Appendix A

Options to Elements of the Proposed Action

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A. OPTIONS TO ELEMENTS OF THE PROPOSED ACTION

This is a new appendix since 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* (DIRS 155970-DOE 2002, all) (Yucca Mountain FEIS) was completed. It describes options to elements of the Proposed Action presented in Chapter 2, Section 2.1 of this *Draft 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-S1D) (Repository SEIS). It evaluates these options in terms of how the potential environmental impacts would differ from what the U.S. Department of Energy (DOE or the Department) anticipates from implementation of elements of the Proposed Action in Chapter 4, Section 4.1, and the Similar Actions in Section 4.3, 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; and
- Extended monitoring period.

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 maximize the use of treated 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 19,000 liters (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 any non-surface 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. However, 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 as a result of installation of a wastewater treatment facility, there would be no additional socioeconomic impacts. Therefore, 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 an additional conventional large-capacity septic system. In addition, a wastewater treatment facility would entail a higher level of regulatory compliance and monitoring in comparison to a conventional septic system; examples would include 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 the commercial spent nuclear fuel in TAD canisters.

During the SEIS public scoping process, DOE received comments from the nuclear industry and others that asked what would happen if less than 90 percent of the commercial spent nuclear fuel arrived at the repository 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 were 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 nondisposable 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 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 included 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

Nominally, 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 150 percent from the baseline case evaluated in Chapter 4 of this Repository SEIS. If fuel was not packaged in TAD canisters at the generator sites, it would be packaged at the repository and therefore would result in no changes in the long-term impacts or performance of the repository.

As stated above, the Department would construct an additional Wet Handling Facility rather than a third Canister Receipt and Closure Facility in the geologic repository operations area. Therefore, 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 the water would be continually filtered and maintained. 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 and maintain the chemistry of the pool as low-level radioactive wastes. 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 99.9 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 impacts to 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 the additional facility would cause 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 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 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 exact 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 illustrate another way 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 were:

- 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 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.

