represented by dividing the waste package surface into subareas called patches. The total number of possible patches on a commercial spent nuclear fuel waste package would be about 1,430 and on a codisposal waste package about 1,410 (DIRS 183478-SNL 2008, Section 6.3.5.1.2).

Manufacturing and material defects could augment corrosion processes and result in early failure of the drip shield and waste package. Early failure is defined as through-wall penetration of a drip shield or waste package at a time earlier than would occur by mechanistic degradation for a defect-free drip shield or waste package. Several types of manufacturing defects (for example, base-metal flaws, improper weld filler material, improper base-metal selection, improper heat treatment, improper handling, and improper stress relief) could lead to early drip shield and waste package failure. Among these defects DOE anticipates that improper heat treatment would occur most often (DIRS 178765-SNL 2007, Table 6-8).

An analysis of manufacturing and testing led to probability distributions for the number of drip shields and waste packages that could fail due to manufacturing and material defects. Table F-2 lists the resultant early failure unconditional probability values. The probability values in this table indicate that more than 44 percent of the TSPA-LA realizations would have early failed waste packages and 56 percent would have no early failed waste packages. Twenty-two percent of the realizations would have only one early failure and 9.6 percent would have two early failed waste packages. This leaves 12 percent of the

Table F-2. Early failure unconditional probability values.

n (number of early failures)	Probability of n failures of waste packages	Probability of n failures of drip shields
0	0.558	0.9834
1	0.2237	0.0155
2	0.0955	0.0009
≥ 3	0.1228	0.0002

Source: DIRS 178765-SNL 2007, all.

remaining realizations with three or more failed waste packages. The expected number of early failed waste packages would be 1.09 (DIRS 178765-SNL 2007, all). Only 1.7 percent of the realizations would have early failed drip shields, 98 percent would have no early failed drip shields. Realizations with only one early failure would account for 1.6 percent and 0.09 percent would have two early failed drip shields. This leaves 0.02 percent of the remaining realizations with three or more failed drip shields. Because only a small number of realizations would have an early failed drip shield, the expected number of early failed drip shields would be 0.018 (DIRS 178765-SNL 2007, all).

It was conservatively assumed in the TSPA-LA that manufacturing or material defects resulted in complete failure. This representation of early drip shield and waste package failures reflects a conservative view because a manufacturing or material defect would not necessarily result in complete failure. The analysis also assumed that a waste package under an early failed drip shield would fail completely due to localized corrosion; this is conservative because a smaller failure would produce smaller releases.

ASSUMPTIONS

In the assessment of postclosure impacts, DOE sometimes used assumptions to formulate models. An assumption is a premise about some element of the modeling and usually something for which there is no absolute proof. Assumptions normally account for qualitative uncertainties (if an absolute probability cannot be assigned). Assumptions are used: (1) when there is a high certainty (although unquantified) that the premise is true, and (2) when the assumption is conservative (that is, all alternative assumptions would lead to a smaller impact). The conservative assumption is often used if there is considerable uncertainty about which alternative premise is more likely. Regulations that prescribe modeling make some assumptions necessary. A set of assumptions defines the conceptual model for the analysis. A set of alternative assumptions would represent an alternative model. Some sensitivity studies compare alternative models to help define the importance of certain assumptions, especially if there is considerable uncertainty (Chapter 5, Section 5.3.4.2.3).

Each assumption has a basis, which is the reason the assumption represents a condition of high certainty, a statement that it is mandated by a regulation, or a statement that it is conservative in relation to the outcome of impact analysis.

F.2.5 WASTE FORM DEGRADATION

The waste form degradation models evaluate the interrelationships of the in-package water chemistry, the degradation of the waste forms, and the mobilization of radionuclides (DIRS 177423-SNL 2007, all; DIRS 177418-SNL 2007, all; DIRS 180472-SNL 2007, all). The model consists of components that:

- Define the radioisotope inventories for representative commercial spent nuclear fuel and codisposal waste packages (this is the inventory abstraction that Section F.3.1 discusses in more detail).

- Evaluate in-package water chemistry. In-package chemistry is modeled in the TSPA-LA using simplified expressions to define the bulk chemistry, which consists of pH, ionic strength, and total carbonate concentration as a function of time inside a waste package. The analysis used chemistry outputs to set conditions for waste form degradation and to determine dissolved concentration limits in the waste package.

- Evaluate the matrix degradation rates for commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste forms. The TSPA-LA model used empirical degradation rate formulas DOE developed for the three different waste forms to model degradation. DOE would combine its spent nuclear fuel and vitrified high-level radioactive waste in codisposal waste packages.

- Evaluate the dissolved radionuclide concentration limits for aqueous phases. Dissolved radionuclide concentration limits abstraction (distributions of solubilities as a function of pH and temperature in the waste package; solubilities are checked for possible limitations due to waste form degradation rate or package inventory).

- Evaluate sorption of radionuclides in the waste package.

- Evaluate the waste form colloidal phases. The colloidal radionuclide concentration component abstraction models the formation, stability, and concentration of radionuclide-bearing colloids in the waste package and Engineered Barrier System, as well as reversible and irreversible sorption of dissolved radionuclides, using empirical relationships and uncertainty distributions for sorption coefficients.

F.2.6 ENGINEERED BARRIER SYSTEM FLOW AND TRANSPORT

The waste form would be the source of radionuclides in the Engineered Barrier System. After a waste package failed (due to general or localized corrosion, rupture due to large seismic ground motions or fault displacements, igneous intrusion, or early waste package failure mechanisms), a portion of the water that seeped into the drift could enter the waste package if the drip shield had also failed, which would mobilize radionuclides from the degraded waste form and transport them by advection into the unsaturated zone. Diffusion would be the primary transport mechanism when the water flux into the waste package was negligibly small or zero, as in the case where the waste package has failed due to stress corrosion cracking. If stress corrosion cracks were the only penetrations through the drip shield and waste package, no advective transport could occur through them (DIRS 177407-SNL 2007, all). Diffusive transport would occur as a result of a gradient in radionuclide concentration and could occur at the same time as advective transport.

The abstraction simulates the following transport modes:

- Advective and diffusive transport of dissolved radionuclides in the waste package and invert to account for the dependence of diffusion on porosity, saturation, and temperature;

- Colloid-facilitated advective and diffusive transport in the waste package and invert;

- The time-dependent quantity of corrosion products inside a breached waste package;
- Radionuclide sorption onto stationary corrosion products in a breached waste package, which includes competition for a finite number of sorption sites and equilibrium and kinetic sorption-desorption processes; and
- Equilibrium linear radionuclide sorption in the invert.

The TSPA-LA model represents diffusion with the use of a diffusion transport equation with an empirical effective diffusivity that is a function of liquid saturation, porosity, and temperature. The analysis used sorption response surfaces based on detailed surface complexation modeling to implement the model for sorption of radionuclides on stationary corrosion products in the waste package.

A linear isotherm (constant ratio of concentration in the water to amount sorbed on the solid) would characterize sorption on invert ballast material. Advective transport is represented by a liquid transport equation with the velocity from the Engineered Barrier System flow abstraction.

F.2.7 UNSATURATED ZONE TRANSPORT

Unsaturated zone transport refers to the movement of radionuclides from the Engineered Barrier System of the proposed repository, through the unsaturated zone, and to the water table. The unsaturated zone would be the first component of the Lower Natural Barrier to radionuclides that escaped from the repository. It would act as a barrier by delaying radionuclide movement. If the delay was long enough for significant decay of a specific radionuclide, the unsaturated zone could have a significant effect on the ultimate dose from releases of that radionuclide to the environment. *Particle Tracking Model and Abstraction of Transport Processes* (DIRS 184748-SNL 2008, all) describes how radionuclides would move through the unsaturated zone. The unsaturated zone model considered transport through welded and nonwelded tuff and flow through the fractures and the rock matrix. In addition, the model accounted for the existence of zeolitic alterations of the tuff in some regions. The zeolitic tuffs have the characteristics of lower permeability and enhanced radionuclide sorption. The unsaturated zone water flow would provide the background on which the unsaturated zone transport took place. The model used the flow fields from the unsaturated zone flow model (Section F.2.2). Radionuclides can migrate in groundwater as dissolved molecular species or in colloids. Dissolved species would typically consist of radionuclide ions complexed with various groundwater species, but still at molecular size. Colloids are particles of solids, typically clays, silica fragments, or organics, such as humic acids or bacteria, that are larger than molecular size, but small enough to remain suspended in groundwater for indefinite periods. Colloids usually have a size range between a nanometer and a micrometer. A radionuclide could be attached to the surface or bound in the structure of the colloid.

Five basic processes affect the movement of dissolved or colloidal radionuclides:

- Water flux and advection. The ability of the unsaturated zone to prevent or substantially reduce the rate of movement of radionuclides depends in part on the flux of water through the unsaturated zone. This flux is distributed between faults, fractures, and the matrix of the host rock and other units in the unsaturated zone. The rate of movement or advection of radionuclides is strongly dependent on the degree of fracture flow, which, in turn, is dependent on the magnitude of the total flux. Total flux is directly dependent on the surficial recharge and infiltration that, in turn, is dependent on climatic conditions. The increase in recharge due to change in climate states could significantly reduce the capability of the unsaturated zone to reduce the rate of radionuclide advection. This reduction would be a function of (1) the increase in fracture flux and corresponding reduction in the effectiveness of matrix diffusion and (2) the rise in the water table and the associated decrease in the unsaturated zone travel distance.

- Matrix diffusion. Matrix diffusion results in the diffusion of dissolved radionuclides from the fractures into the matrix of the rock. Because advective transport is significantly slower in the matrix than in the fractures, matrix diffusion can be a very efficient retarding mechanism, especially for moderately to strongly sorbed radionuclides, due to the increase in rock surface accessible to sorption. Matrix diffusion is incorporated in the unsaturated zone radionuclide transport abstraction model in the TSPA-LA model. However, matrix diffusion of colloidally transported radionuclides has been excluded for conservatism.

- Sorption. Radionuclides released from the repository would have varying retardation characteristics. Several radionuclides that would be the dominant contributors to the total dose would be significantly retarded in the unsaturated zone if there was significant matrix diffusion or matrix-dominated flow in the vitric Calico Hills Tuff. These would include strontium-90, cesium-137, plutonium-239 and -240, and americium-241 and -243. The sorption of these radionuclides that were transported in the matrix of the vitric tuff would prevent their movement or significantly reduce the rate of movement from the repository to the accessible environment (DIRS 184748-SNL 2008, Figures 6.6.2-5[b], D.2-1[b], D.2-2[b], D.2-3[b], and D.2-6[b]).

- Colloidal transport. Several radionuclides could move in colloidal particles in the unsaturated zone. These include plutonium-239 and -240 and americium-241 and -243 (DIRS 184748-SNL 2008, Section 6.4.5). The analysis considered reversible and irreversible colloidal transport. Retardation of a large fraction of the colloidally transported radionuclides would be sufficient to prevent the movement or significantly reduce the rate of movement of the more rapidly decaying of these radionuclides from the repository to the accessible environment (DIRS 184748-SNL 2008, Figure 6.6.2-6[b]). The analysis conservatively assumed that a small fraction of the colloids would be unretarded in the unsaturated zone (DIRS 184748-SNL 2008, Section 6.5.13). The unsaturated zone transport model includes retardation of colloids in fractures during reversible and irreversible colloid transport and size exclusion and fracture-rock matrix interfaces and filtration at rock matrix unit boundaries for irreversible colloid transport (DIRS 184748-SNL 2008, Section 6.4.5).

- Radioactive decay and ingrowth. As radionuclides moved along groundwater flow paths from the repository to the accessible environment, they would decay. The degree of decay would be a function of the half-life of the radionuclide in comparison with the transport time to the environment. In addition, the analysis considered the ingrowth of some radionuclides (in particular, neptunium-237 from the decay of americium-241). This included decay and ingrowth processes for dissolved and colloidal radionuclides.

The analysis implemented the unsaturated zone transport model in the TSPA-LA model as an embedded computer program that simulates the three-dimensional transport with a residence-time, transfer-function, particle-tracking technique. The model, which incorporates the unsaturated zone flow fields, is based on a dual-continuum formulation, which accounts for the effects of fracture flow and fracture-matrix interactions on radionuclide transport. The model includes future changes in water table elevations, which shorten the path length for unsaturated zone transport, and implements those as instantaneous changes that occur with climate change. The key parameters such as sorption coefficients, fracture frequency, fracture porosity, and colloid parameters (partitioning, retardation, colloid size distribution) were input as uncertainty distributions. The unsaturated zone radionuclide transport provides the rate and spatial distribution of radionuclide releases to the saturated zone flow and transport model as output.

F.2.8 SATURATED ZONE FLOW AND TRANSPORT

The saturated zone at Yucca Mountain is the region beneath the ground surface where rock pores and fractures are fully saturated with groundwater. The upper boundary of the saturated zone is the water table. The proposed repository would be in the unsaturated zone approximately 300 meters (1,000 feet) above the water table.

Underground water flows down hydraulic gradients. Based on water-level observations in area wells, groundwater near Yucca Mountain flows generally in a north-to-south direction (DIRS 177391-SNL 2007, Section 6.3.1.3). The major purpose of the *Saturated Zone Flow and Transport Model Abstraction* (DIRS 183750-SNL 2008, all) is to evaluate the migration of radionuclides from their introduction at the water table below the proposed repository to the point of release to the biosphere. A radionuclide could move through the saturated zone as a dissolved solute or a colloid. The input to the saturated zone is the spatial and temporal distribution of mass flux of radionuclides from the unsaturated zone. The output of the saturated zone flow and transport model is a mass flow rate of radionuclides in the water that a hypothetical farming community would use.

F.2.8.1 Saturated Zone Flow

The *Saturated Zone Site-Scale Flow Model* (DIRS 177391-SNL 2007, all) receives inputs from the unsaturated zone flow model and produces outputs in the form of flow fields. The saturated zone flow model incorporates a significant amount of geologic and hydrologic data from drill holes near Yucca Mountain. The saturated groundwater flow in the vicinity of Yucca Mountain can be estimated by knowing the porosity of the flow media, the hydraulic conductivity, and the recharge of water into the flow media. Water flow in the saturated zone occurs through two rock types—fractured volcanic rocks and alluvium (DIRS 177391-SNL 2007, Section 1). The groundwater flow rates, the rate of transport of radionuclides, and the radionuclide retardation characteristics of these different rock types are significantly different (DIRS 183750-SNL 2008, Section 6.5.2.2). In addition to the differences in flow and transport characteristics of the different lithologic units in the saturated zone, the presence of discrete flow features in the fractured tuff units would affect the rate of movement of radionuclides to the accessible environment. Matrix flow in the alluvium would provide a significant reduction in the movement of radionuclides to the environment. The primary tool used to describe saturated zone flow is a numerical model in three dimensions. DOE developed the three-dimensional saturated zone flow model specifically to determine the groundwater flow field at Yucca Mountain. The model produced a library of flow fields (maps of groundwater fluxes) that the saturated zone transport model used.

F.2.8.2 Saturated Zone Transport

The saturated zone transport model (DIRS 184806-SNL 2008, all) receives inputs in the form of radionuclide mass fluxes from the unsaturated zone transport model and produces outputs in the form of radionuclide mass fluxes to the biosphere model. It incorporates laboratory and field data from a variety of sources.

Radionuclides that were released from a repository at Yucca Mountain to the groundwater would enter the saturated zone beneath the repository and travel southeast and then south toward the Amargosa Desert (DIRS 184806-SNL 2008, Section 8.1.2, Figure 6.5-2). The groundwater could transport radionuclides in two forms: as dissolved species or bound in colloids. Advection would be the principal transport mechanism for dissolved and colloidal radionuclides in the saturated zone. The advective flux would depend on the hydrogeologic characteristics of the water-conducting features in the saturated zone and on the groundwater flux through these features. Dispersive processes would tend to spread transient radionuclide pulses that could move to the saturated zone (for example, following a water table rise due to climate changes).

The analysis primarily used a three-dimensional, particle-tracking model for transport through the saturated zone (DIRS 183750-SNL 2008, all). This model generated a library of breakthrough curves—distributions of transport times—along with a time-varying source term from the unsaturated zone, to calculate the releases at the boundary between the geosphere and biosphere. The model accounted for the flow of groundwater and its interaction with media along the flow path. In the volcanic rocks that comprise the saturated media in the immediate vicinity of Yucca Mountain, groundwater flows primarily through fractures, while a large volume of water is relatively immobile in the surrounding rock matrix (DIRS 184806-SNL 2008, Section 6.3). Radionuclides would travel with the moving fracture water but, if dissolved, could diffuse between the matrix water and fracture water. This transfer between fracture and matrix water is characteristic of a dual-porosity system. The saturated zone transport model is a dual-porosity model. The media at greater distances from Yucca Mountain are alluvial gravels, sands, and silts (DIRS 184806-SNL 2008, Section 6.3). The model simulated these areas as more uniformly porous.

Because the three-dimensional particle-tracking model does not consider ingrowth from decay chains, it is used to evaluate only the first and second members of decay chains. The influence of a decaying parent species on the second member of a decay chain is approximated with the use of an inventory-boosting method in which release of the parent species from the unsaturated zone is predecayed and added to the decay species source term from the unsaturated zone model. A one-dimensional saturated zone model accounts for decay and ingrowth of all other members of a decay chain during transport. This model was incorporated directly in the GoldSim model as a series of pipes. The advantage of using the one-dimensional model is that the radionuclide masses can be accounted for directly. The disadvantage is that the flow and transport geometry is necessarily simplified.

F.2.9 BIOSPHERE

If the radionuclides were removed from the saturated zone in water pumped from wells, the radioactive material could result in dose to humans in several ways. For example, water could be used to irrigate crops that would be consumed by humans or livestock, to water stock animals that would be consumed by humans as dairy or meat products, or to provide drinking water for humans. In addition, if the water from irrigation wells evaporated on the surface, the radionuclides could be left as fine particulate matter that could be picked up by the wind and inhaled by humans. The biosphere model (DIRS 177399-SNL 2007, all) tracks the environmental transport of radionuclides through the biosphere and calculates annual radiation exposure to a person who lived in the general vicinity of the proposed repository if there was a release of radioactive material to the biosphere after closure. The primary outputs of the biosphere model are sets of biosphere dose conversion factors equivalent to the annual dose from all potential exposure pathways that the person would receive as a result of a unit concentration of a radionuclide in groundwater or volcanic ash (DIRS 177399-SNL 2007, all). The biosphere scenarios assumed a reference person who lived in the Amargosa Valley region at various distances from the repository. People who lived in the town of Amargosa Valley would be the group most likely to be affected by radioactive releases, specifically an adult who lived year-round at this location, used a well as the primary water source, and otherwise had habits similar to those of the inhabitants of the region (such as the consumption of local foods). Because changes in human activities over millennia are unpredictable, the analysis assumed that the present-day reference person was the basis for future inhabitants. The EPA standard at 40 CFR Part 197 provides the definition for the reference person as the reasonably maximally exposed individual (RMEI).

DOE did not use the biosphere model to evaluate the chemically toxic materials because there are no usable comparison values for radiological and nonradiological doses. Rather, the Department made a separate analysis of concentrations of these materials that compared the concentrations to available regulatory standards, such as the maximum contaminant level goal, if available, or the appropriate oral reference dose.

The biosphere is the last component in the chain of TSPA-LA model subsystem components. There are two connections between the biosphere model and other TSPA models. One is for the scenario classes and modeling cases that involve exposure through the groundwater pathway (Nominal, Drip Shield and Waste Package Early Failure, Seismic Ground Motion Damage and Fault Displacement, and Igneous Intrusion), where the biosphere is coupled with the saturated zone flow and transport model; the other is for the Volcanic Eruption Modeling Case, where the biosphere is coupled with the volcanic eruption model. For the Human Intrusion Scenario, the biosphere model is coupled with the saturated zone flow and transport model.

F.2.10 IGNEOUS ACTIVITY DISRUPTIVE EVENTS

Igneous activity could compromise the natural and engineered barriers in the proposed repository. The TSPA-LA model represents igneous activity with the Igneous Scenario Class, which includes features, events, and processes that describe the possibility that low-probability igneous activity could affect repository performance. Two modeling cases in the TSPA-LA simulate the significant features, events, and processes: (1) the Igneous Intrusion Modeling Case, which addresses the possibility that magma (molten rock), in the form of a dike (ridge of material), could intrude into the repository and disrupt expected repository performance; and (2) the Volcanic Eruption Modeling Case, which includes features, events, and processes that describe an eruption that would rise through the repository footprint and damage a number of waste packages. The low-probability volcanic eruption could disperse volcanic tephra (solid material of all sizes explosively ejected from a volcano into the atmosphere) and entrained waste into the atmosphere and deposit it on the surface where soil and near-surface geomorphic (of or relating to the form or surface features of the Earth) processes would redistribute it.