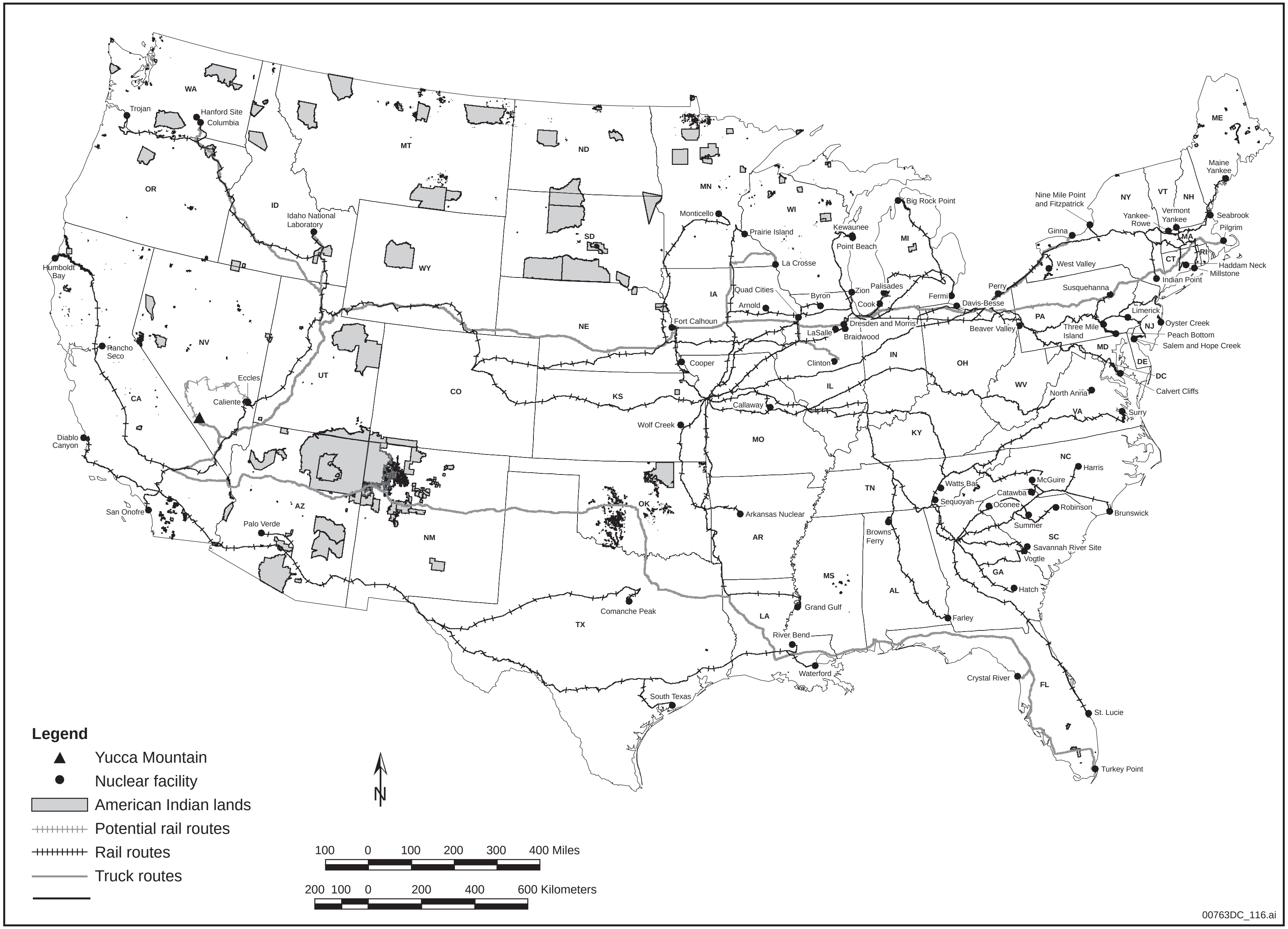


Figure A-1. Representative rail and truck transportation constrained routes if DOE selected the Caliente rail corridor in Nevada.

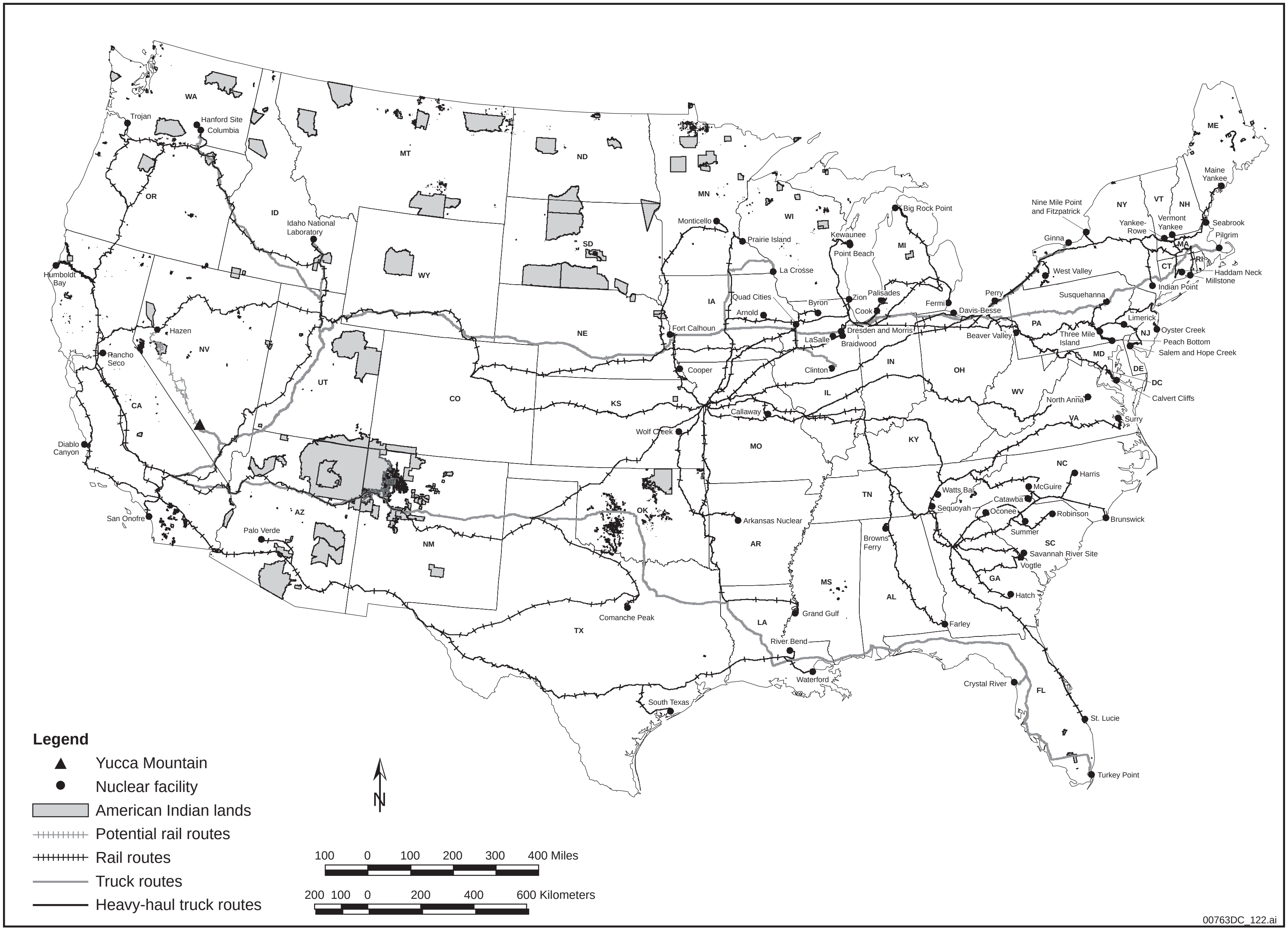


Figure A-2. Representative rail and truck transportation constrained routes if DOE selected the Mina rail corridor in Nevada.

These fatalities included 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 that would reside in the 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 this percentage is not based on historical precedent like that 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 periods, this sensitivity analysis addresses those periods. Impacts during the monitoring or closure period 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 on U.S. Highway 95 in Nye County, particularly during the repository construction phase. Before construction, as described in Chapter 4, Section 4.3, DOE would move the access road southeast to coincide with the Nevada State Route 373 intersection and provide acceleration and deceleration lanes. Based on current projections of traffic volumes in the vicinity of the intersection, however, no additional actions would be required to maintain adequate levels of service prior to repository construction.

A.4.1 SOCIOECONOMIC IMPACTS

The evaluation in Chapter 4, Section 4.1.6 assumed that 80 percent of the proposed repository site workers would live in Clark County and included 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 the 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 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 periods, this analysis addresses 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 period, the Proposed Action would directly employ about 2,590 workers. About 1,860 of these workers, who 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 (engineering, safety analysis, safety and health, and others). 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 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.