The intrusion of a dike or eruption of volcanic material through the repository would not substantially affect the capability of the natural barriers at Yucca Mountain to prevent or reduce the flow of water or the movement of radionuclides in groundwater away from the repository. Movement of radionuclides entrained in magma (rather than contained in groundwater) through the natural system during a volcanic eruption would have some adverse effect on the ability of the natural barrier system to prevent a release of radionuclides. Igneous or volcanic events could adversely affect the Engineered Barrier System's ability to prevent or reduce the release of radionuclides to the natural system.

If igneous activity occurred at Yucca Mountain, possible effects on the repository could fall into three areas:

- Igneous activity that would not directly intersect the repository (no effect on dose from the repository);
- Volcanic eruptions in the repository that would result in the entrainment of waste material in the volcanic magma or pyroclastic material and would bring waste to the surface (which would result in atmospheric transport of volcanic ash contaminated with radionuclides and subsequent human exposure downwind); and
- An igneous intrusion that intersected the repository (no eruption but damage to waste packages from exposure to the igneous material that would enhance release to the groundwater and, thus, transport to the biosphere).

Field geologic investigations, laboratory analyses, analogue studies, and reviews of published literature provide the technical basis for the description of past igneous activity in the Yucca Mountain region and for the development of the conceptual, process, and consequence models that represent potential future events. The process models have been used to develop simplified models or abstractions that are incorporated in the TSPA-LA model to generate a probabilistic representation of the likelihood and consequences of the Igneous Scenario Class.

DOE addressed the probability of a future igneous event that intersected the repository through a probabilistic volcanic hazard analysis that used expert judgment to consider applicable geologic processes and uncertainty. Probability distributions were developed to define the likelihood of a volcanic event and the length and orientation of dikes that could intersect the repository footprint. Information from the probabilistic volcanic hazard analysis was used to estimate the number of eruptive centers in the footprint. The mean annual frequency of intersection of the repository footprint by a potential future igneous event would be 1.7×10^{-8}, which is equivalent to an annual probability of about 1 in 60 million. The 5th- and 95th-percentile uncertainties associated with the frequency of intersection span almost 2 orders of magnitude, from 7.4×10^{-10} magnitude to 5.5×10^{-8} (DIRS 169989-BSC 2004, Table 7-1), or about 1 in 1.4 billion to 1 in 18 million per year. The results of the probabilistic volcanic hazard analyses indicate that the mean annual probability of future igneous activity at Yucca Mountain would be greater than 1×10^{-8}; therefore, the Igneous Scenario Class for disruptive events would be an unlikely event that could affect repository performance.

F.2.10.1 Igneous Intrusion Modeling Case

The Igneous Intrusion Modeling Case simulates flow and transport through the Engineered Barrier System and the unsaturated and saturated zones in the same manner as the Nominal Scenario Class Modeling Case (Section F.4.1.1).

In the Igneous Intrusion Modeling Case, a basaltic dike would intersect one or more emplacement drifts and magma would flow in and fill them, which would engulf the waste packages and drip shields. The magma would then cool and solidify. The model conservatively assumes that such an intrusion would destroy all waste packages in the repository; that is, all waste packages would lose structural integrity and their ability to prevent or limit the flow of water, and the movement of radionuclides would be completely compromised. After the drifts returned to temperatures lower than the boiling point of water, seepage into drifts would resume. The model conservatively assumes that the cooled magma would have hydrologic properties similar to the surrounding welded tuff, so the percolation flux into the intruded drift and waste package would be equivalent to percolation flux through the host rock. The rate of transport of radionuclides would depend on the temperature and chemistry of the groundwater. Thus, the percolation of water through cooled basalt would provide a mechanism for radionuclide release and transport.

F.2.10.2 Volcanic Eruption Modeling Case

The Volcanic Eruption Modeling Case considers the intrusion of one or more dikes into the repository and the formation of one or more eruptive conduits that would intersect emplacement drifts. Magma would destroy the waste packages in the conduits and entrain their waste. Contaminated volcanic tephra would be erupted into the atmosphere in a vertical column that reached altitudes up to 8.2 kilometers (5.1 miles), and would be dispersed by wind to the accessible environment (DIRS 177431-SNL 2007, Section 6.5.2.7). Surface processes (erosion and deposition by water and wind) could redistribute the tephra. DOE used information from the probabilistic volcanic hazard analysis to estimate the probability that one or more eruptive centers would form in the repository to assess the number of waste packages in the eruptive conduits. The Volcanic Eruption Modeling Case provides the TSPA-LA model with the number of waste packages that volcanic conduits would intercept, the aerial density of contaminated tephra, and the concentration of contaminated tephra from redistribution.

F.2.11 SEISMIC ACTIVITY DISRUPTIVE EVENTS

The Seismic Scenario Class describes future performance of the repository system if seismic activity disrupted the system. It represents the direct effects of vibratory ground motion and fault displacement associated with seismic activity, and it considers indirect effects of drift collapse. The Seismic Scenario Class considers the effects of seismic hazards on drip shields and waste packages. It also considers changes in seepage, waste package degradation, and flow in the Engineered Barrier System that could result from a seismic event. The *Seismic Consequence Abstraction* documents the conceptual models and abstractions for the mechanical response of Engineered Barrier System components to seismic hazards at a geologic repository (DIRS 176828-SNL 2007, all).

The Seismic Scenario Class estimates the mean annual dose due to a seismic event by accounting for the probability of occurrence of the event in terms of its mean annual exceedance frequency. The estimate of mean annual dose considers the relevant processes that would come into play and affect system performance. The Seismic Scenario Class has two modeling cases: (1) The Seismic Ground Motion Modeling Case includes waste packages that would fail solely due to the ground motion damage associated with the seismic event; and (2) the Seismic Fault Displacement Modeling Case includes only those waste packages that would fail due to fault displacement damage. These two cases have the same framework as the Nominal Scenario Class Modeling Case; that is, the framework includes the TSPA-LA model components to evaluate the mobilization of radionuclides that were exposed to seeping water, released from the Engineered Barrier System, transported in the unsaturated zone down to the saturated zone, and transported in the saturated zone from the repository to the location of the RMEI. Each component considers the effects of the seismic event, as appropriate.

F.2.11.1 Seismic Activity

The probabilistic seismic hazard analyses for ground motion used an expert elicitation process to determine the annual probability at which various levels of ground motion would be exceeded at Yucca Mountain (DIRS 103731-CRWMS M&O 1998, all). The results of this process provided hazard curves for a reference rock outcrop with the same seismic-wave propagation properties as the rock at the repository horizon inside Yucca Mountain. These results were modified to account for the effects of the site-specific geology of Yucca Mountain. The effects of the site materials [approximately the upper 300 meters (980 feet) of rock and soil] on ground motions at the waste emplacement level were calculated with the use of a ground motion site-response model. The acceleration response spectrum consists of the maximum response of a single-degree-of-freedom oscillator system (for a given damping ratio) to an input motion (accelerogram) as a function of the natural frequency of the system. The outputs of the site-response model (location-specific response spectra and peak ground velocity values) were used to scale recordings from past earthquakes to produce acceleration and velocity time histories (seismograms) for dynamic analyses to support postclosure performance assessment. Finally, when the models in the probabilistic seismic hazard analyses were applied, low-probability ground motion values were allowed to increase without bounds to eventually reach levels that are not credible for Yucca Mountain; that is, at low annual probabilities of exceedance, the calculated ground motions would produce strain levels in excess of the strength of the rock mass. Therefore, a separate analysis was performed to bound peak horizontal ground velocity at the waste emplacement level, with consideration of the maximum strain levels repository rocks could sustain (DIRS 170137-BSC 2005, Section 6). As Figure F-2 shows, the damage as a function of peak ground velocity level would be bounded by the combined hazard curve that results in a maximum peak ground velocity of approximately 4 meters (13 feet) per second at the 1×10^{-8} annual exceedance frequency. The analyses for the Seismic Scenario Class, therefore, fulfill the 10 CFR 63.114(d) requirements for performance assessment to consider events that have a frequency of at least 1×10^{-8} per year (1 chance in 10,000 of occurring within 10,000 years). The emphasis on peak horizontal ground velocity reflects the use of that ground motion measure to set parameters for rockfall and damage to Engineered Barrier System features for postclosure analyses.

The fault displacement analysis derives from the probabilistic seismic hazard analyses. This analysis used an expert elicitation process to determine how the annual probability of exceedances for fault displacement at the surface would vary as a function of the size of the displacement.

F.2.11.2 Mechanical Damage to the Engineered Barrier System

The *Seismic Consequence Abstraction* documents models for mechanical damage to the Engineered Barrier System from seismic activity (DIRS 176828-SNL 2007, all). The Seismic Scenario Class modeling cases consider vibratory ground motion, rockfall, and drift collapse from ground motion and fault displacement.

The seismic damage models for this Repository SEIS represent the current waste package design and respond to the requirement to analyze repository releases over periods that extend well beyond 10,000 years. The presence of a standardized TAD canister system (DIRS 177627-BSC 2006, all) is represented in the structural response calculations and corresponding damage abstractions. The degradation and potential failures of waste package components, the drip shield plates, and the drip shield framework due to general corrosion is represented in the structural response calculations and resultant damage

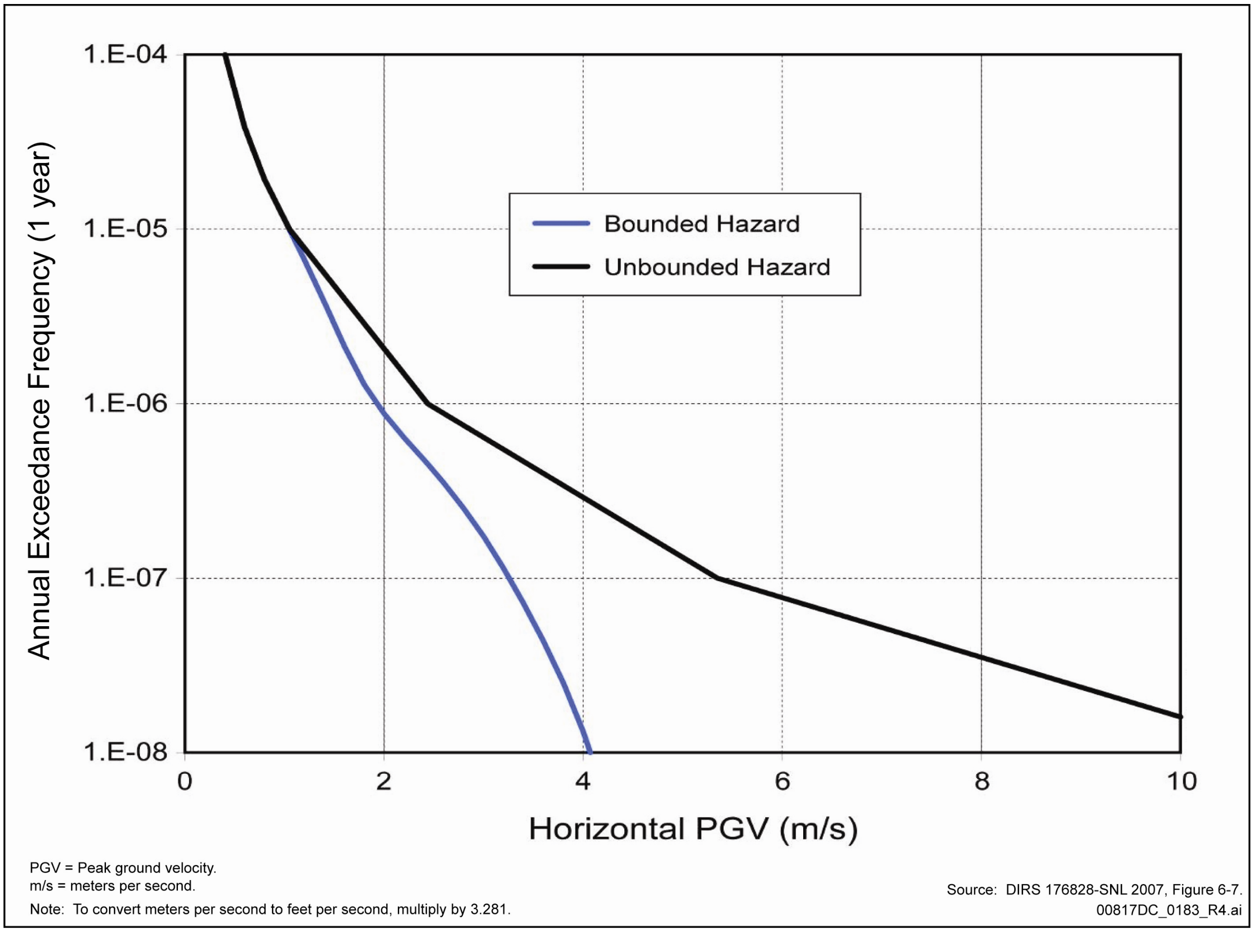

Figure F-2. Hazard curve for the Seismic Scenario Class.

abstractions. General corrosion thins and weakens the drip shields and waste packages over long periods by gradually thinning the drip shield plates and framework and waste package outer barrier.

Thinning makes these components more susceptible to being damaged by vibratory ground motion. In addition, once a waste package is breached by a through-wall crack or general corrosion, the waste package internal structures could degrade and reduce the structural resilience of the waste package. These factors were included in the TSPA-LA seismic damage calculations. Lastly, the TSPA-LA model considered the cumulative effects from multiple seismic events over very long time scales. The seismic damage abstractions capture the full range of these changes, with the associated uncertainties, for the Seismic Scenario Class for TSPA-LA.

F.2.11.3 Ground Motion Damage Modeling Case

Seismic events capable of causing damage in the Seismic Ground Motion Modeling Case could occur with a horizontal peak ground velocity greater than 0.219 meter (0.718 foot) per second and mean exceedance frequencies smaller than 4.29×10^{-4} per year (DIRS 176828-SNL 2007, Table 6-88). Seismic events were modeled as Poisson processes that were generated randomly with the specified rate of 4.29×10^{-4} per year (equal to the difference between the minimum annual exceedance frequency of 1×10^{-8} per year and the maximum annual exceedance frequency of 4.29×10^{-4} per year (DIRS 176828-

SNL 2007, Sections 5.2 and 6.12.2). The duration of the dose assessment is specified by EPA to end at 1 million years. During this period, the number of seismic events with the potential to damage Engineered Barrier System components would be, on average, 429 events (computed by multiplying the specified rate of the Poisson process, 4.29×10^{-4} per year, by the simulation period of 1 million years), so multiple seismic events would occur in each realization of the TSPA-LA model (70 FR 49014, August 22, 2005). The model accounts for the potential for deformation and rupture of Engineered Barrier System components from multiple seismic events. The probability of damage from an event was calculated separately for the codisposal and commercial spent nuclear fuel waste packages due to the inclusion of the TAD canister in the commercial spent nuclear fuel waste packages, which increased their structural strength. The structural damage from vibratory ground motion would be a function of the amplitude of the ground motion, expressed as horizontal peak ground velocity at the repository horizon. The peak ground velocity for a particular mean annual exceedance frequency, λ_S, is defined by the mean bounded hazard curve in Figure F-2. Note that since the value of the largest exceedance frequency in this figure is 1.0×10^{-4} per year, extrapolation was used to determine the peak ground velocities that correspond to exceedance frequencies between 1.0×10^{-4} per year and 4.29×10^{-4} per year. The extent of drift collapse, rockfall, and damage to the waste packages and drip shields was determined from rockfall and structural response calculations for different peak ground velocity values in *Seismic Consequence Abstraction* (DIRS 176828-SNL 2007, all). The same degree of damage to the drip shields and the same degree of damage to the waste packages were applied to all drip shields and waste packages; that is, there would be no spatial variability in degrees of damage from vibratory ground motion. The mechanical response of a drip shield and waste package would be determined by the time-dependent thickness of the drip shield and waste package components, dynamic and static rockfall loads on the drip shield and waste package, residual stress thresholds for the drip shield and waste package, and horizontal component of peak ground velocity. The mechanical response to vibratory ground motion could produce the following significant changes in the Engineered Barrier System components and the in-drift environment:

- Drift collapse and changes in seepage flux, temperature, and relative humidity for the emplacement drifts.

- Damage to the waste package (expressed as an area of stress corrosion cracks on the waste package surface) or by rupture/puncture probability of the waste package outer barrier as a result of deformation due to vibratory motion while the drip shield was intact and protected the waste package from rockfall.

- Damage to the drip shield plates (expressed as an area of stress corrosion cracks on the drip shield surface) or rupture/puncture probability as a result of accumulated rockfall or impact from rock blocks.

- Probability of failure (fragility) of the drip shield plates by tensile tearing or buckling of the drip shield framework as a result of accumulated rockfall and dynamic load amplification for future states of general corrosion thinning.

- Damage to the waste package (expressed as an area of stress corrosion cracks on the waste package surface) or rupture/puncture probability of the waste package outer barrier as a result of drip shield framework buckling collapse. The drip shield would continue to act as a seepage barrier, but would mechanically load the waste package outer barrier with static and dynamically amplified rubble loads.

This would account for future states of general corrosion thinning of the drip shield framework and waste package outer barrier, and degradation of waste package internals.

- Damage to the waste package (expressed as an area of stress corrosion cracks on the waste package surface) or rupture/puncture probability of the waste package outer barrier as a result of drip shield plate tearing failure. The drip shield would fail as a seepage and rockfall barrier, with subsequent rubble in direct contact with the waste package outer barrier, thus applying static and dynamically amplified rubble loads. This would account for future states of general corrosion thinning of the drip shield plates and waste package outer barrier, and degradation of waste package internals.

- Failure of the fuel cladding. Failure of the fuel cladding could occur from fuel assembly accelerations during the seismic event. However, the TSPA-LA does not take credit for the cladding as a barrier to radionuclide release, so it does not incorporate the dynamic response of the cladding and associated damage abstraction.

- The most likely failure mechanism from a seismic event would be accelerated stress corrosion cracking in the damaged areas that exceeded the residual stress threshold for Alloy 22 (the waste package outer barrier). Other failure mechanisms as noted above included the potential for rupture or puncture of the outer corrosion barrier of the waste package in response to a high-amplitude, low-probability earthquake after general corrosion had significantly weakened the Engineered Barrier System components. Stress corrosion cracks on the waste package surface would be a potential pathway for diffusive transport of radionuclides out of the waste package. Rupture or puncture of the waste package would be a potential pathway for advective transport of radionuclides out of the waste package.

F.2.11.4 Fault Displacement Modeling Case

Seismic events capable of causing damage in the Seismic Fault Displacement Modeling Case would not occur with mean exceedance frequencies greater than 2.2×10^{-7} per year (DIRS 183478 SNL-2008, Table 6.6-1). For a fault displacement along an emplacement drift, a sudden discontinuity in the floor and roof of the drift could occur and, if severe enough, cause shearing failure of a waste package and drip shield. If a waste package was breached by fault displacement, the damaged area on the waste package would be determined by sampling a uniform distribution with a lower bound of zero and an upper bound equal to the area of the waste package lid. The drip shield for this waste package is also assumed to breach (DIRS 183478 SNL-2008, all).

The area on the waste package represents the extremes of response. The damaged area could be none for a package that experienced very minor crimping without breach. It could be as large as the waste package lid if the lid welds were broken from severe crimping of the package due to fault displacement. The expected number of waste package failures that could occur would depend on the annual exceedance frequency of a seismic event and could range from 25 waste packages for an annual exceedance frequency of approximately of 2×10^{-7} per year to 214 waste packages for a very low probability, annual exceedance frequency of 2.6×10^{-8} per year. These numbers of waste packages would be a small fraction of the total number of waste packages in the repository. The estimated number of failed waste packages is based on an understanding of the displacements that could occur on these faults and geometric considerations, as described in *Seismic Consequence Abstraction* (DIRS 176828-SNL 2007, all). The conceptual model specifies that when a waste package failed from fault displacement, the associated drip shield and fuel rod cladding would also fail. A sheared drip shield would allow all seepage to pass through it; that is, the damaged area would be the total surface area of the drip shield, so there would be no flux splitting (diversion of seepage) (DIRS 176828-SNL 2007, all).

F.2.12 NUCLEAR CRITICALITY

A nuclear criticality occurs when sufficient quantities of fissionable materials come together in a precise manner and the required conditions exist to start and sustain a nuclear chain reaction. In the proposed repository, one of the required conditions would be the presence of a moderator, such as water, in the waste package. The waste package design would make the probability of a criticality inside a waste package extremely small. In addition, based on an analysis of anticipated repository conditions, the accumulation of a sufficient quantity of fissionable materials outside the waste packages in the precise configuration and with the required conditions to create a criticality would be very unlikely. As a result, nuclear criticality has been excluded from this Repository SEIS.