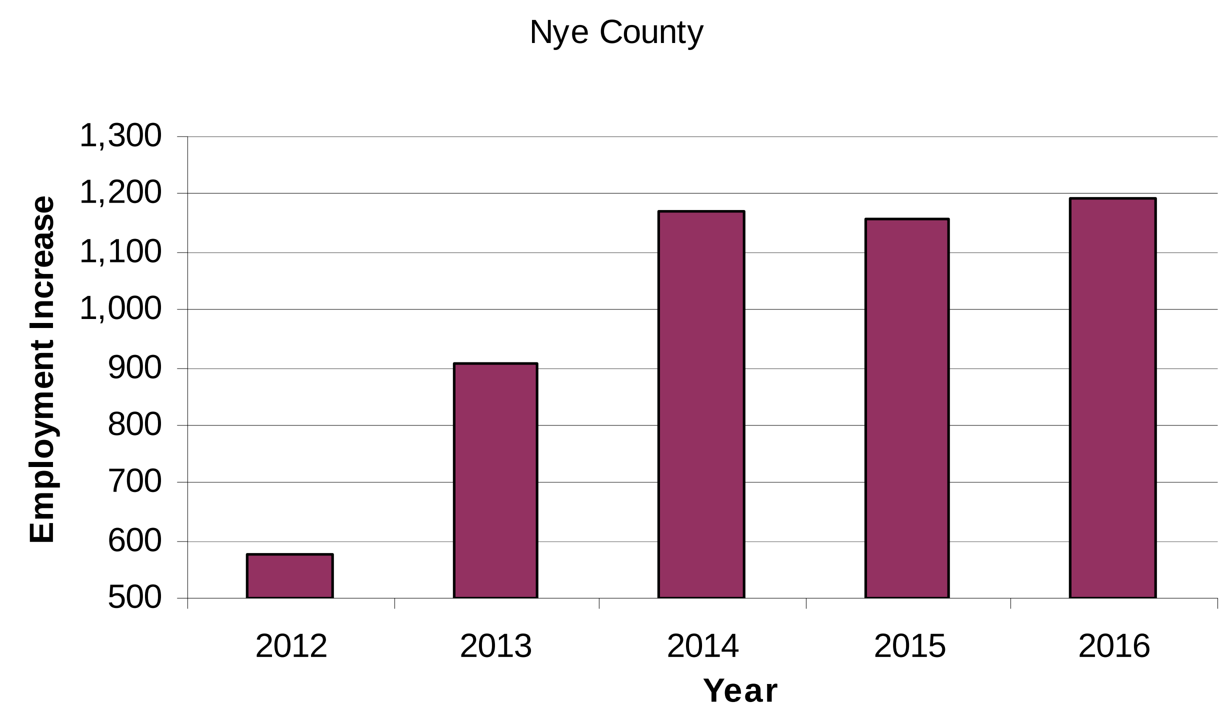
Table A-2. Direct project employment during construction, 2012 to 2016.

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 have been rounded to three significant figures.
a. Includes current workers.

During the construction 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 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.



Source: DIRS 182646-Bland 2007, all

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 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 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 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.