F.3 Inventory

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

F.3.1 INVENTORY FOR WATERBORNE RADIOACTIVE MATERIALS

There would be more than 200 radionuclides in the materials in the repository (DIRS 177424-SNL 2007, all). In the Proposed Action, these radionuclides would be present in five basic waste forms: (1) commercial spent nuclear fuel, (2) mixed-oxide fuel and plutonium ceramic (plutonium disposition waste), (3) borosilicate glass formed from liquid wastes on DOE sites (high-level radioactive waste), (4) DOE spent nuclear fuel, and (5) naval spent nuclear fuel (DIRS 180472-SNL 2007, all). DOE would place these wastes in several different types of waste packages of essentially the same construction but of varying sizes and with varying types of internal details. It is neither necessary nor practical to model the exact configuration of waste packages for postclosure performance assessment. The details of each package design are not significant parameters in the modeling of processes for waste package degradation, waste form degradation, and radionuclide transport. Construction of a TSPA-LA model with each waste package and its unique design would result in a model too large to run on any available computer in a practicable time.

DOE developed the abstracted inventory to maintain essential characteristics of the waste forms for input to the TSPA-LA model. The TSPA-LA model is a high-level system model that performs hundreds of calculations in a Monte Carlo framework. To make such a calculation practicable, DOE had to reduce highly complex descriptions of behaviors to simplified concepts that capture the essential characteristics. In the case of inventory, DOE considered the highly complex array of waste streams for the five fundamental waste categories in the development of the abstraction to representative waste packages that captures the essential features of the total inventory of radionuclide materials. The analysis used two representative types—a commercial spent nuclear fuel waste package and a codisposal waste package that would contain DOE spent nuclear fuel and vitrified high-level radioactive waste. For this analysis, naval spent nuclear fuel was conservatively modeled as commercial spent nuclear fuel (DIRS 183478-SNL

2008, all). The plutonium disposition waste was split into the commercial spent nuclear fuel package (mixed-oxide fuel) and codisposal package (immobilized plutonium in a high-level radioactive waste container) (DIRS 177424-SNL 2007, all). Table F-3 summarizes the abstracted inventory.

Table F-3. Initial radionuclide inventories (grams per package)[a] in 2117 for each idealized waste package type in the TSPA-LA model.[b]

Radionuclide	Commercial spent nuclear fuel package	Codisposal package
Actinium-227	0.00000627	0.00233282
Americium-241	9,838.2	249.081
Americium-243	1,234.2	7.2453
Carbon-14	1.3418	1.791
Cesium-135	4,359.9	224.397
Cesium-137	1,861.1	53.842
Chlorine-36	3.2296	4.2292
Curium-245	17.428	0.145759
Iodine-129	1,730	108.3
Lead-210	0	0.0000000233
Neptunium-237	5318.8	216.66
Protactinium-231	0.012205	3.6655
Plutonium-238	1,022.2	25.9096
Plutonium-239	43,143	2,761.11
Plutonium-240	20,391	476.687
Plutonium-241	240.33	0.468165
Plutonium-242	5,279.5	34.0844
Radium-226	0.00012909	0.000207
Radium-228	0.000000000019	0.0000208233
Selenium-79	41.895	13.8272
Strontium-90	745.69	27.8785
Technetium-99	7,548.8	1,167.96
Thorium-229	0.0000207	0.532074
Thorium-230	0.43187	0.2419906
Thorium-232	0.056268	51,500
Tin-126	462.94	26.3937
Uranium-232	0.0061966	0.53893173
Uranium-233	0.13657	557.195
Uranium-234	2,239.2	521.445
Uranium-235	62,661	26,516.4
Uranium-236	38,507	1,314.216
Uranium-238	7,820,000	921,000

Source: DIRS 183478-SNL 2008, all.

a. To convert grams to ounces, multiply by 0.035274.

b. While the total inventory is represented by the material in the idealized waste packages, the actual number of waste packages DOE emplaced in the proposed repository could be different.

Note that the abstracted inventory does not apply to any other analysis, because it does not specifically model each waste form but rather models a surrogate waste form that is a useful and defensible abstraction for the purpose. The averaging, blending, and screening of radionuclides to reduce the total number, while retaining essential physical characteristics of the waste, were tailored to the TSPA-LA model. Therefore, a comparison of this abstracted inventory with other abstractions for other analyses would not be valid.

F.3.2 INVENTORY FOR WATERBORNE CHEMICALLY TOXIC MATERIALS

DOE would use several corrosion-resistant metals that contain chemically toxic materials in the construction of the repository. The Department used a screening analysis in the Yucca Mountain FEIS to determine which, if any, of these materials would have the potential for transport to the accessible environment in sufficient quantities to be toxic to humans. Chemicals in the EPA substance list for the Integrated Risk Information System (DIRS 103705-EPA 1997, all; DIRS 148219-EPA 1999, all; DIRS 148221-EPA 1999, all; DIRS 148224-EPA 1998, all; DIRS 148227-EPA 1999, all; DIRS 148228-EPA 1999, all; DIRS 148229-EPA 1999, all; DIRS 148233-EPA 1999, all) were evaluated to determine a concentration that could occur in drinking water downgradient from the repository. The chemicals on that list that would be in the repository are barium, boron, cadmium, chromium, copper, lead, manganese, mercury, molybdenum, nickel, selenium, uranium, vanadium, and zinc. These chemicals would occur in construction materials of the repository and waste package and in the waste forms in the waste packages.

Only a few waste packages would fail during the first 10,000 years after closure (Section F.2.4). The period of consideration for chemically toxic material impacts is 10,000 years. Therefore, only toxic materials outside the waste package were of concern in this analysis. The Yucca Mountain FEIS described a screening analysis of materials in the proposed repository (DIRS 155970-DOE 2002, p. I-29), which this Repository SEIS incorporates by reference. The materials of concern from that screening analysis are chromium, copper, manganese, molybdenum, nickel, and vanadium.

F.4 Postclosure Radiological Impacts

For the Proposed Action, DOE conducted a detailed postclosure consequence analysis to assess compliance with the individual protection and groundwater protection standards (40 CFR 197.20 and 40 CFR 197.30). The analysis provided projections of doses and radionuclide concentrations for the period up to 10,000 years after closure and the post-10,000-year period. The doses calculated for comparison with individual protection standards are the mean annual dose for the first 10,000 years after closure and median annual dose for the post-10,000-year period.

The individual protection and groundwater standards apply to the designated location of the RMEI, which is prescribed in the EPA regulation as about 18 kilometers (11 miles) downgradient from the repository. This is the farthest southern point on the boundary of the controlled area and the location of the accessible environment (40 CFR 197.12). It corresponds to where the RMEI would consume and use groundwater. DOE evaluated compliance at the point where the highest radionuclide concentration in the simulated contamination plume would cross the southernmost boundary of the controlled area (at a latitude of 36 degrees, 40 minutes, 13.6661 seconds north) (40 CFR 197.21 and 197.31).

For the individual protection standard, DOE estimated the mean and median annual individual doses by combining performance assessment results for four primary scenario classes:

- Nominal Scenario Class (natural evolution of the repository system in the absence of disruptive events),

- Early Failure Scenario Class (early failure of waste packages and drip shields due to material defects, process failures, human errors),
- Igneous Scenario Class (hypothetical intrusion and volcanic eruption), and
- Seismic Scenario Class (hypothetical vibratory ground motion and fault displacement).

For the individual and groundwater protection standards, DOE computed the estimates of annual doses and radionuclide concentrations for the RMEI location using the NRC-specified representative volume of 3.7 million cubic meters (3,000 acre-feet) of groundwater (10 CFR 63.332) that would be drawn annually from the aquifer at the accessible environment to calculate the concentration of radionuclides. The TSPA-LA model collects all the radionuclides that would be released from the repository and transported through the unsaturated and saturated zones to the accessible environment and subsequently mixed in the representative volume or annual water demand of the RMEI.

The postclosure consequence analysis for the Proposed Action conformed to the NRC technical requirements (10 CFR 63.114). The TSPA-LA model calculates estimates of projected annual dose and groundwater concentrations in a probabilistic framework. It uses a Monte Carlo simulation technique to address the epistemic uncertainty and aleatory uncertainty in the values of the input parameters. It generates multiple realizations of input parameters by sampling from assigned probability distributions and simulating the performance of the repository system. As noted above, the postclosure analysis provided projections of doses and radionuclide concentrations for the first 10,000 years after closure and for the post-10,000-year period. For all scenario classes, the analysis for this Repository SEIS made separate TSPA calculations for each period to ensure adequate numerical accuracy and statistical stability of results. For example, to achieve sufficient accuracy in the 10,000-year period results, it was necessary to implement much smaller time steps in the numerical calculations. The largest time step in the 10,000-year calculations was 80 years. The largest time step in the post-10,000-year calculations was 4,000 years. In addition, the smallest time step in the post-10,000-year calculations was 400 years, which was used as the time step for the first 10,000 years of the post-10,000-year calculations. As a result, the projected doses at 10,000 years, for the 10,000-year and post-10,000-year calculations, would in general be different but sufficiently accurate to project groundwater concentrations and mean and median annual doses.

COLOR FIGURES

The figures illustrating results of the performance analysis presented in Chapter 5 and Appendix F can be found in color on the CD on the inside back cover of the Summary of this Repository SEIS and the Office of Civilian Radioactive Waste Management Web site: http://www.ocrwm.doe.gov. Some of the figures can also be found in color in the Summary.

The main result of the Monte Carlo simulation process is a set of realizations for the expected annual dose histories for the RMEI, which are generally plotted in the form of a multi-realization plot. Chapter 5, Figure 5-4 shows the multi-realization plots DOE developed for demonstrating compliance with the individual protection standard for 10,000 years after closure, and Figure 5-6 shows the plots for the post- 10,000-year period [that is, after 10,000 years but within the period of geologic stability (as defined by the proposed NRC rule, (70 FR 53313, September 8, 2005) to end at one million years)].

Curves for the mean, median, and 5th- and 95th-percentile dose histories are superimposed on each multi-realization plot. The total mean annual dose history (the red curve) was computed by taking the arithmetic average of the 300 epistemic uncertainty vectors that were sampled for each modeling case for individual time planes along the curves. Similarly, the median dose history (the blue curve) was constructed from points that were obtained by sorting the 300 expected values from lowest to highest and then averaging the two middle values. Curves for the 5th- and 95th-percentile dose histories are plotted in yellow and green, respectively, to illustrate the spread in the expected annual dose histories; 90 percent (or 270 of the 300 epistemic realizations) of the projected dose histories fall between these two percentile curves. For a detailed description of the calculation of total annual dose, see *Total System Performance Assessment Model/Analysis for the License Application* (DIRS 183478-SNL 2008, Section 6.1.2.2).

F.4.1 IMPACTS FROM REPOSITORY PERFORMANCE IN THE ABSENCE OF DISRUPTIVE EVENTS

This section discusses repository performance in the absence of seismic and igneous activity. It examines two scenario classes—Nominal Scenario Class and Early Failure Scenario Class. In this and subsequent sections, impacts from repository performance are described using annual dose histories that illustrate the calculated mean and median annual doses for the different modeling cases. In addition, dose histories of major radionuclides that contribute to the estimate of mean annual dose are presented. These latter time histories illustrate the important radionuclides that contribute to mean annual dose and generally are typical of key radionuclides that contribute to median dose.

F.4.1.1 Nominal Scenario Class

The Nominal Scenario Class for the TSPA-LA model includes the features, events, and processes relevant to the natural evolution and degradation of the repository system, but excludes those features, events, and processes for the Early Failure, Igneous, and Seismic Scenario Classes. More specifically, the Nominal Scenario Class includes features, events, and processes for waste package and drip shield degradation as a function of expected corrosion processes (for example, general corrosion, stress corrosion cracking, and seepage-induced localized corrosion) that the hydrologic and geochemical environments, which would vary with time, would induce. The Nominal Scenario Class also includes the important effects and system perturbations due to climate change and repository heating, which would occur after repository closure. DOE modeled the failure of the waste packages and drip shields, degradation of the waste forms, mobilization of radionuclides, and subsequent release from the Engineered Barrier System. The Nominal Scenario Class includes migration of radionuclides by groundwater that would percolate through the unsaturated zone to the saturated zone and then travel to the accessible environment.

Figure F-3 shows the projected annual dose results of 300 probabilistic simulations for the Nominal Scenario Class Modeling Case at the RMEI location [about 18 kilometers (11 miles) downgradient from the proposed repository] for the post-10,000-year period. The mean, median, and 5th- and 95th-percentile curves in Figure F-3 show uncertainty in the value of the projected annual dose, with consideration of epistemic uncertainty from incomplete knowledge of the behavior of the physical system.

The results for this modeling case show zero mean annual dose for the first 10,000 years after closure because no waste packages are estimated to fail (by general corrosion, localized corrosion, or stress corrosion cracking) during this period. The first waste package failure (by nominal stress corrosion cracking) would occur at approximately 170,000 years, and the drip shields would begin to fail by general corrosion at approximately 260,000 years. Undetected manufacturing or material defects, including improper preemplacement operations, would contribute to dose releases before 170,000 years. As Figure

F-3 shows, the projected mean and median annual doses are 0.5 and 0.3 millirem, respectively, for the post-10,000-year period. Figure F-4 shows the radionuclides that dominate the projected mean annual

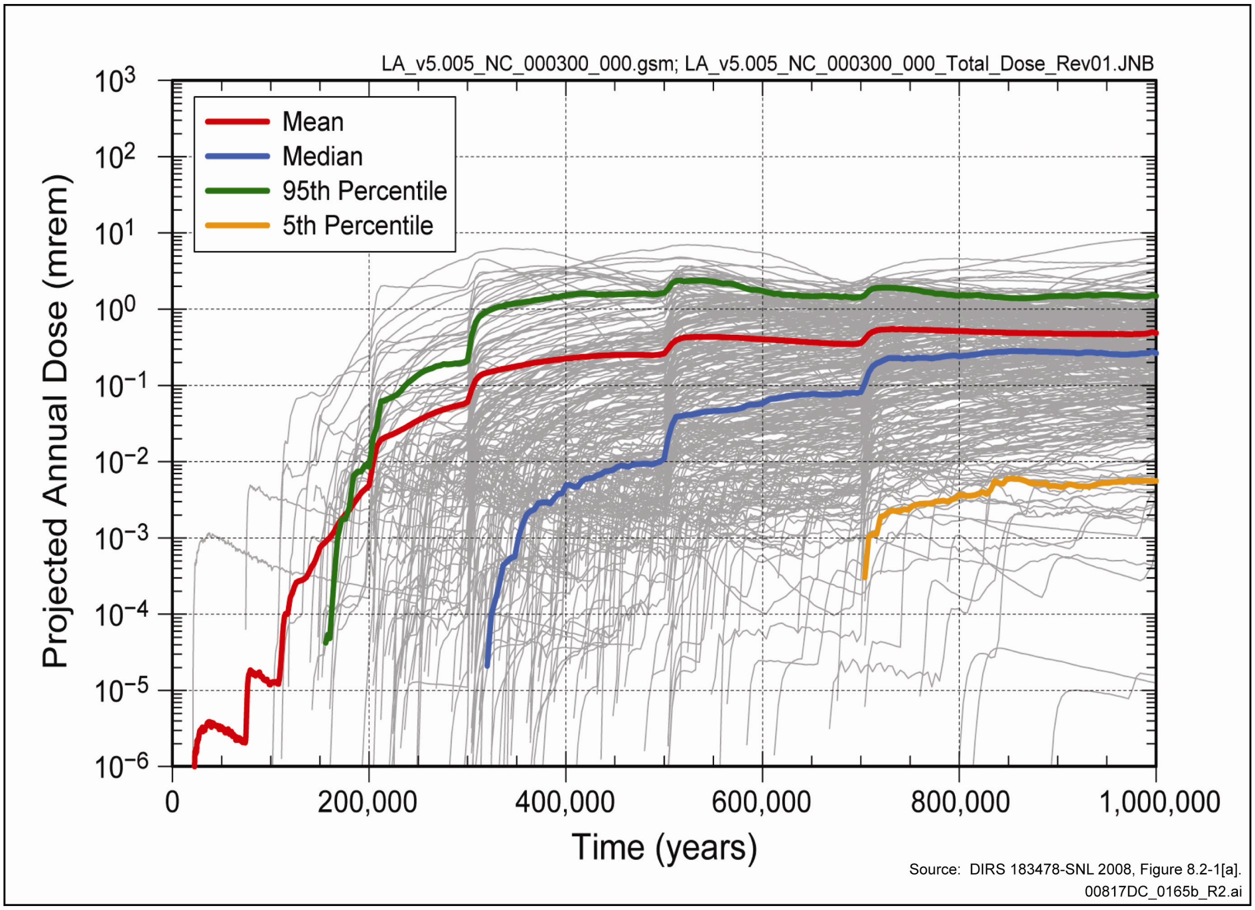

Figure F-3. Projected annual dose for the Nominal Scenario Class Modeling Case for the post-10,000-year period.

dose for the Nominal Scenario Case. The main contributors to mean annual dose would be the highly soluble and mobile radionuclides iodine-129 and technetium-99.

F.4.1.2 Early Failure Scenario Class

The Early Failure Scenario Class includes features, events, and processes that relate to early waste package and drip shield failure due to manufacturing, material defects, or preemplacement operations that would include improper heat treatment. In addition, this scenario class includes all features, events, and processes in the Nominal Scenario Class. As in the Nominal Scenario Class, failure of the waste packages and drip shields would ultimately lead to waste form exposure to water and mobilization and eventual release of radionuclides from the repository. Groundwater percolation through the unsaturated zone would transport the radionuclides to the saturated zone and then to the accessible environment by water flow in the saturated zone. Section F.2.4 describes the analysis of drip shield and waste package early failures in the TSPA-LA model.

DOE evaluated two modeling cases for this scenario class—Drip Shield Early Failure and Waste Package Early Failure. The following sections describe these modeling cases.

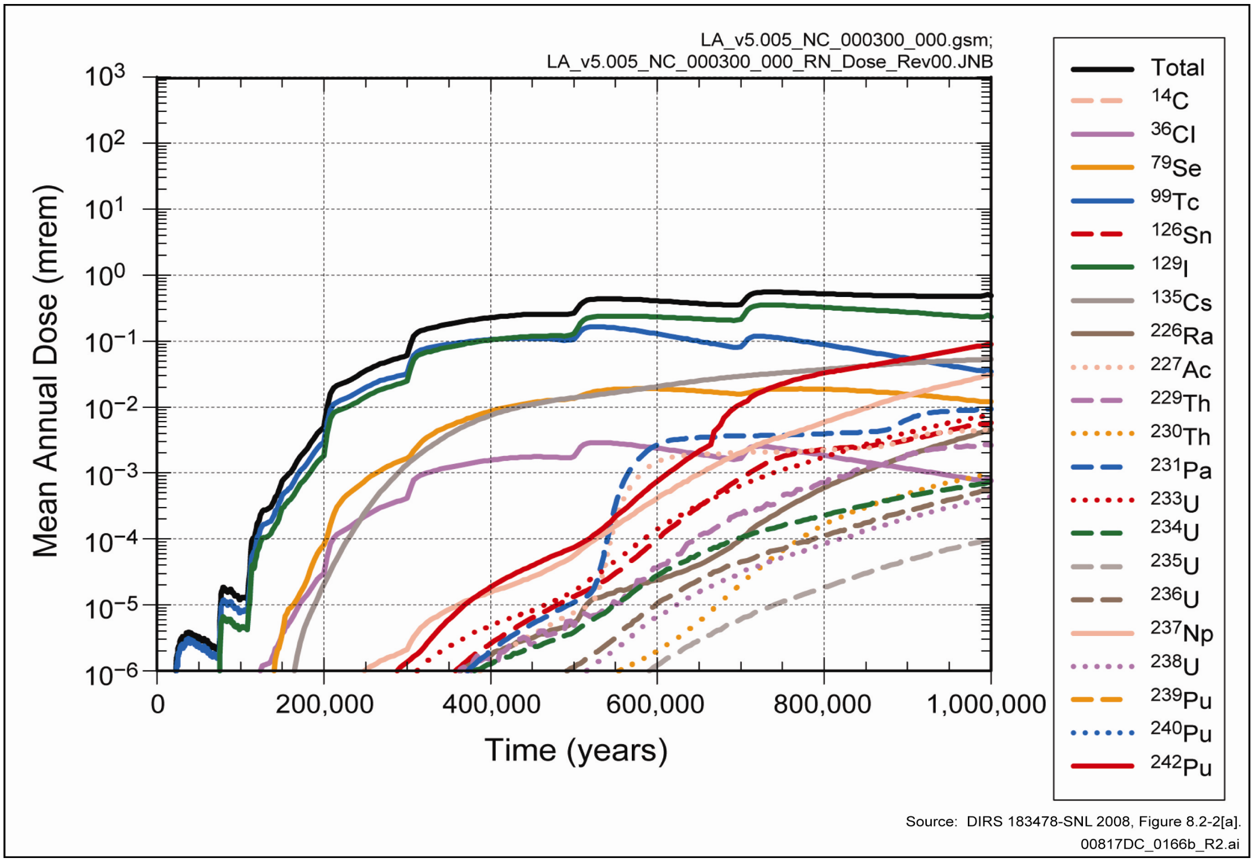

Figure F-4. Mean annual dose histories of major radionuclides for the Nominal Scenario Class Modeling Case for the post-10,000-year period.