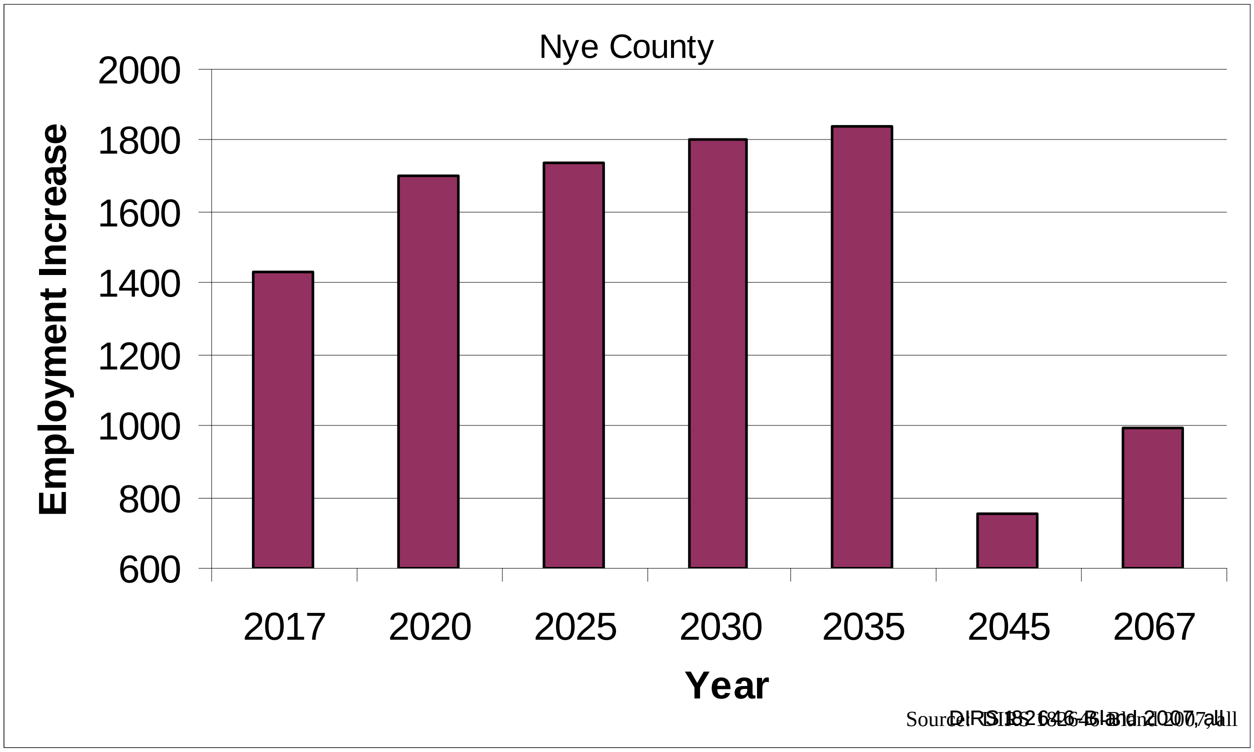


Figure A-4. Changes in Nye County employment from repository operations, 2017 to 2067.

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 required. The Proposed Action would contribute jobs to the Nye County economy during the entire operations 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 here – from 21,700 in 2020 to 26,300 in 2040. Employment in years 2040 and 2041 is very similar, and repository employment after 2040 is too small to affect the county as noted in the text. The narrative focuses on the 20-year period when the repository employment impacts the county, 2020 to 2040. Table A-3 lists the baseline and the changes in employment for 2020 to 2040 in Nye County. Although the operations 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 operations 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 have been rounded to three significant figures.

b. Percentages have been 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 period would be professional and technical services positions, followed by federal civilian service positions, retail trade positions, jobs in food and beverage places, 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 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 period years. 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 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 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 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.