F.4.1.2.1 *Drip Shield Early Failure Modeling Case*

The analysis for this modeling case assumed that the defective drip shields would fail at the time of repository closure. It also assumed that waste packages under these defective drip shields would fail early. (The Nominal Scenario Class Modeling Case does not include these unexpected conditions.) Figure F-5 shows the performance assessment calculations of the annual dose histories; the plot shows projections for annual doses for the first 10,000 years after closure and the post-10,000-year period. The estimated doses account for aleatory uncertainty for characteristics of the early failed drip shields such as the number of early failed drip shields, types of waste package under failed drip shields, and their locations in the repository. The mean, median, and 5th- and 95th-percentile curves in this plot show the uncertainty in the magnitude of the projected annual dose, which reflects the epistemic uncertainty from incomplete knowledge of the behavior of the physical system. The calculations for the first 10,000 years give a projected mean annual dose of approximately 0.0003 millirem estimated to occur at approximately 2,000 years. The projected annual doses decline thereafter and drop to less than 0.0003 millirem for the post-10,000-year period.

Figure F-6 shows the radionuclides that would contribute most to the total mean annual dose for the Drip Shield Early Failure Modeling Case. In the first 2,000 years after repository closure, soluble and mobile

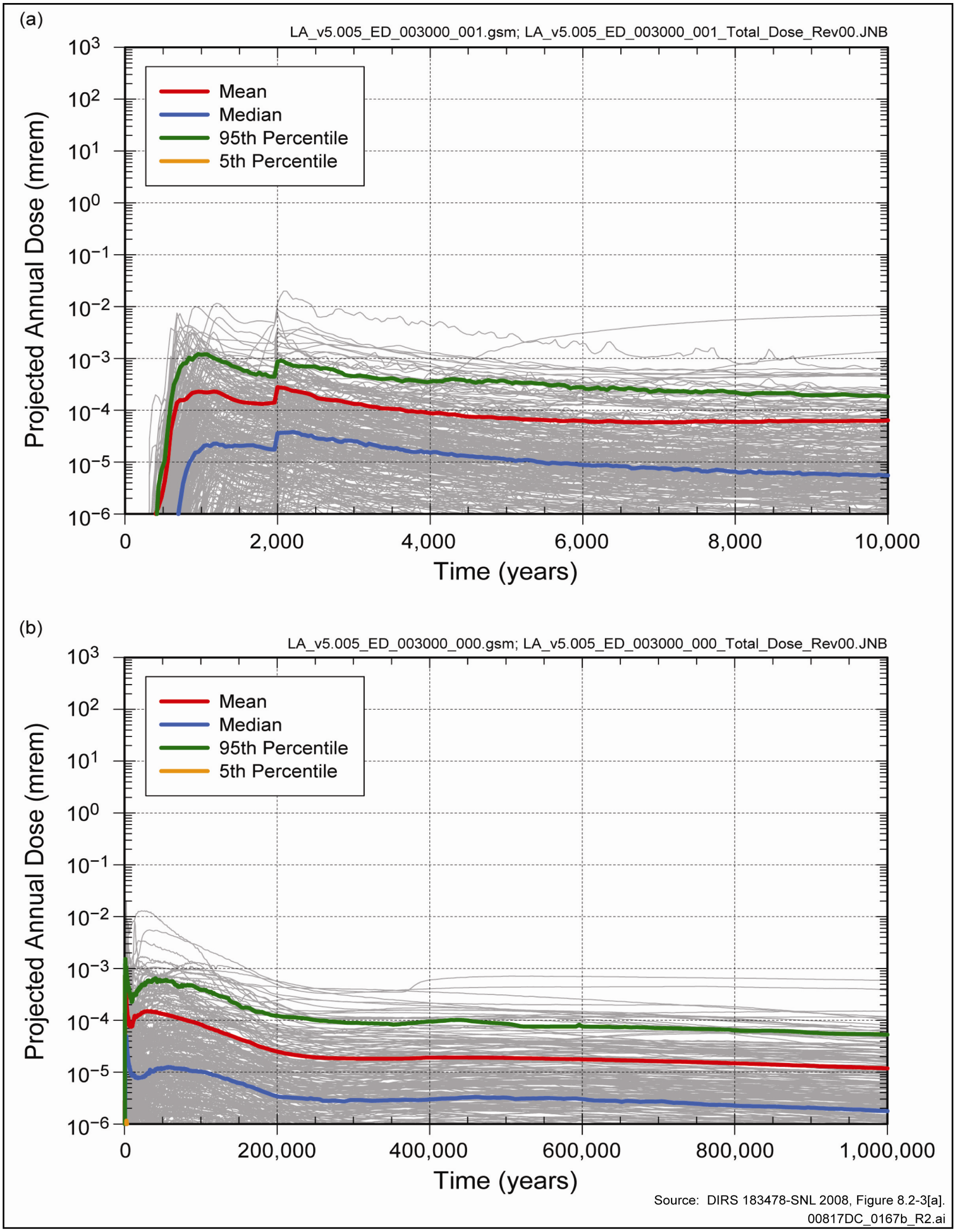

Figure F-5. Projected annual dose for the Drip Shield Early Failure Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

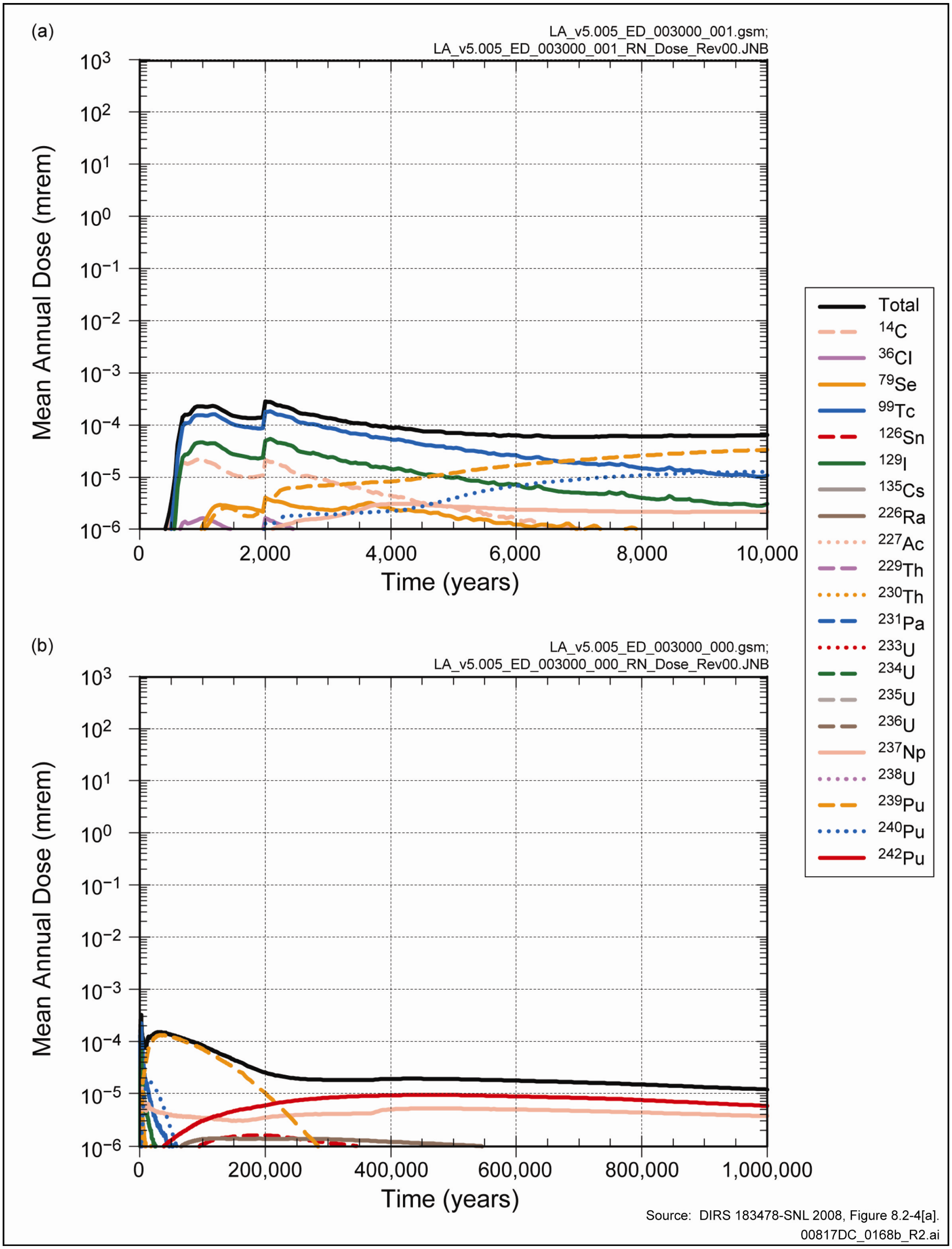

Figure F-6. Mean annual dose histories of major radionuclides for the Drip Shield Early Failure Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

radionuclides, in particular technetium-99, iodine-129, and carbon-14, would be the primary contributors to the mean annual dose. During the post-10,000-year period, the radionuclides plutonium-239, plutonium-242, and neptunium-237 would dominate the mean annual dose.

F.4.1.2.2 *Waste Package Early Failure Modeling Case*

This modeling case assumes that the defective waste packages would fail at the time of repository closure. However, it assumes that the drip shields would degrade by general corrosion and fail in accordance with the Nominal Scenario Class Modeling Case. Figure F-7 shows the annual dose histories for this modeling case for the first 10,000 years after closure and the post-10,000-year period. The projected dose accounts for aleatory uncertainty for characteristics of the early failed waste packages such as the number of early failed waste packages, types of early failed waste packages, and their locations in the repository. The mean, median, and 5th- and 95th-percentile curves in Figure F-7 show uncertainty in the value of the projected annual dose, with consideration of epistemic uncertainty from incomplete knowledge of the behavior of the physical system.

For the first 10,000 years after repository closure, the projected mean annual dose is simulated to be about 0.004 millirem and to occur at about 9,900 years. Annual doses would increase after the climate changed at 10,000 years. The projected mean and median annual doses are simulated to reach levels of about 0.2 and 0.006 millirem, respectively, before 15,000 years and gradually to decline thereafter.

Figure F-8 shows the projected mean annual dose from the radionuclides that would contribute most to the total mean annual dose for the Waste Package Early Failure Modeling Case. In the first 10,000 years after closure, more soluble and mobile radionuclides, in particular technetium-99, iodine-129, and carbon-14, would dominate the estimate of mean annual dose. During the post-10,000-year period, the mobile radionuclides technetium-99, iodine-129, and carbon-14 are projected to dominate the annual dose.

F.4.2 IMPACTS FROM DISRUPTIVE EVENTS

This section discusses disruptive events that include those due to seismic and igneous activity. Chapter 5, Section 5.8 discusses inadvertent intrusion into the repository by a drilling crew.

F.4.2.1 Igneous Scenario Class

The Igneous Scenario Class describes the performance of the repository system in the event of igneous activity that would disrupt the repository. This class includes all features, events, and processes in the Nominal Scenario Class (Section F.4.1.1). In addition, it includes the set of features, events, and processes specific to igneous disruption. The Igneous Scenario Class consists of two modeling cases: (1) the Igneous Intrusion Modeling Case, which represents the interaction of an intrusive magma dike into the repository and subsequent release of radionuclides to the groundwater pathway, and (2) the Volcanic Eruption Modeling Case, which represents a hypothetical volcanic eruption through the repository that would emerge at the land surface and cause releases of radionuclides to the atmospheric pathway.

F.4.2.1.1 *Igneous Intrusion Modeling Case*

In this modeling case, a magmatic dike would intersect the footprint of the repository. Radionuclide release and transport away from the repository would be similar to the Nominal Scenario Class Modeling

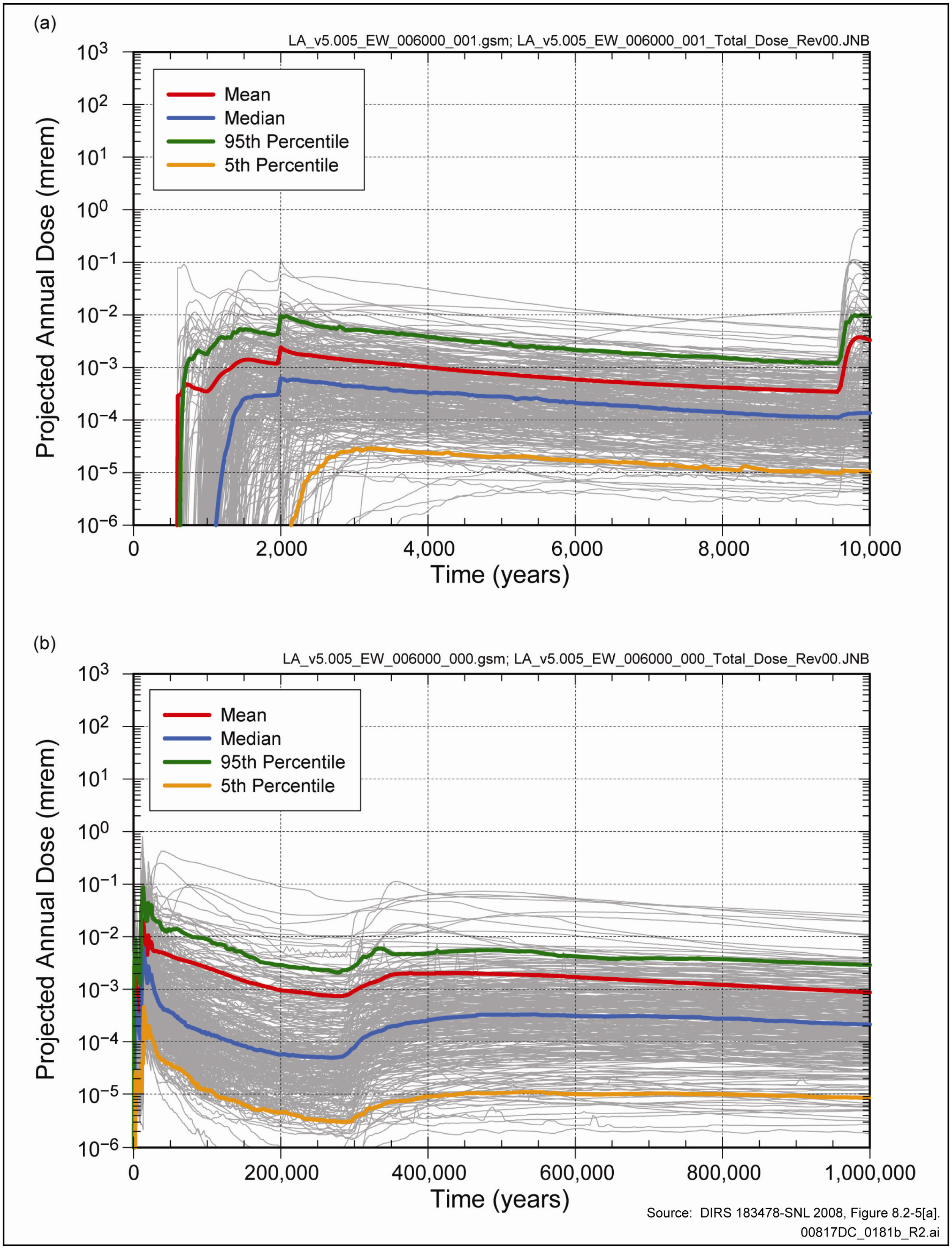

Figure F-7. Projected annual dose for the Waste Package Early Failure Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

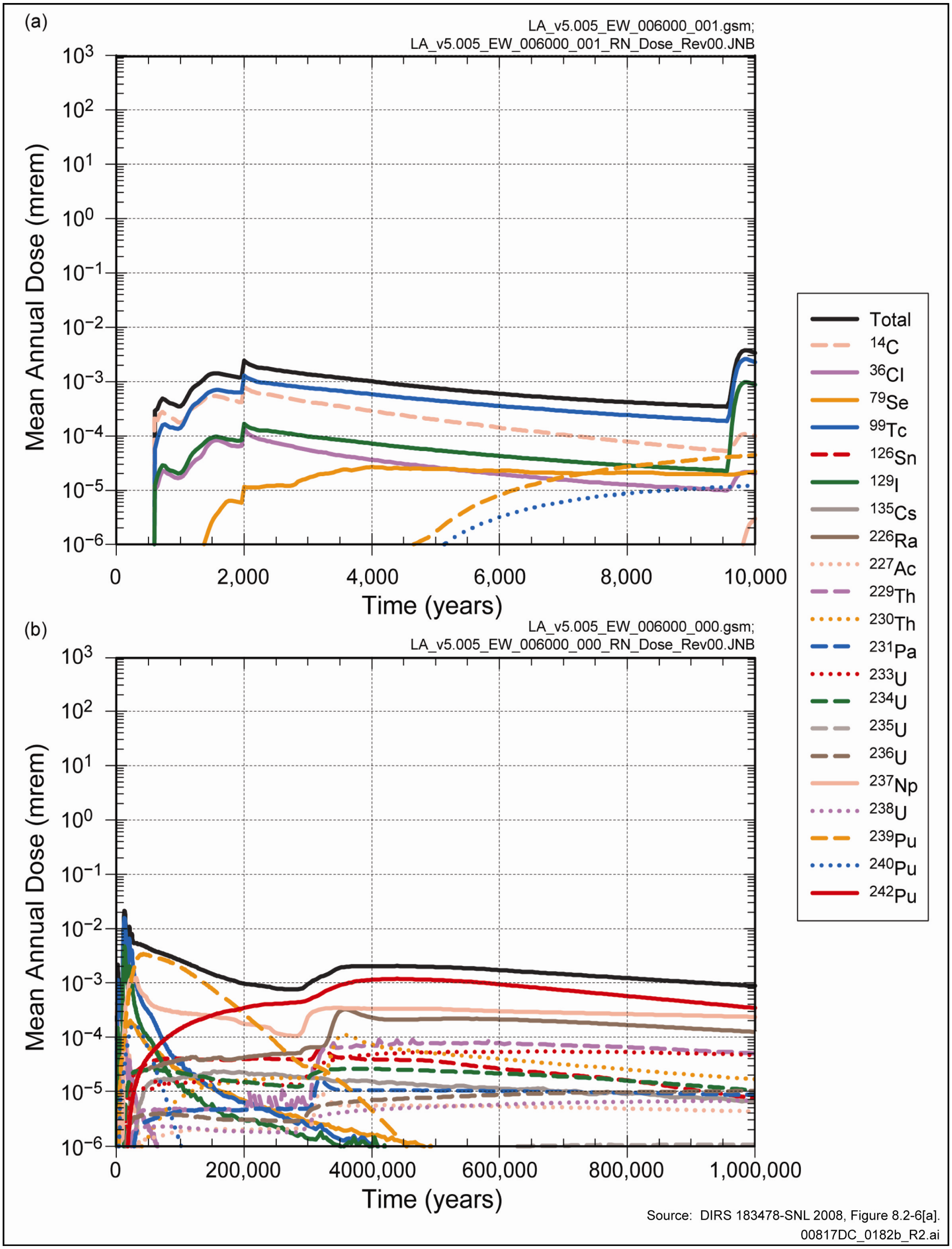

Figure F-8. Mean annual dose histories of major radionuclides for the Waste Package Early Failure Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

Case for radionuclide release and transport (Chapter 5, Section 5.5), but this case included the intrusion. There are two main components to the model—the behavior of the waste packages and other Engineered Barrier System elements damaged by an igneous intrusion, and groundwater flow and radionuclide transport away from the waste packages. The modeling case conservatively assumed that all of the drip shields and waste packages in the repository would be damaged, which would expose the waste forms to percolating groundwater with subsequent degradation, and radionuclide mobilization and transport.

Radionuclide transport would occur through the invert into the unsaturated zone, depending on solubility limits and the rate of water flux through the intruded drifts. The modeling case conservatively assumed that the drifts would not act as a capillary barrier, and the seepage water flux into a magma-intruded drift would be equal to the percolation flux in the overlying host rock. It took no credit for water diversion by the remnants of the drip shield, waste package, or cladding. Actual thermal, chemical, hydrological, and mechanical conditions in the drift after igneous intrusion are unknowable, but a conservative assumption that the engineered barriers completely failed would be sufficient to compensate for the uncertainty about drift conditions.

Figure F-9 shows projected annual dose histories for the Igneous Intrusion Modeling Case for the first 10,000 years after closure and the post-10,000-year period. The projected dose accounts for aleatory uncertainty for characteristics of the igneous intrusion such as the number of future events and the time at which they occurred. The mean, median, and 5th- and 95th-percentile curves in Figure F-9 show uncertainty in the value of the projected annual dose, with consideration of epistemic uncertainty from incomplete knowledge of the behavior of the physical system during and after the disruptive event. These figures show that the mean projected dose for 10,000 years after closure is less than 0.07 millirem and for the post-10,000-year period is about 0.9 millirem. The median projected annual dose for the post-10,000-year period is less than 0.3 millirem.