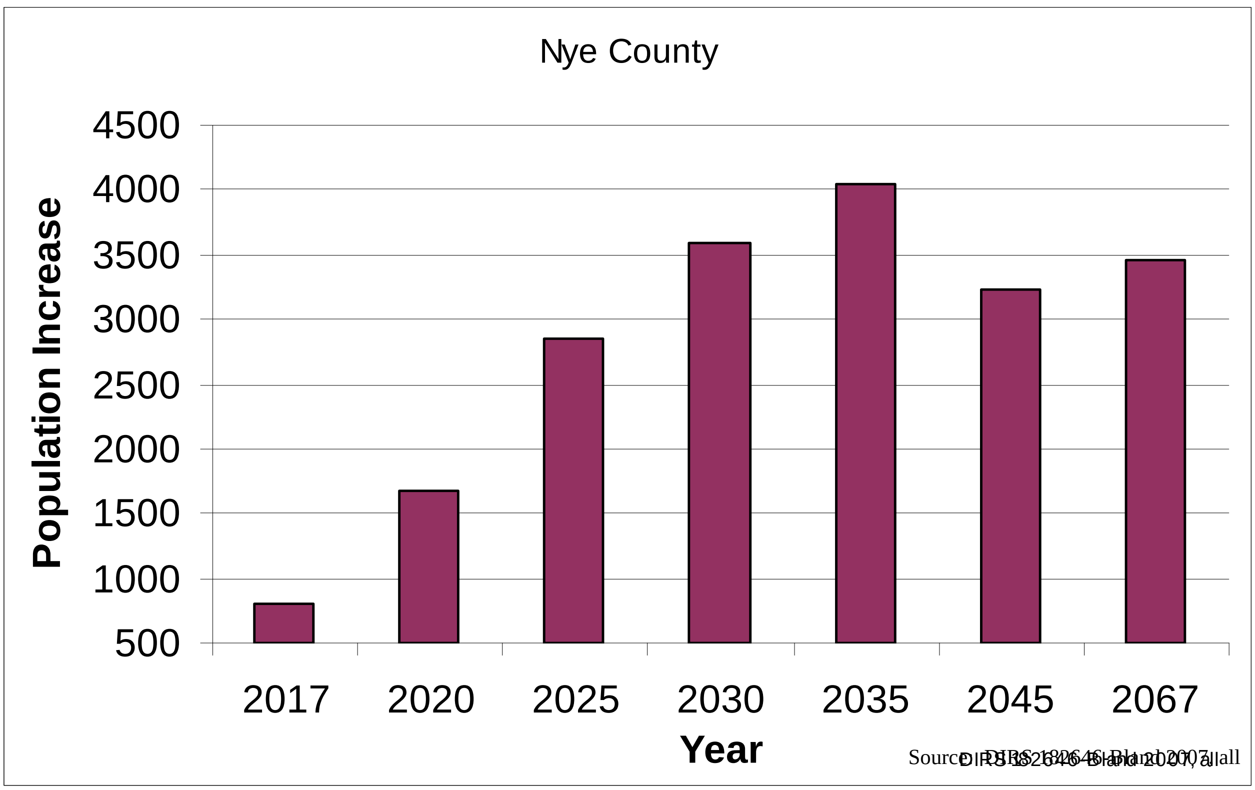


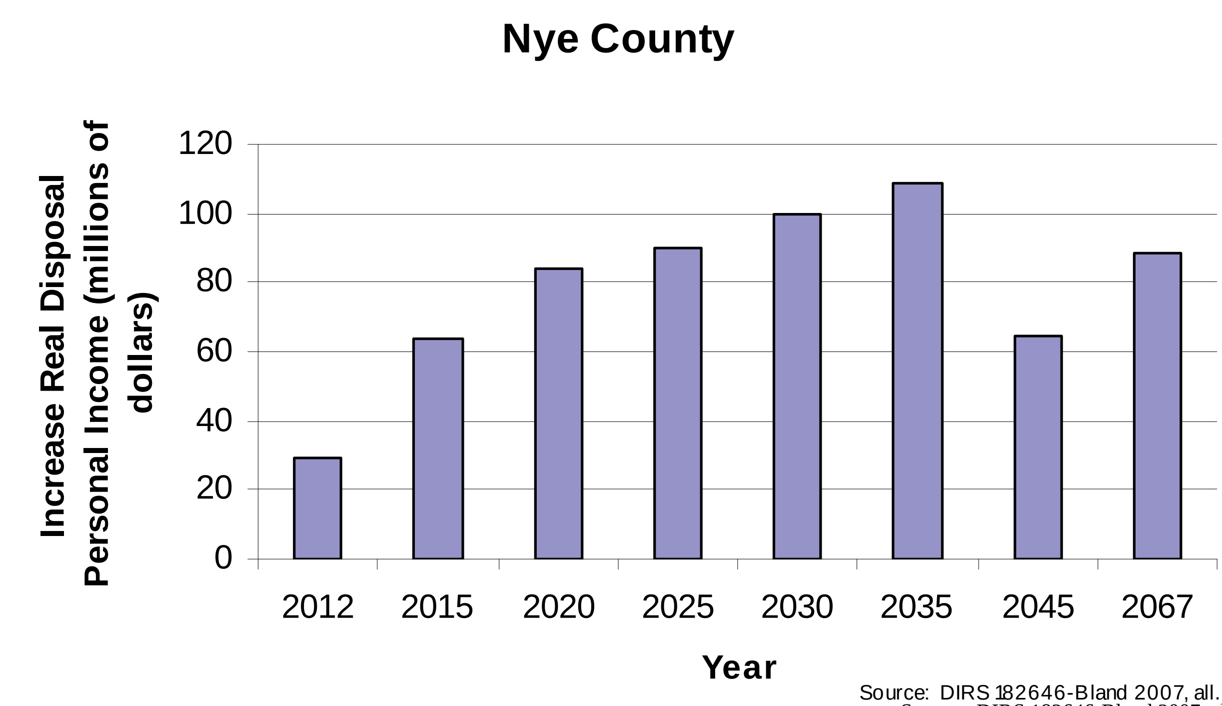
Figure A-5. Changes in Nye County population from repository operations, 2017 to 2067.