The results in Figure F-10 show the radionuclides that would contribute most to the estimate of mean projected dose for the Igneous Intrusion Modeling Case. Figure F-10a shows that technetium-99 and iodine-129 would dominate the estimate of the mean for the first 4,000 years and that plutonium-239, technetium-99, and plutonium-240 would dominate the estimate of the mean for the 10,000-year postclosure period. Figure F-10b shows that plutonium-239 in both dissolved and colloidal forms would dominate the estimate of the mean for the next 170,000 years, and that plutonium-242, neptunium-237, and radium-226 would dominate the estimate of the mean for the remainder of the post-10,000-year period.

F.4.2.1.2 *Volcanic Eruption Modeling Case*

The conceptualization of a volcanic eruption at Yucca Mountain envisioned an igneous dike that would rise through the Earth's crust and intersect one or more repository drifts. An eruptive conduit could form somewhere along the dike as it neared the surface and fed a volcano. Waste packages in the direct path of the conduit would be destroyed, and the waste in those packages would be entrained in the eruption. Volcanic ash would be contaminated, erupted, and transported by wind. Ash would settle out of the plume as it was transported downwind, which would result in an ash layer on the land surface. Members of the public would receive a radiation dose from exposure pathways for the contaminated ash layer.

Model development included the incorporation of conservative assumptions about the event, selection of input parameter distributions that characterize important physical properties of the system, and use of a

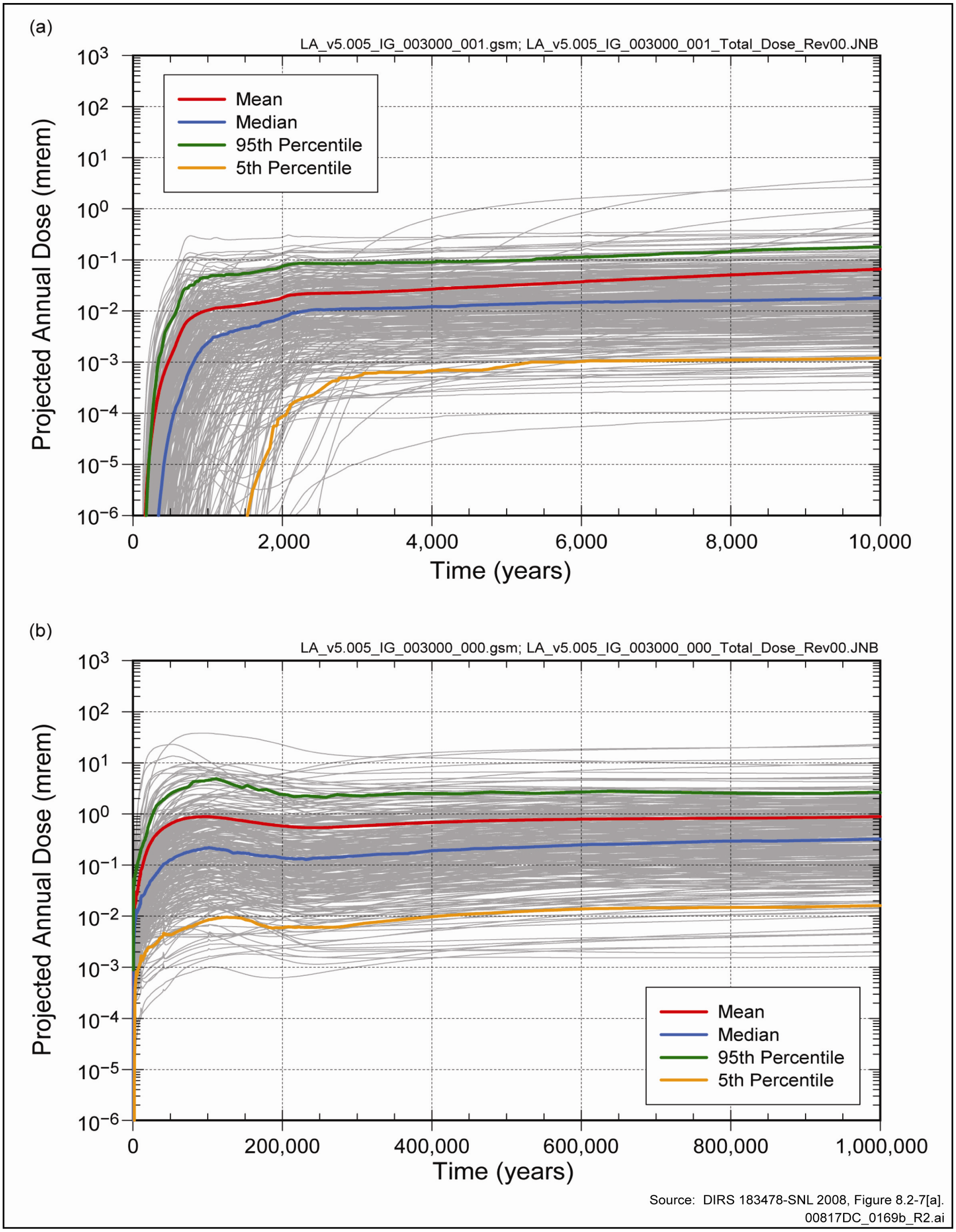

Figure F-9. Projected annual dose for the Igneous Intrusion Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

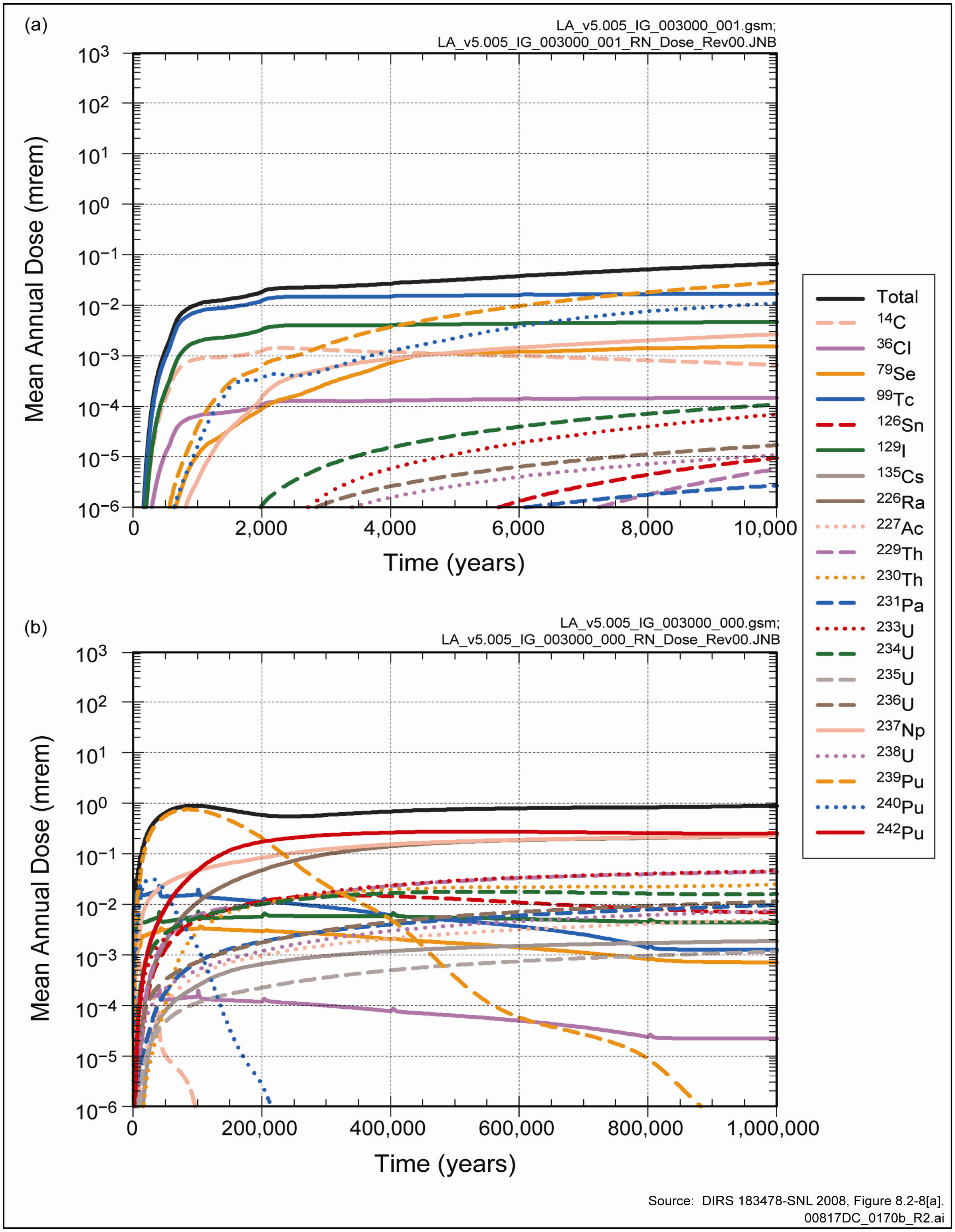

Figure F-10. Mean annual dose histories of major radionuclides for (a) the Igneous Intrusion Modeling Case for the first 10,000 years after repository closure and (b) the post-10,000-year period.

computational model to calculate entrainment of waste in the erupting ash. Each intrusive event (a swarm of one or more dikes) could generate one or more volcanoes somewhere along its length, but eruptions would not have to occur in the repository footprint. Approximately 28 percent of intrusive events that intersected the repository would result in one or more surface eruptions in the repository footprint. The number of eruptive conduits (volcanoes) would be independent of the number of dikes in a swarm. The analysis included characteristics of the eruption such as eruptive power, style (violent or normal), velocity, duration, column height, and total volume of erupted material.

Figure F-11 shows an estimate of the uncertainty in the projected dose for the volcanic eruption modeling case for the first 10,000 years after closure and the post-10,000-year period. The projected dose considers aleatory uncertainty for characteristics of the eruption such as number of waste packages intersected by the eruption, the fraction of waste packages intersected that are ejected, eruption power, wind direction, and wind speed. The mean, median, and 5th- and 95th-percentile curves in Figure F-11 show uncertainty in the value of the projected annual dose and consider epistemic uncertainty from incomplete knowledge of the behavior of the physical system during and after the disruptive event. The plots show that the mean projected dose for 10,000 years after closure is less than 0.0002 millirem and that for the post-10,000-year period is less than 0.0002 millirem. The median projected annual dose is less than 0.0001 millirem for the post-10,000-year period.

Figure F-12 shows the radionuclides that dominate the estimate of mean annual dose. Because transport of radionuclides to the location of the RMEI would be more rapid in the Volcanic Eruption Modeling Case than in the Igneous Intrusion Modeling Case, short-lived radionuclides would contribute to the estimate of the mean annual dose estimate. Figure F-12 shows that short-lived radionuclides (for example, cesium-137 and plutonium-238) would be significant contributors at early times, but their contributions would drop rapidly because of radioactive decay. At 300 years, americium-241 would dominate the total, but its contribution would diminish rapidly after about 1,000 years, also due to decay.

These short-lived radionuclides would be able to reach the location of the RMEI before they decayed because atmospheric transport to this location would be relatively rapid. After 1,000 years, plutonium-239 and -240 would become dominant contributors until approximately 100,000 years after closure, when radium-226 and thorium-229 would become the primary dose contributors for the remainder of the post-10,000-year period.

F.4.2.2 Seismic Scenario Class

The Seismic Scenario Class describes future performance of the repository system in the event of seismic activity that could disrupt the repository system. The Seismic Scenario Class represents the direct effects of vibratory ground motion and fault displacement associated with seismic activity. Indirect effects of drift collapse are also considered in this Scenario Class. The Seismic Scenario Class considers the effects of the seismic hazards on drip shields and waste packages. The Seismic Scenario Class also takes into account changes in seepage, waste package degradation, and flow in the Engineered Barrier System that might be associated with a seismic event. The conceptual models and abstractions for the mechanical response of Engineered Barrier System components to seismic hazards at a geologic repository are summarized in *Seismic Consequence Abstraction* (DIRS 176828-SNL 2007, all).

The Seismic Scenario Class estimates the mean annual dose due to a presumed seismic event and takes into account the relevant processes that come into play and affect system performance. The Seismic Scenario Class is represented by two modeling cases, the Seismic Ground Motion Modeling Case and the Seismic Fault Displacement Modeling Case.

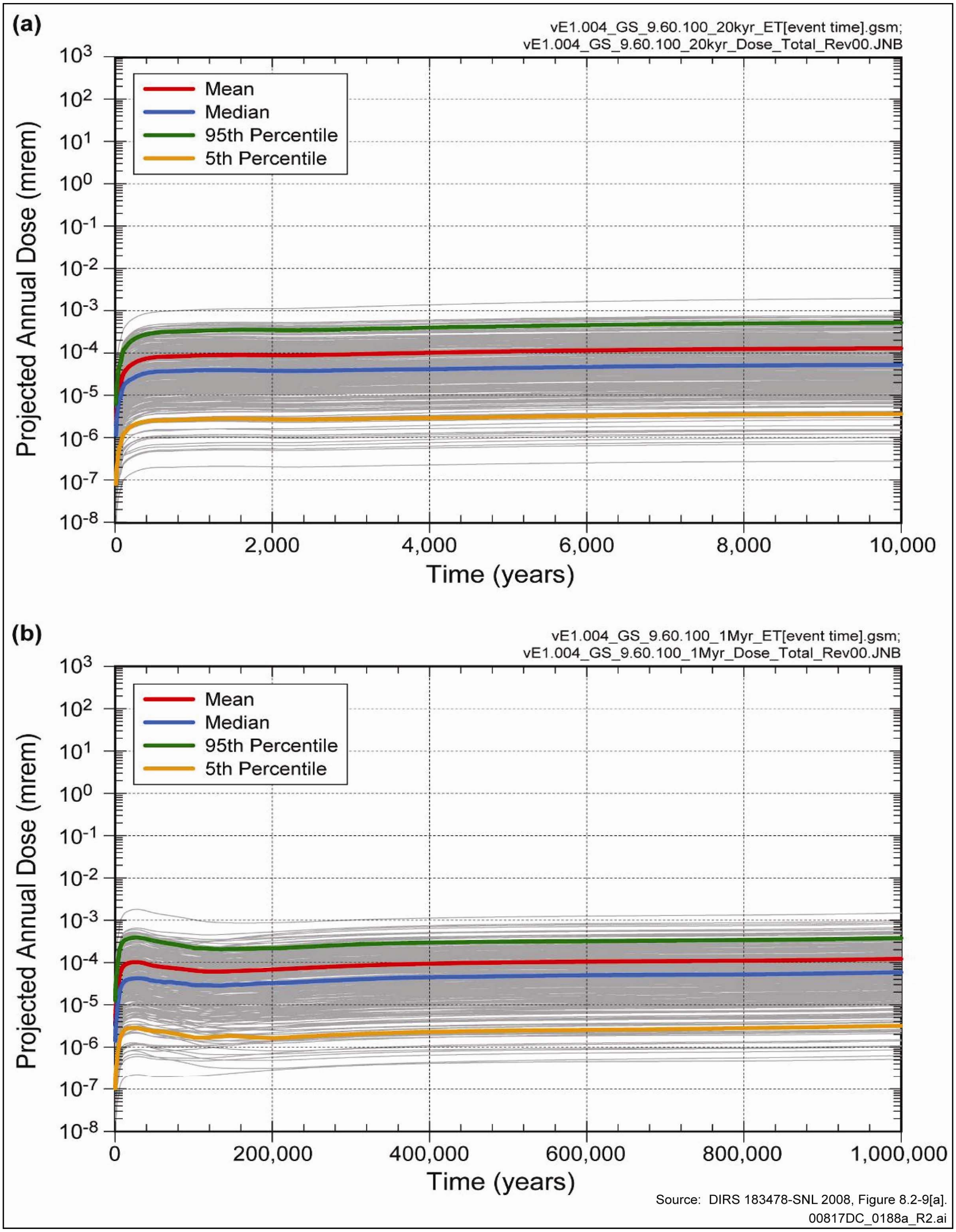

Figure F-11. Projected annual dose for the Volcanic Eruption Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

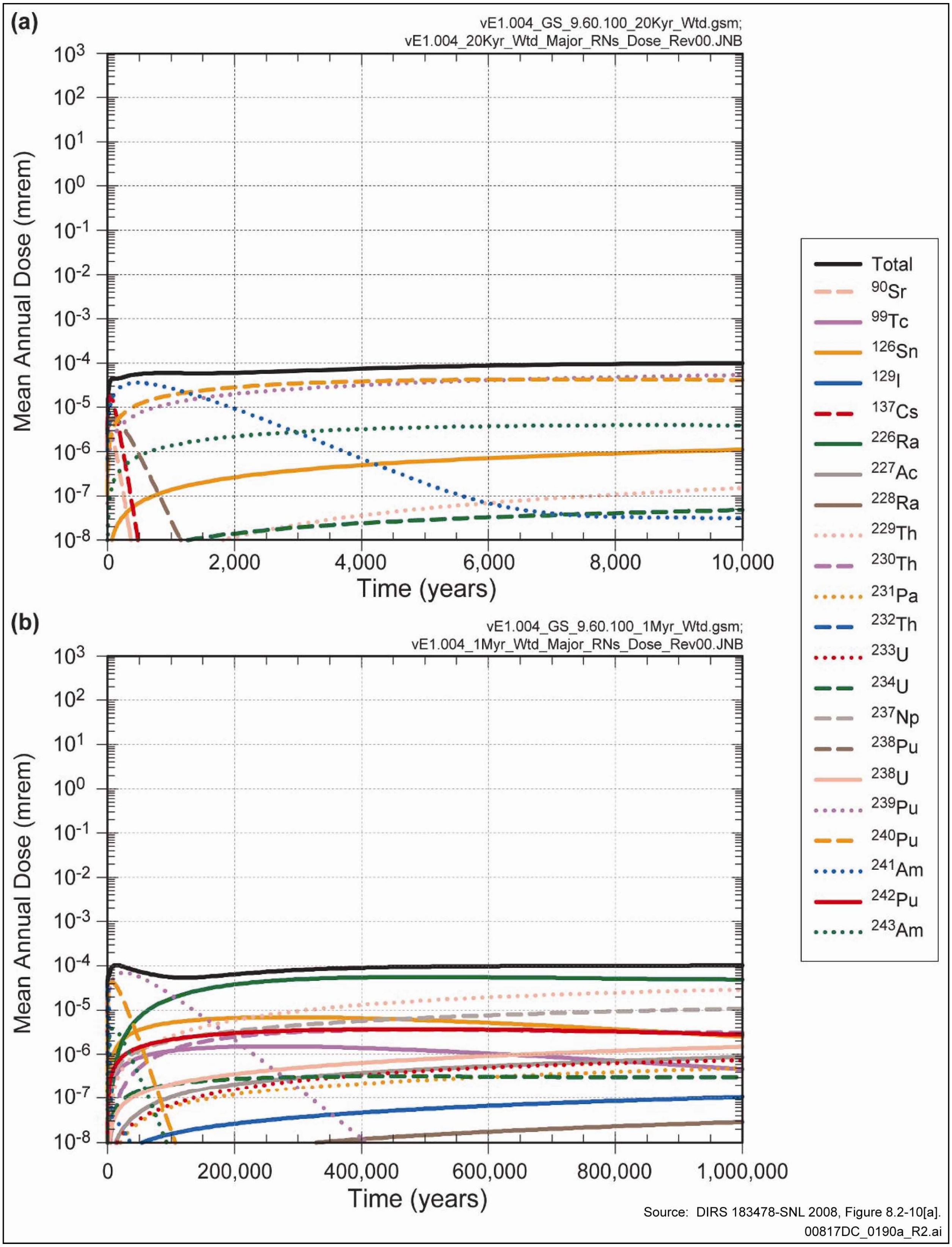

Figure F-12. Mean annual dose histories of major radionuclides for the Volcanic Eruption Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

F.4.2.2.1 *Seismic Ground Motion Modeling Case*

The first modeling case represents drip shields and waste packages that fail from mechanical damage associated with seismic vibratory ground motion. This modeling case is referred to as the Seismic Ground Motion Modeling Case. The Seismic Ground Motion Modeling Case includes the following degradation mechanisms on the drip shields and waste packages: stress corrosion cracking of the waste packages, tearing or rupture, and collapse of drip shield supports. Figure F-13 presents projected annual dose histories for the Seismic Ground Motion Modeling Case for the first 10,000 years after closure and the post-10,000-year period. The projected dose takes into account aleatory uncertainty associated with characteristics of future events such as number of events, times of events, and the events peak ground velocity.

The mean, median, and 5th- and 95th-percentile curves on Figure F-13 show uncertainty in the value of the projected annual dose and consider epistemic uncertainty due to incomplete knowledge of the behavior of the physical system during and after the disruptive event. These figures show that the mean projected annual dose for 10,000 years after closure is approximately 0.2 millirem and for the post-10,000-year period is approximately 1.1 millirem. The median projected dose for the post-10,000-year period is less than 0.4 millirem. Coarse time steps for the period after 200,000 years result in observable jumps in expected annual dose at 200,000, 300,000, 500,000, and 700,000 years. The spikes in the results correspond to the calculated increase in the number of waste packages that fail due to stress corrosion cracking. The large time steps after 100,000 years, in combination with the sensitivity of the crack growth rate to the stress intensity factor (a function that is evaluated at the beginning of each time step), could cause the crack growth rate to change dramatically from a small value for the time step in which the crack initiates to a much larger value at the beginning of the next time step, which resulted in almost immediate penetration of many cracks and waste package failures.

The results in Figure F-14 show the radionuclides that would contribute most to the estimate of mean projected annual dose for the Seismic Ground Motion Case. Figure F-14a shows that technetium-99, carbon-14, iodine-129, and chlorine-36 would dominate the estimate of the mean for 10,000 years after closure. Figure F-14b shows that radionuclides technetium-99, iodine-129, selenium-79, and plutonium-239 would dominate the estimate of the mean for the post-10,000-year period up to about 250,000 years. Plutonium-242, iodine-129, and neptunium-237 become dominant radionuclides later in time. The influence of carbon-14 would decrease completely by 100,000 years because of radioactive decay. The codisposal waste packages would be the primary waste packages damaged during the first 10,000 years after closure because the commercial spent nuclear fuel waste packages would be much stronger and more failure resistant. The commercial spent nuclear fuel waste packages would be more robust than codisposal waste packages because they include two inner stainless-steel vessels instead of one; the inner vessel and its lids similar to the codisposal waste packages, and an additional stainless-steel TAD canister. The predominant mechanism that would cause damage to codisposal and commercial spent nuclear fuel waste packages would be small cracks (stress corrosion cracking) that resulted in releases from the waste packages by diffusion (DIRS 183478-SNL 2008, Section 6.6). Diffusive transport of dissolved radionuclides through the cracks would be sufficiently high that these radionuclides would contribute significantly to the total mean projected annual dose.

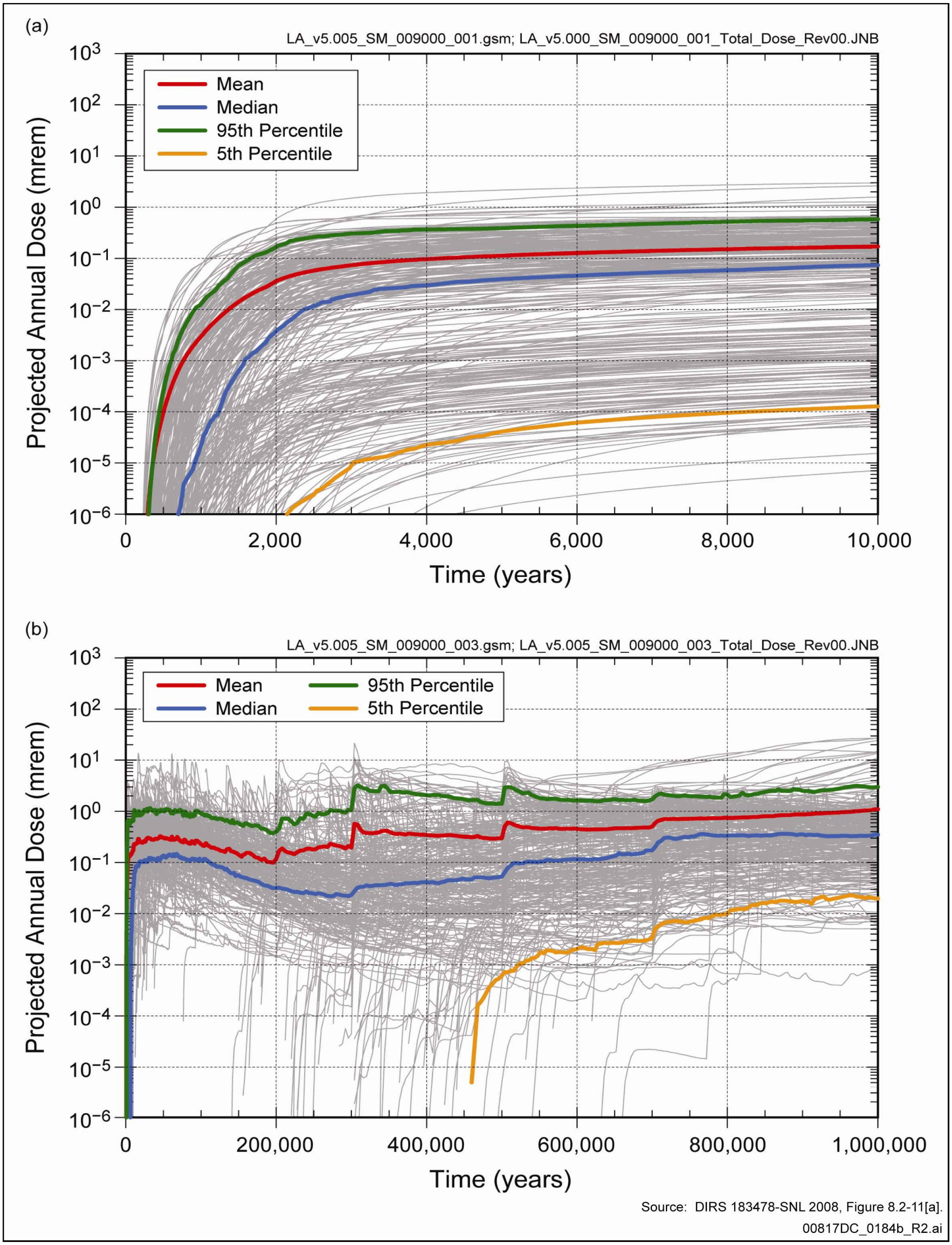

Figure F-13. Projected annual dose for the Seismic Ground Motion Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

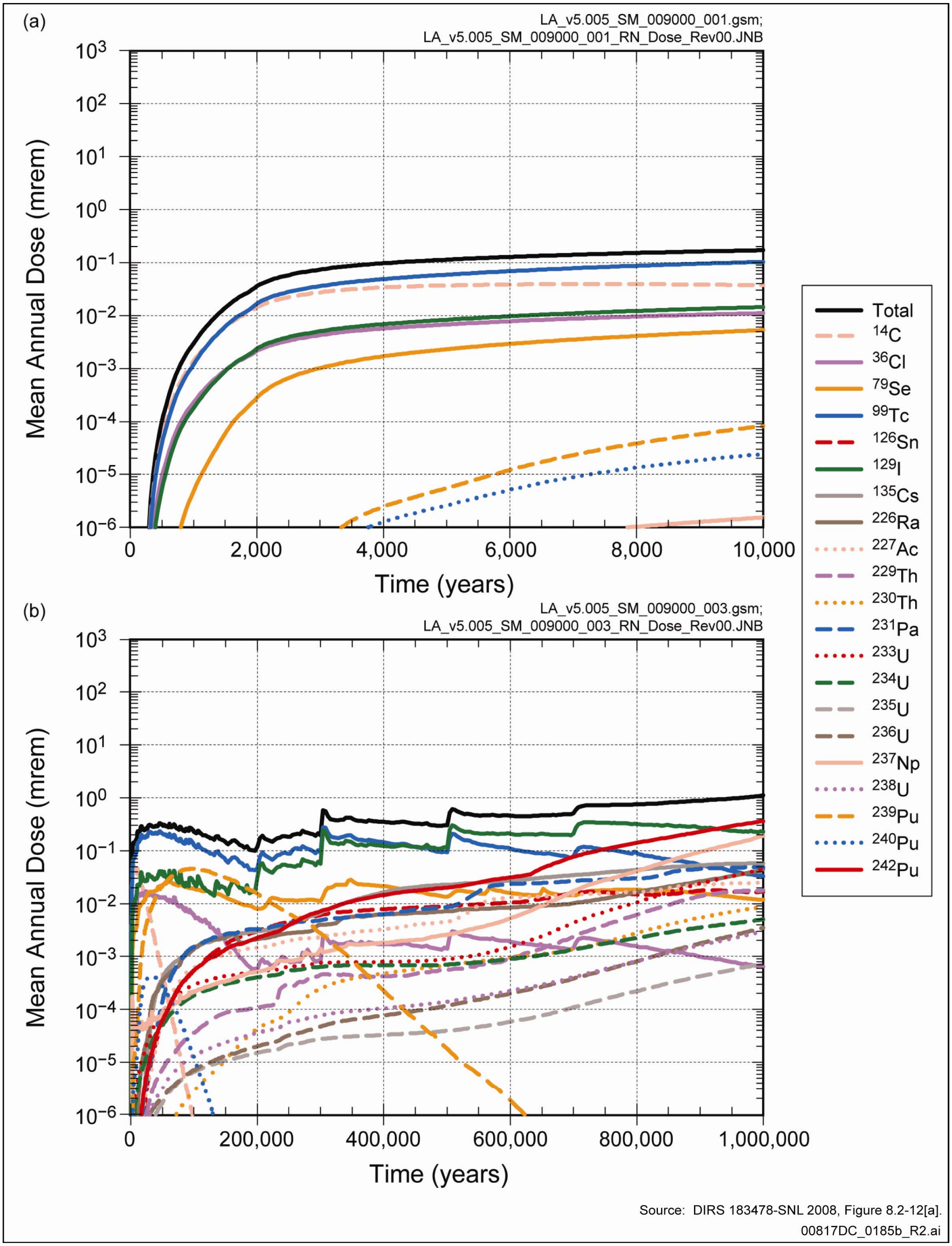

Figure F-14. Mean annual dose histories of major radionuclides for the Seismic Ground Motion Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

F.4.2.2.2 *Seismic Fault Displacement Modeling Case*

The Seismic Fault Displacement Modeling Case includes disruption of waste packages and drip shields by the displacement of faults, as well as local corrosion failure of waste packages onto which water would flow through drip shield breaches. As Section F.2.11.4 notes, the annual probability of an event that resulted in fault displacement is on the order of 1×10^{-7} to 1×10^{-8}. This is low enough that it is reasonable to expect no occurrence of fault displacement during the post-10,000-year period.

Figure F-15 shows the projected annual dose histories for the Seismic Fault Displacement Modeling Case for the first 10,000 years after closure and the post-10,000-year period. The projected dose accounts for aleatory uncertainty for characteristics for the number of disrupted drip shields and waste packages. The mean, median, and 5th- and 95th-percentile curves on Figure F-15 show uncertainty in the value of the projected annual dose, taking into account epistemic uncertainty from incomplete knowledge of the behavior of the physical system during and after the disruptive event. These figures show that the mean projected annual dose for 10,000 years after closure would be less than 0.002 millirem and for the post-10,000-year period would be less than 0.02 millirem. The median projected dose for the post-10,000-year period would be approximately 0.01 millirem.

The results in Figure F-16 show the radionuclides that contribute most to the estimate of mean projected annual dose. Figure F-16a shows that plutonium-239, iodine-129, and plutonium-240 would dominate the estimate of the mean projected annual dose for 10,000 years after closure. Figure F-16b shows that plutonium-239, radium-226, and technetium-99 would dominate the mean at 100,000 years and that plutonium-242, radium-226, and neptunium-237 would dominate the mean for the remainder of the post-10,000-year period.

F.4.3 TOTAL IMPACTS FROM ALL SCENARIO CLASSES

DOE evaluated the total impacts of postclosure repository performance by summing the annual projected doses histories for each modeling case. The result is the total projected annual dose to the RMEI from the waste packages that would fail in the Nominal, Early Failure, Igneous, and Seismic Scenario classes.

Equation F-1 represents the distribution for total expected annual dose $\overline{D}_T(\tau,\boldsymbol{e_i})$ as a function of time τ:

$$\overline{D}_T(\tau,\boldsymbol{e_i}) \quad \overline{D}_N(\tau,\boldsymbol{e_i})+\overline{D}_{EF}(\tau,\boldsymbol{e_i})+\overline{D}_I(\tau,\boldsymbol{e_i})+\overline{D}_S(\tau,\boldsymbol{e_i}) \qquad \text{(Equation F-1)}$$

where e_i denotes a realization or sampling of epistemic uncertainty i (Chapter 5, Section 5.3.4.2.1) and $i = 1, 2, \ldots$. The quantity $\overline{D}_N(\tau,\boldsymbol{e_i})$ is the expected annual dose resulting from nominal processes, and quantities $\overline{D}_{EF}(\tau,\boldsymbol{e_i})$, $\overline{D}_I(\tau,\boldsymbol{e_i})$, and $\overline{D}_S(\tau,\boldsymbol{e_i})$ are the expected values of annual dose resulting from the occurrence of early failure, and igneous and seismic events, respectively.

Equation F-1 shows the calculation of total mean annual dose as the sum of mean annual dose for each scenario class. In turn, the mean annual dose for each scenario class is the sum of mean annual doses for the modeling cases comprising the scenario class, with the exception of the Seismic Scenario Class. The Nominal and Seismic Scenario classes were combined for the calculation of dose during the post-10,000-year period because the nominal processes of corrosion affect the susceptibility of the engineered

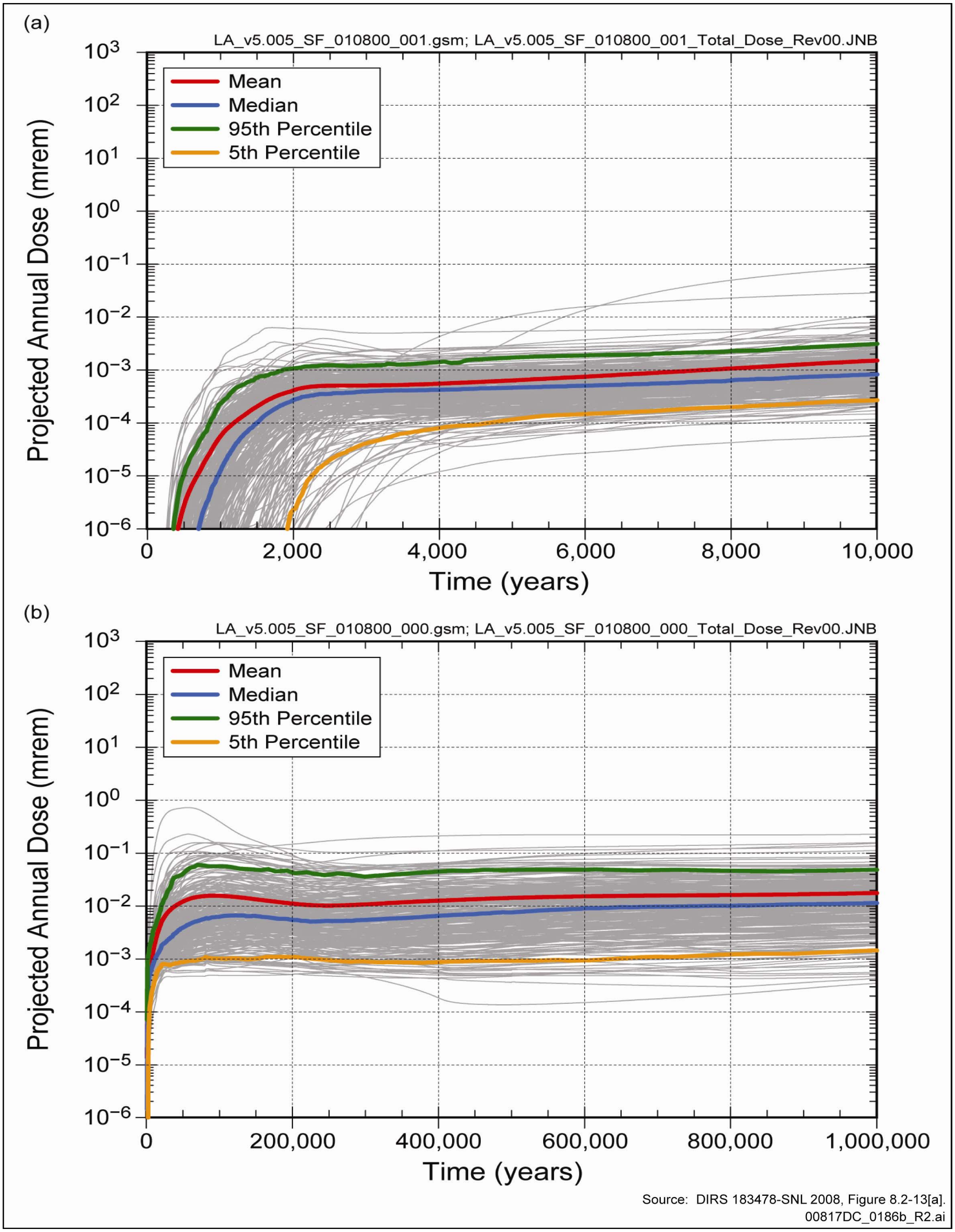

Figure F-15. Projected annual dose for the Seismic Fault Displacement Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

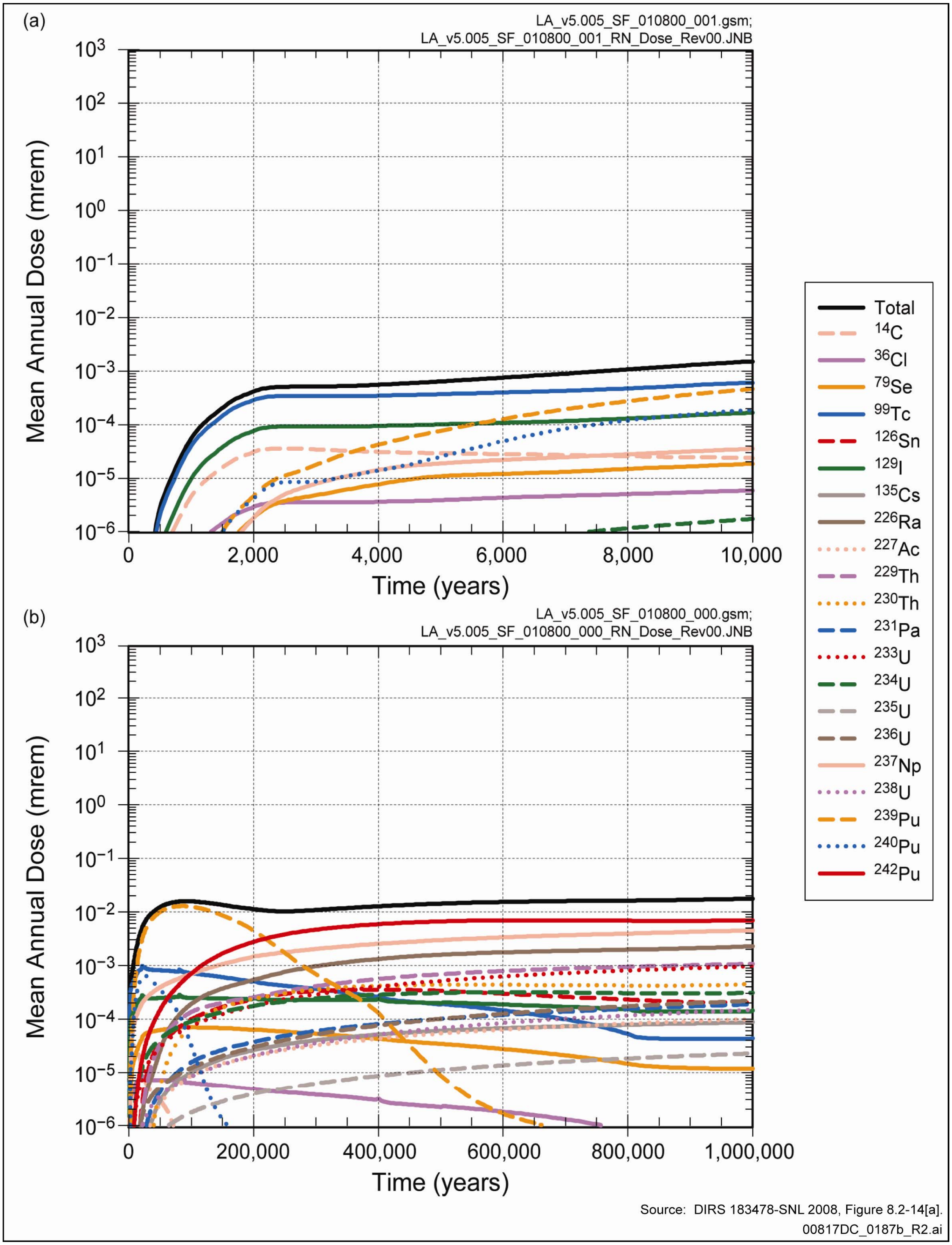

Figure F-16. Mean annual dose histories of major radionuclides for the Seismic Fault Displacement Modeling Case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

barrier to damage from seismic events. For the post-10,000-year period, the expected annual dose for the Nominal and the Seismic Scenario classes are combined and computed as:

$$\overline{D}_N(\tau,\boldsymbol{e_i})+\overline{D}_S(\tau,\boldsymbol{e_i}) \quad \overline{D}_{GM}(\tau,\boldsymbol{e_i})+\overline{D}_{FD}(\tau,\boldsymbol{e_i}) \qquad \text{(Equation F-2)}$$

where $\overline{D}_{GM}(\tau,\boldsymbol{e_i})$ is the expected annual dose from the combined effects of seismic ground motion events and nominal corrosion processes and $\overline{D}_{FD}(\tau,\boldsymbol{e_i})$ is the expected annual dose from seismic fault displacement events.

Figures 5-4 and 5-6 (Chapter 5, Section 5.5) show representations of the epistemic distributions for $\overline{D}_T(\tau,\boldsymbol{e_i})$ for the first 10,000 years and the post-10,000-year period, respectively, where each individual dose curve or history in the figures corresponds to expected time histories over aleatory uncertainty. The mean and median histories derive directly from this distribution, as shown on the figures. For example, the total mean annual dose, $\overline{\overline{D}}_T(\tau)$, is calculated as the expected value of $\overline{D}_T(\tau,\boldsymbol{e_i})$ as given by Equation F-3:

$$\overline{\overline{D}}_T(\tau)\cong\frac{1}{N}\sum_{i\ 1}^{N}\overline{D}_T(\tau,\boldsymbol{e_i}) \qquad \text{(Equation F-3)}$$

This approach does not enable the display of uncertainty, but it illustrates the important modeling case contributors to the total mean annual dose. Figure F-17 shows the total mean annual dose and the median annual dose contributions from each modeling case. The contribution to total annual dose from the Nominal Scenario Modeling Case is included in the Seismic Ground Motion Modeling Case and, therefore, is not shown separately in this figure. The figure shows that for the first 10,000 years after closure (Figure F-17a) and post-10,000-year period (Figure F-17b), the Seismic Ground Motion and Igneous Intrusion Modeling cases, respectively, would provide the largest contributions to the total annual dose. *Total System Performance Assessment Model/Analysis for the License Application* (DIRS 183478-SNL 2008, Section 6.1) provides the details for the development of Equation F-1, the distribution for D_T $(\tau, \boldsymbol{e_i})$, and the calculation of the mean and median total annual doses.

F.4.4 COMPARISON WITH GROUNDWATER PROTECTION STANDARDS

DOE excluded unlikely natural processes and events from the performance calculations to evaluate conformance with groundwater protection, as required by EPA rule (40 CFR 197.30 and 197.31). The standards require compliance with three groundwater protection performance measures:

1. Maximum annual concentration of radium-226 and -228 in a representative volume of 3.7 million cubic meters (3,000 acre-feet) of groundwater.

2. Gross alpha activity (excluding radon and uranium) in the representative volume of groundwater.

3. Dose to the whole body or any organ of a human for beta- and photon-emitting radionuclides in groundwater.

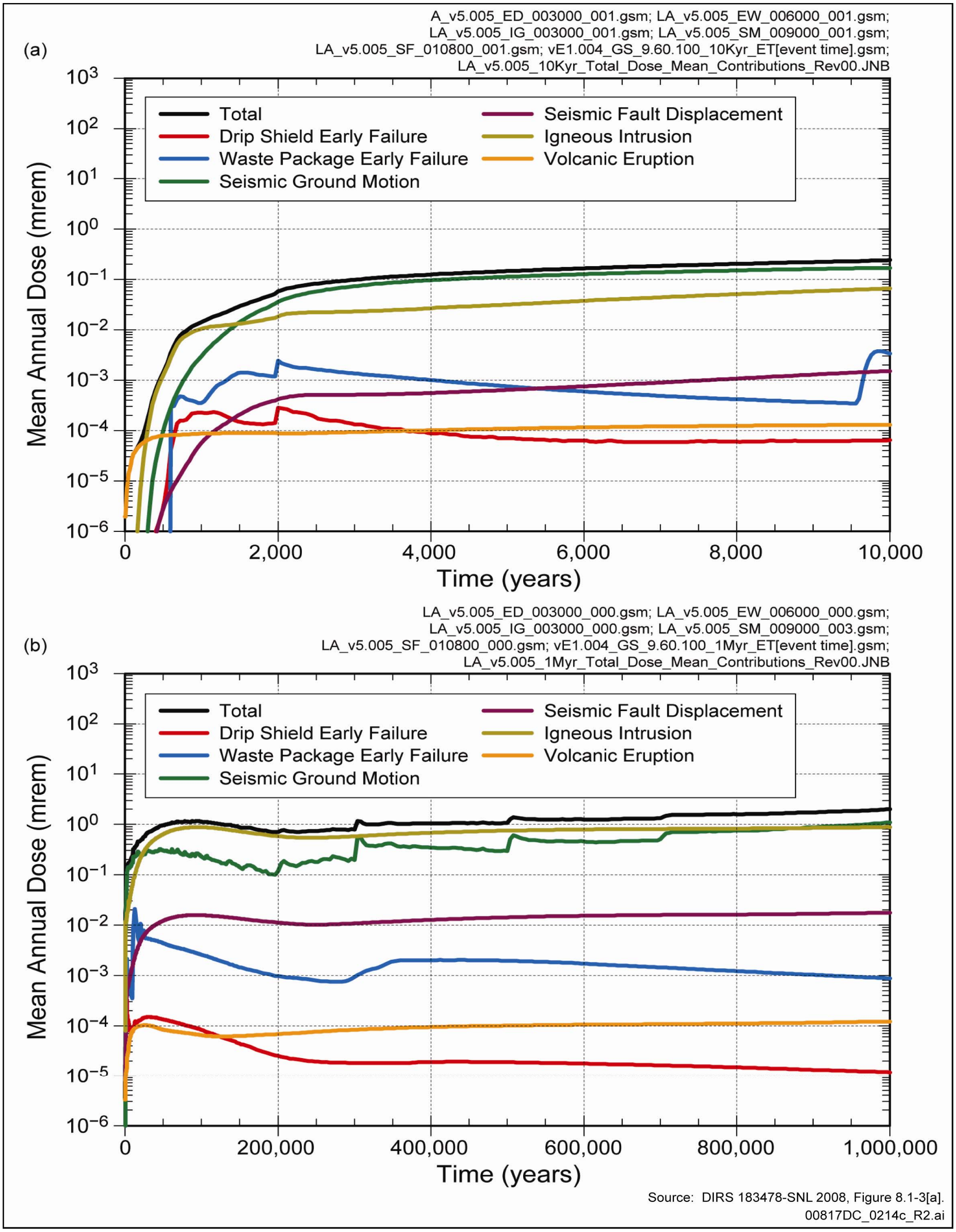

Figure F-17. Total mean annual dose and median annual doses for each modeling case for (a) the first 10,000 years after repository closure and (b) the post-10,000-year period.

The calculations for the first two performance measures apply to releases from natural sources and from the repository at the same location as the RMEI.

The exposed individual would consume 2 liters (0.5 gallon) per day from the representative volume of groundwater. In the scenario, groundwater would be withdrawn annually from an aquifer that contained less than 10,000 milligrams per liter of total dissolved solids, and that was centered on the highest concentration in the plume of contamination at the same location as the RMEI.

Figures F-18 and F-19 show projected total radium (radium-226 plus radium-228) and mean activity concentrations of gross alpha activity (excluding radon and uranium), respectively, in the representative volume of groundwater for the Proposed Action inventory. The projected mean concentration for total radium in the first 10,000 years after closure is less than 2×10^{-7} picocurie per liter. The projected mean concentration of gross alpha activity during that period is less than 7×10^{-5} picocurie per liter. Naturally occurring background radionuclide concentrations are illustrated in the figures but were not included in the calculations because the calculated values would be negligible in comparison with background concentrations (about 0.5 picocurie per liter) up to 10,000 years after closure.

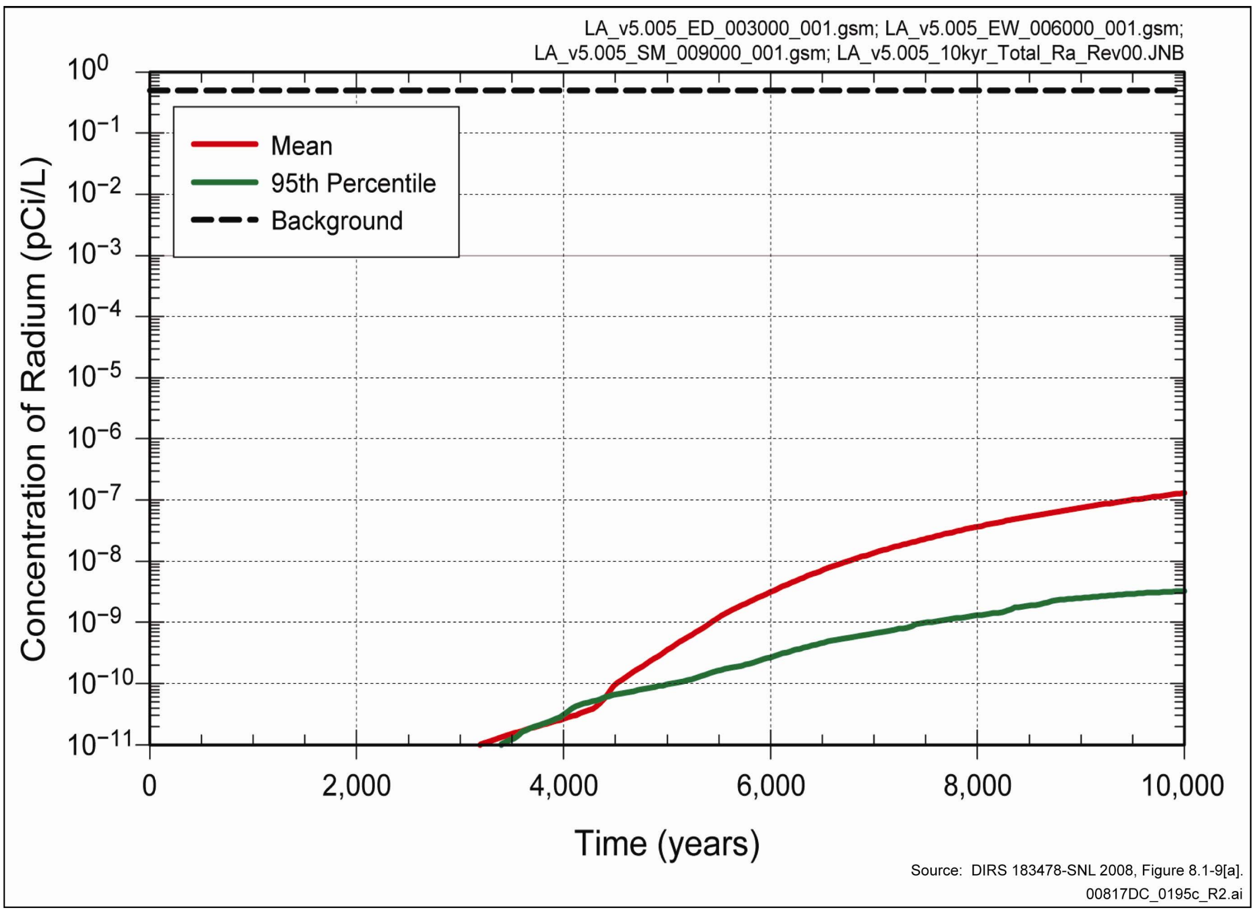

Figure F-18. Combined radium-226 and -228 activity concentrations, excluding natural background, for likely features, events, and processes using nominal, early failure, and seismic ground motion damage processes.

Figure F-20 shows calculated whole-body and organ annual doses due to beta- and photon-emitting radionuclides in the groundwater. DOE calculated these annual doses from the concentrations of all of the beta- and photon-emitting radionuclides in the TSPA-LA model. The concentrations of these radionuclides were evaluated in terms of total annual release from the repository dissolved in the representative water volume. Figure F-20 shows the mean annual drinking water doses for thyroid and whole body (without their organ-dose weighting factors). The organ with the highest annual dose would be the thyroid, and the projected mean annual drinking water dose to the thyroid is less than 0.3 millirem.

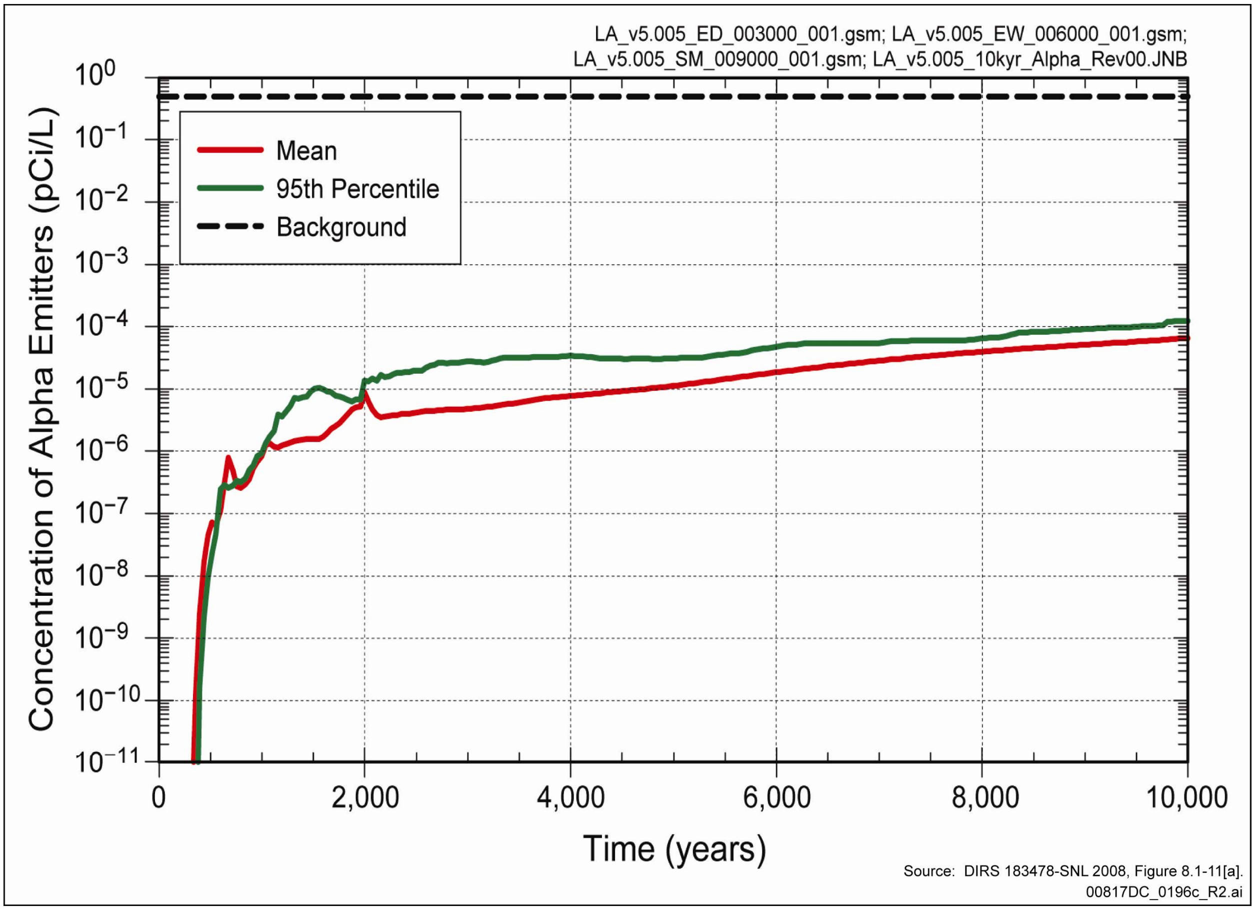

Figure F-19. Combined activity concentrations of all alpha emitters (including radium-226 but without radon and uranium isotopes), excluding natural background, for likely features, events, and processes using nominal, early failure, and seismic ground motion damage processes.

The whole-body dose in the figure accounts for the effect on all organs and includes the organ dose weighting factors. The projected mean annual drinking water dose to the whole body in this case is about 0.06 millirem.

Table F-4 summarizes the standards and projected impacts in relation to the groundwater protection standard. In addition, it lists the combined whole-body or organ doses over 10,000 years for the total of all beta- and photon-emitting radionuclides.

F.5 Waterborne Chemically Toxic Material Impacts

DOE did not use the TSPA-LA model to estimate the postclosure impacts from waterborne chemically toxic materials because the model is unsuitable for this purpose. Rather, it used a bounding analysis to estimate impacts. Waterborne chemically toxic materials are products of the degradation of repository and waste package construction materials. The following sections describe the development of a final list of materials of concern from the larger list in Section F.3 and the bounding analysis DOE performed on those materials of concern.

F.5.1 SCREENING ANALYSIS

The Yucca Mountain FEIS contains a discussion of the screening analysis, which this Repository SEIS summarizes and incorporates by reference (DIRS 155970-DOE 2002, pp. I-52 to I-59). DOE eliminated copper and manganese from further consideration due to bounding concentration limits from low

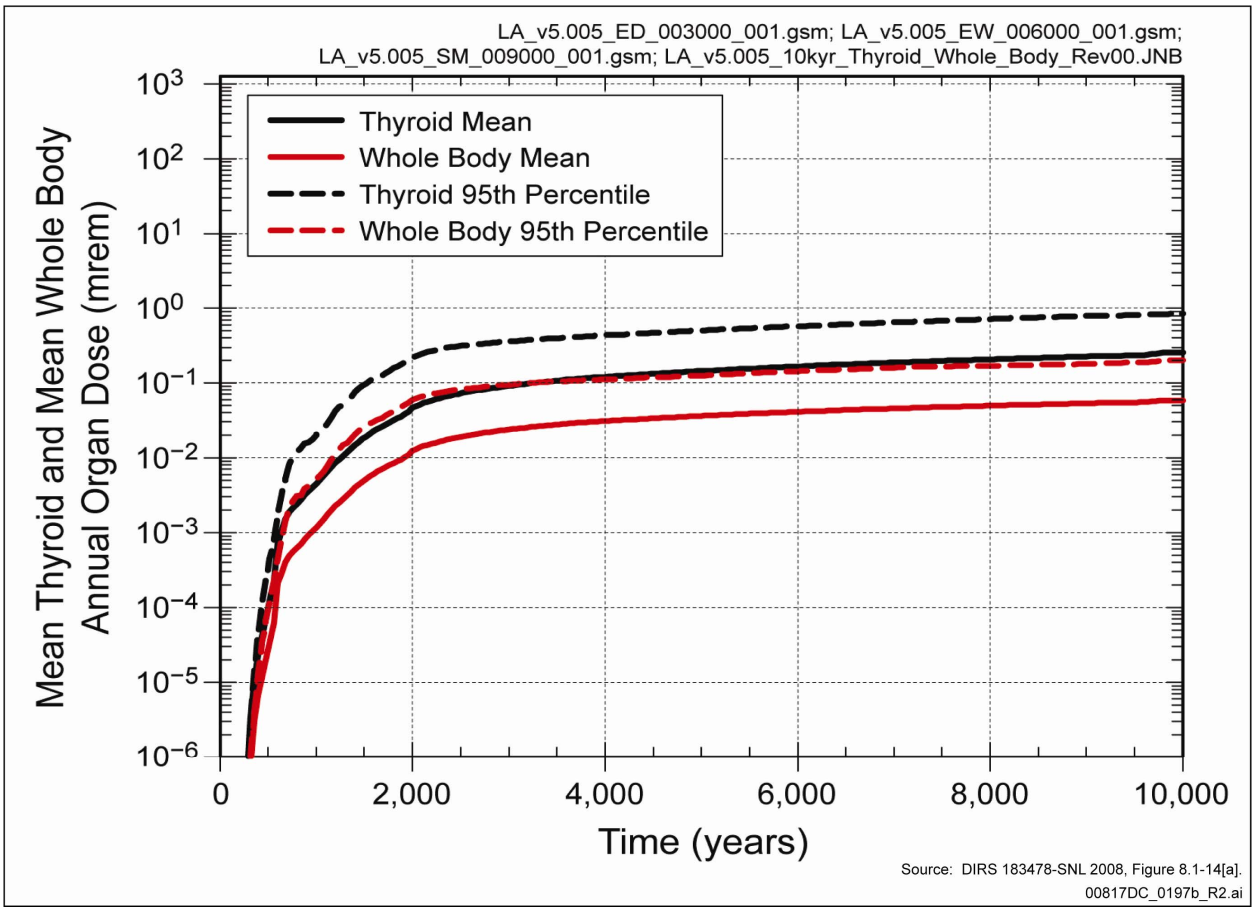

Figure F-20. Mean annual drinking water dose from combined beta and photon emitters for likely features, events, and processes using the nominal, early failure, and seismic ground motion damage processes.

Table F-4. Comparison of estimated postclosure impacts at the RMEI location with groundwater protection standards during 10,000 years after repository closure for likely features, events, and processes using the nominal, early failure, and seismic ground motion damage processes.

Radionuclide or type of radiation	EPA limit	Mean	95th-percentile
Combined radium-226 and -228 (picocuries per liter)	5	1.3×10^{-7}	9.9×10^{-8}
Gross alpha activity (including radium-226 but excluding radon and uranium) (picocuries per liter)	15	6.7×10^{-5}	3.2×10^{-3}
Combined beta-and photon-emitting radionuclides (millirem per year to the whole body or any organ), based on drinking 2 liters (0.5 gallon) of water per day from the representative volume	4	0.3	0.8

Source: DIRS 183478-SNL 2008, all.
Note: Radium values do not include natural background radiation.
EPA = U.S. Environmental Protection Agency.
RMEI = Reasonably maximally exposed individual.

solubility. Since completion of the FEIS, an additional substance, palladium, has been added to the repository structural materials; specifically, as 0.15 to 0.25 percent of the titanium alloy in the drip shields. Palladium is not listed as a hazardous material in the EPA Integrated Risk Information System and, therefore, is excluded from further analysis due to its lack of toxicity.

Since the Yucca Mountain FEIS was completed, there has been additional research on the corrosion behavior of many of the metals within the repository. One aspect of this research was a shift in the conclusions concerning speciation of chromium evolving from corrosion of materials such as Alloy 22 and various grades of stainless steel. At the time of the FEIS, it was conservatively assumed that corrosion of these materials would result in a dominant valence +6 form of chromium [chromium(VI)]. More recent work has revealed that the chemical conditions within the repository will result in corrosion products dominated by chromium valence +3 [chromium(III)] (DIRS 169860-BSC 2004, Section 6.8.1.2).

Chromium(VI) is a highly soluble form of chromium, while chromium(III) is a nearly insoluble form. This means that as chromium is dissolved from the corroding materials, it rapidly precipitates as a mineral (Cr_2O_3, various hydroxides, or other species depending on pH and other chemicals present). The solubility of chromium(III) is dependent on pH but is generally very low. The repository drift environment would have a pH ranging from about 6 to 12 (DIRS 169860-BSC 2004, Figure 6.13-26). Geochemical simulations in the repository drift environment showed chromium(III) solubility would be less than 1×10^{-3} milligram per liter for pH of 6 to 12 (DIRS 169860-BSC 2004, Figure 6.8-4). Another study in the general literature showed measurements of solubility in a high pH environment at temperatures up to 288 degrees Celsius (°C) [550 degrees Fahrenheit (°F)] on the order of 5×10^{-6} milligram per liter (DIRS 181408-Ziemniak et al. 1998, all). Another study with solutions ranging from pH 6 to 12 found the solubility to be 5×10^{-3} milligram per liter (DIRS 182718-Rai and Rao 2005, Figure 4). All of these values fall well below the maximum contaminant level goal of 0.1 milligram per liter set by EPA (40 CFR 141.51). As water leaves the repository and is captured in the representative volume, it would have concentrations much less than the source values at the repository due to the dilution in the representative volume of 3.7 million cubic meters (3,000 acre-feet) per year. Thus, chromium can be expected to have a concentration in the representative volume of much less than the maximum contaminant level goal. Therefore, chromium was excluded from further analysis.

F.5.2 BOUNDING CONSEQUENCE ANALYSIS FOR CHEMICALLY TOXIC MATERIALS

DOE evaluated waterborne chemically toxic materials (molybdenum, nickel, and vanadium) because the screening analysis (Section F.6.1) indicated that the repository could release such materials into groundwater in substantial quantities and that these materials could represent a potential human health impact. This section contains a bounding calculation for concentrations in the biosphere of these elements and shows that the estimated impacts are low enough to preclude a need for more detailed modeling.

F.5.2.1 Assumptions

DOE applied the following assumptions to the bounding impact analysis for waterborne chemically toxic materials:

1. The general corrosion rate of Alloy 22 is for fresh water at 100°C (37.8°F) under expected bounding repository conditions; this does not include local corrosion because that mechanism would not release a significant amount of material.

2. The general corrosion rate of Stainless Steel Type 316NG is for fresh water at 50° to 100°C (122° to 212°F) under expected bounding repository conditions; this does not include local corrosion because that mechanism would not release a significant amount of material.

3. Drip shields do not effectively delay the onset of general corrosion of Alloy 22 in the outer barrier layer of waste packages or the emplacement pallets; the basis for this is conservatism.

4. Consistent with Assumptions 1, 2, and 3, exposed Alloy 22 and Stainless Steel Type 316NG in the drip shield rail, external surface of the waste packages, and emplacement pallets would be subject to corrosion at the same time.

5. Consistent with Assumptions 1, 2, and 3, all waste packages would be subject to general corrosion at the same time and would not experience variability in the time corrosion began.

6. A migration pathway for mobilized waterborne chemically toxic materials through the Engineered Barrier System to the vadose zone would exist at all times when general corrosion was in progress.

7. This bounding impact estimate neglected time delays, mitigation effects by sorption in rocks, and other beneficial effects of transport in the geosphere; the mass of mobilized waterborne chemically toxic materials would be instantly available at the biosphere exposure locations.

8. The concentration in groundwater was estimated by diluting the released mass of waterborne chemically toxic materials in the representative volume of 3.7 million cubic meters (3,000 acre-feet) of water per year.

9. Release rates of molybdenum, nickel, and vanadium would be equivalent to the corrosion loss of Stainless Steel Type 316NG or Alloy 22 multiplied by the fraction of each element in the alloys.

F.5.2.2 Surface Area Exposed to General Corrosion

Corrosion of materials that contained molybdenum, nickel, and vanadium would occur over all exposed surface areas. This section describes the calculation of the total exposed surface area of Alloy 22 surfaces (drip shield rails, outer layer of waste packages, and portions of the emplacement pallets) and Stainless Steel Type 316NG surfaces (portions of the emplacement pallets and ground control structures).

Tables F-5 and F-6 summarize the calculation of the total exposed surface areas for Alloy 22 in the waste packages and drip shields, respectively, under the Proposed Action. Table F-7 summarizes the calculation of total exposed surface area for the Alloy 22 components of the emplacement pallets. The sum of exposed total surface areas for waste packages, drip shield rails, and emplacement pallet components fabricated from Alloy 22 (from Tables F-5 to F-7) would be 641,426 square meters (6.9 million square feet). This would be the area of Alloy 22 subject to general corrosion under the assumptions for this bounding impact estimate.

Table F-8 summarizes the calculation of the total exposed surface areas for the Stainless Steel Type 316NG DOE would use in the emplacement pallets for the Proposed Action.

The stainless-steel ground support components for the emplacement drifts in the proposed repository would consist of perforated steel sheets, friction-type rock bolts, and bearing plates. The estimated

Table F-5. Total exposed surface area of the Alloy 22 outer layer of all waste packages.

Waste package type	Number[a]	Outer diameter[b] (millimeters)[c]	Length[b] (millimeters)[c]	Surface area (square millimeters)[d]	Total surface area (square meters)[e]
21 PWR/44 BWR TAD	7,365	1,963	5,850	36,076,636	265,704
5 DHLW Short/1 DSNF Short	1,147	2,126	3,697	24,692,359	28,322
5 DHLW Long/1 DSNF Long	1,406	2,126	5,304	35,425,554	49,808
2 MCO/2 DHLW	149	1,831	5,279	30,366,160	4,525
5 DHLW Long/1 DSNF Short	31	2,126	5,304	35,425,554	1,098
HLW Long Only	679	2,126	5,304	35,425,554	24,054
Naval Short	90	1,963	5,215	32,160,625	2,894
Naval Long	310	1,963	5,850	36,076,636	11,184
Totals	11,177				387,589

a. Number of waste packages from DIRS 176937-DOE 2006, Table 2-11. The numbers of packages might vary slightly from those given in the application for construction authorization. The differences, if any, would be small and would have negligible effect on the results of the bounding metals release calculation.
b. Waste package data from DIRS 179710-BSC 2007, all; DIRS 179580-BSC 2007, all; DIRS 179955-BSC 2007, all; DIRS 180192-BSC 2007, all; DIRS-184410-BSC 2007, all; DIRS 179870-BSC 2007, all; DIRS 184405-BSC 2007, all; DIRS 175303-BSC 2007, all; DIRS 175304-BSC 2007, all; DIRS 175305-BSC 2007, all; DIRS 180180-BSC 2007, all; DIRS 180183-BSC 2007, all; DIRS 180184-BSC 2007, all; DIRS 180187-BSC 2007, all; DIRS 180188-BSC 2007, all; DIRS 180189-BSC 2007, all; DIRS 182714-Morton 2007, all.
c. To convert millimeters to inches, multiply by 0.03937.
d. To convert square millimeters to square inches, multiply by 0.00155.
e. To convert square meters to square feet, multiply by 10.764.

BWR = Boiling-water reactor.
DHLW = DOE high-level radioactive waste.
DSNF = DOE spent nuclear fuel.
MCO = Multicanister overpack.
PWR = Pressurized-water reactor.
TAD = Transportation, aging, and disposal (canister).

Table F-6. Total exposed surface area of the Alloy 22 rails for all drip shields under the Proposed Action inventory.

Drip shield component	Total waste package emplacement length[a] (meters)[b]	Mass[c] (kilograms)[d] per meter[b] of length	Thickness[c] (millimeters)[e]	Total surface area for repository[f] (square meters)[g]
Rail	62,736	19.9	10	28,732

a. Sum of the waste package lengths plus a 0.1-meter (4-inch) spacing between packages. Waste package data from DIRS 179710-BSC 2007, all; DIRS 179580-BSC 2007, all; DIRS 179955-BSC 2007, all; DIRS 180192-BSC 2007, all; DIRS 184410-BSC 2007, all; DIRS 179870-BSC 2007, all; DIRS 184405-BSC 2007, all; DIRS 175303-BSC 2007, all; ; DIRS 175304-BSC 2007, all; DIRS 175305-BSC 2007, all; DIRS 180180-BSC 2007, all; DIRS 180183-BSC 2007, all; DIRS 180184-BSC 2007, all; DIRS 180187-BSC 2007, all; DIRS 180188-BSC 2007, all; DIRS 180189-BSC 2007, all.

b. To convert meters to feet, multiply by 3.2808.

c. Rail mass and thickness from DIRS 180028-BSC 2007, all.

d. To convert kilograms to pounds, multiply by 2.205.

e. To convert millimeters to inches, multiply by 0.03937.

f. Surface area calculated for the wetted surfaces (top and sides) of the rail. Total surface area is 2 times the total length times mass per length divided by thickness divided by 8,690 kilograms per cubic meter.

g. To convert square meters to square feet, multiply by 10.764.

exposed surface area of the stainless-steel ground support components is 1,008,538 square meters (approximately 11 million square feet) for one side of the perforated plates, 12,225 square meters (approximately 1.3 million square feet) for one side of the bearing plates, and 276,376 square meters

Table F-7. Total exposed surface area of the Alloy 22 components for all emplacement pallets under the Proposed Action.[a]

Emplacement pallet component	Number of pieces	Thickness (meters)[b]	Mass (kilograms)[c]	Total surface area per pallet[d] (square meters)[e]
Plate 1	2	0.009525	84	4.06
Plate 2	2	0.017500	102	2.68
Plate 3	2	0.017500	87	2.29
Plate 4	4	0.009500	26	2.52
Plate 5	4	0.022200	8	0.33
Plate 6	4	0.022200	132	5.47
Plate 7	4	0.017500	1	0.03
Plate 8	4	0.017500	1	0.03
Plate 9	4	0.006400	4	0.58
Tube 2	4	0.006400	3	0.43
Tube 3	8	0.006400	6	1.73
Totals				
Surface area per pallet (square meters)[d]				20.14
Number of pallets				11,177
Surface area repository (square meters)[d]				225,105

a. Emplacement pallet details from DIRS 172623-BSC 2004, all; DIRS 172611-BSC 2004, all; DIRS 171783-BSC 2004, all; DIRS 185194-BSC 2004, all; DIRS 185195-BSC 2004, all; DIRS 170982-BSC 2004, all; DIRS 185196-BSC 2004, all; DIRS 185197-BSC 2004, all; DIRS 185198-BSC 2004, all; DIRS 172617-BSC 2004, all; DIRS 172615-BSC 2004, all.
b. To convert meters to feet, multiply by 3.28.
c. To convert kilograms to pounds, multiply by 2.205.
d. Area obtained by multiplying the total mass by 2 and by the number of pieces and dividing by a density of 8,690 kilograms per cubic meter (542 pounds per cubic foot) and dividing by the thickness. The area of edges is ignored.
e. To convert square meters to square feet, multiply by 10.764.

Table F-8. Total exposed surface area of the Stainless Steel Type 316NG components for all emplacement pallets under the Proposed Action inventory.

Emplacement pallet tubes	Number of pieces[a]	Length[a] (millimeters)[b]	Width[a] (millimeters)[b]	Number of sides[a]	Total surface area per average waste package[c] (square meters)[d]	Number of waste packages[e,f]	Total surface area repository (square meters)[d]
Long pallets	4	4,164	609.6	2	20.31[e]	10,030	203,709
Short pallets	4	2,466	609.6	2	12.03[f]	1,147	13,798
Totals						11,177	217,507

a. Emplacement pallet tube details from DIRS 185199-BSC 2004, all.
b. To convert millimeters to inches, multiply by 0.03937.
c. Calculated for area of all wetted rectangular sides.
d. To convert square meters to square feet, multiply by 10.764.
e. Waste package data from DIRS 180187-BSC 2007, all; DIRS 180188-BSC 2007, all; DIRS 180189-BSC 2007, all.
f. Only waste packages of type 5 DHLW Short/1 DSNF Short use the short pallets.

DHLW = DOE high-level radioactive waste.
DSNF = DOE spent nuclear fuel.

(approximately 3 million square feet) for the rock bolts (DIRS 182709-Duan 2007, all). To be conservative, both sides of the plates are assumed to contribute to the metal source term. Therefore, the total ground support area would be twice those of the perforated and bearing plates plus the rock bolt area. Therefore, the total stainless-steel area in the ground support would be 2,317,902 square meters (approximately 25 million square feet). These figures accounted for overlap of sheets and did not account for material facing the rock. The figures were increased for this analysis to include all surfaces with no reduction for overlap.

The total exposed stainless steel would be the sum of the pallets (Table F-8) plus the ground support, which would be 2,535,409 million square meters (approximately 28 million square feet).

F.5.2.3 General Corrosion Rates

DOE used the general corrosion rates of the alloys to calculate the dissolution rates of individual metals. These general corrosion rates are the same as those that DOE used in the TSPA-LA model.

F.5.2.3.1 *Alloy 22 Corrosion Rate*

This analysis used the mean value of the distribution of Alloy 22 corrosion rates. The mean was used as representative of a variety of locations and conditions of the waste packages.

The general corrosion rate of Alloy 22 in the TSPA-LA model is (DIRS 178519-SNL 2007, Equation 6-28):

$$\ln(R_T) = \ln(R_0) + C_1\left(\frac{1}{T_0} - \frac{1}{T}\right) \qquad \text{(Equation F-4)}$$

where

R_T	=	General corrosion rate (nanometers per year) at temperature T (kelvin)
R_0	=	General corrosion rate at 333.15 kelvin
T_0	=	333.15 kelvin
C_1	=	temperature coefficient (in kelvin).

The parameter C_1 is a truncated normal distribution (plus 2 or minus 3 standard deviations) with a mean of 4,905 kelvin and standard deviation of 1,413 kelvin (DIRS 181031-SNL 2007, Table 1-1). DOE used the mean value for this analysis.

R_0 is a two-parameter Weibull distribution. The scale parameter b for the distribution is 8.134 for 90-percent realizations (medium uncertainty), and the shape parameter c for the distribution is 1.476 for medium uncertainty (DIRS 181031-SNL 2007, Table 1-1).

The mean of the Weibull distribution is given by ReliaSoft Corporation (DIRS 182720-ReliaSoft 2007, all). Then:

$$R_0 = b\,\Gamma\left(1 + \frac{1}{c}\right) \qquad \text{(Equation F-5)}$$

where Γ is a gamma function. Then:

$$R_0 = 8.134\,\Gamma\left(1 + \frac{1}{1.476}\right) = 8.134\,\Gamma(1.677) \qquad \text{(Equation F-6)}$$

where $\Gamma(1.677) = 0.905$ so that $R_0 = 7.36$ nanometers per year.

Let $T = 373.15$ kelvin (100°C); then, substitute into Equation F-4:

$$\ln(R_T) \quad \ln(7.36) + 4905\left(\frac{1}{333.15} - \frac{1}{373.15}\right) \quad 3.5756 \qquad \text{(Equation F-7)}$$

Then $R_T = 35.7$ nanometers (0.0000014 inch) per year. For the bounding calculations, DOE used this rate for Alloy 22 general corrosion to estimate the release of the component metals.

F.5.2.3.2 *Corrosion Rate of Stainless Steel*

DOE used the mean stainless-steel corrosion rates for the TSPA-LA model (DIRS 169982-BSC 2004, Table 7-1, p. 7-1). The mean was used as representative of a variety of locations and types of materials over the entire repository. The mean corrosion rate for Stainless Steel Type 316NG in fresh water at 50° to 100°C (122° to 212°F) would be 0.248 micrometer (0.0000242 inch) per year.

F.5.2.4 Dissolution Rates

DOE calculated the rate of dissolution of waterborne chemically toxic materials as the product of the surface area exposed to general corrosion, the general corrosion rate, and the weight fraction of the alloy for the toxic material of interest. Alloy 22 consists of, among other elements, 14.5 percent (maximum) molybdenum, 57.2 percent nickel, and 0.35 percent vanadium (DIRS 104328-ASTM 1998, all). Stainless Steel Type 316NG is essentially the same as Stainless Steel Type 316L, which consists of, among other elements, 12 percent nickel, and 2.5 percent molybdenum, with no vanadium (DIRS 102933-CRWMS M&O 1999, p. 13).

Table F-9 lists the calculation of the bounding mass dissolution rates for the Proposed Action.

Table F-9. Bounding mass dissolution rates from Alloy 22 and Stainless Steel Type 316NG components from general corrosion for the Proposed Action.

Alloy	Total exposed surface area in repository (square meters)[a]	General corrosion rate (meters per year)[b]	Alloy release volume (cubic meters per year)[c]	Alloy density (grams per cubic meter)[d]	Bounding mass dissolution rate (grams per year)[e]			
					Alloy	Molybdenum	Nickel	Vanadium
Alloy 22	641,426	3.57×10^{-8}	0.023	8,690,000	199,870	28,981	114,326	700
316NG	2,535,409	2.48×10^{-7}	0.629	7,980,000	5,017,676	125,442	602,121	0
Totals						154,423	716,447	700

a. To convert square meters to square feet, multiply by 10.764.
b. To convert meters to feet, multiply by 3.2808.
c. To convert cubic meters to cubic feet, multiply by 35.314.
d. To convert grams per cubic meter to pounds per cubic foot, multiply by 0.0000624.
e. To convert grams to ounces, multiply by 0.035274.

F.5.2.5 Summary of Bounding Impacts

DOE based the bounding maximum concentration on the release rate of the source materials and the representative volume for dilution EPA prescribes in the proposed regulations at 40 CFR Part 197. Dilution of the bounding release rates in Section F.6.2.4 for molybdenum, nickel, and vanadium in the prescribed representative volume of water (3.7 million cubic meters, or exactly 3,000 acre-feet per year) for calculation of groundwater protection impacts for waterborne radioactive materials resulted in the bounding concentration in groundwater at exposure locations for these chemically toxic materials (Table F-10).

Table F-10. Bounding concentrations of waterborne chemical materials.

Material	Maximum bounding concentration (milligrams per liter)
Molybdenum	0.042
Nickel	0.19
Vanadium	0.00019

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 (DIRS 148228-EPA 1999,all).

In order to put these concentrations in perspective, Table F-11 presents a comparison of the intake from the maximum bounding concentrations in Table F-10 with the oral reference dose for each of these materials. Table F-11 lists the intakes by chemical under the assumption of water consumption of 2 liters (0.5 gallon) per day by a 70-kilogram (154-pound) person and the relevant oral reference dose.

Table F-11. Intake of waterborne chemical materials of concern based on maximum bounding concentrations listed in Table F-10 compared with oral reference doses (milligrams per kilogram of body mass per day).

Material	Oral reference dose	Intake[a]
Molybdenum	0.005[b]	0.0012
Nickel	0.02[c]	0.0054
Vanadium	0.007[d]	0.0000054

a. Assumes a daily intake of 2 liters (0.5 gallon) per day by a 70-kilogram (154-pound) individual.
b. Source: DIRS 148228-EPA 1999, all.
c. Source: DIRS 148229-EPA 1999, all.
d. Source: DIRS 103705-EPA 1997, all.

Because the bounding concentrations of molybdenum, nickel, and vanadium in groundwater yield intakes well below the respective oral reference doses, there was no further need to refine the calculation to account for physical processes that would further reduce concentration of these elements during transport in the geosphere.

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