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 periods. (Values are in 2006 dollars.)

A.4.1.3.1 *Impact to Economic Measures During Construction*

Increases in real disposable personal income (after-tax income) in the 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 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.



Source: DIRS 182646-Bland 2007, all

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

During the construction 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.

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, about $10. Figure A-8 shows estimated changes in spending by state and local governments for the construction and operations periods.

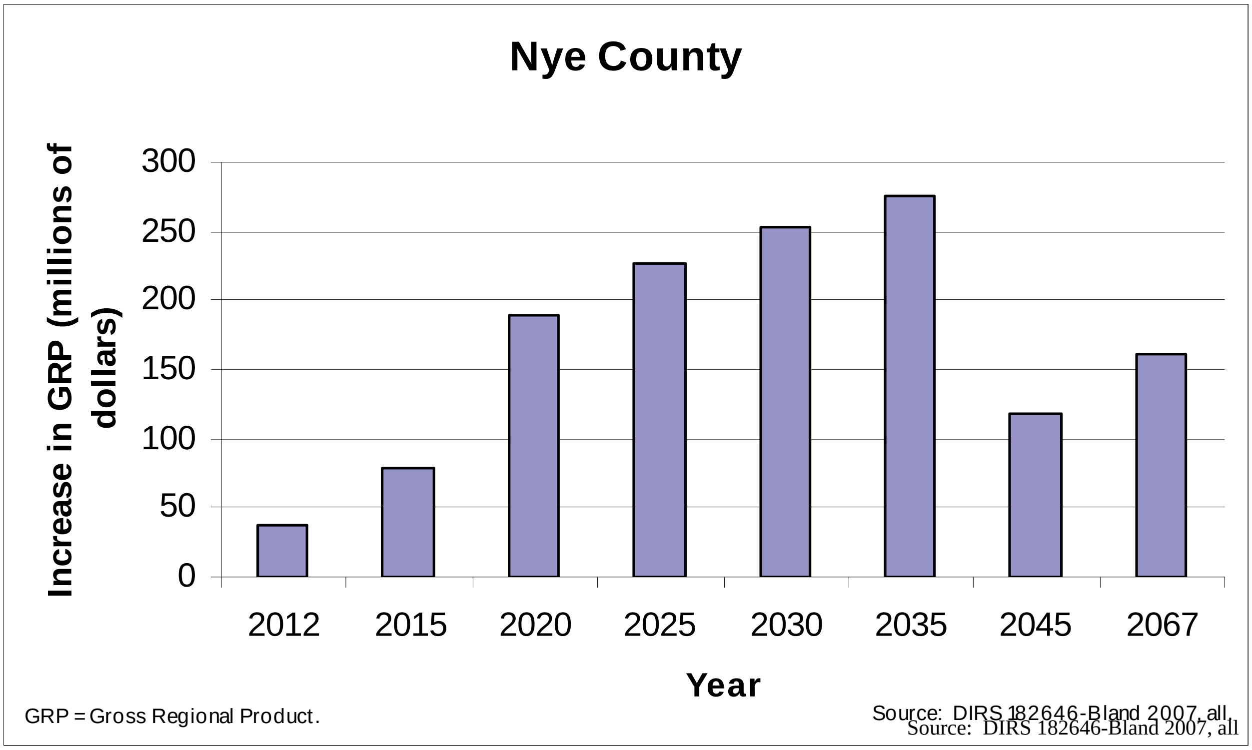


Figure A-7. Changes in Gross Regional Product in Nye County from repository activities during construction and operations periods, 2012 to 2067.

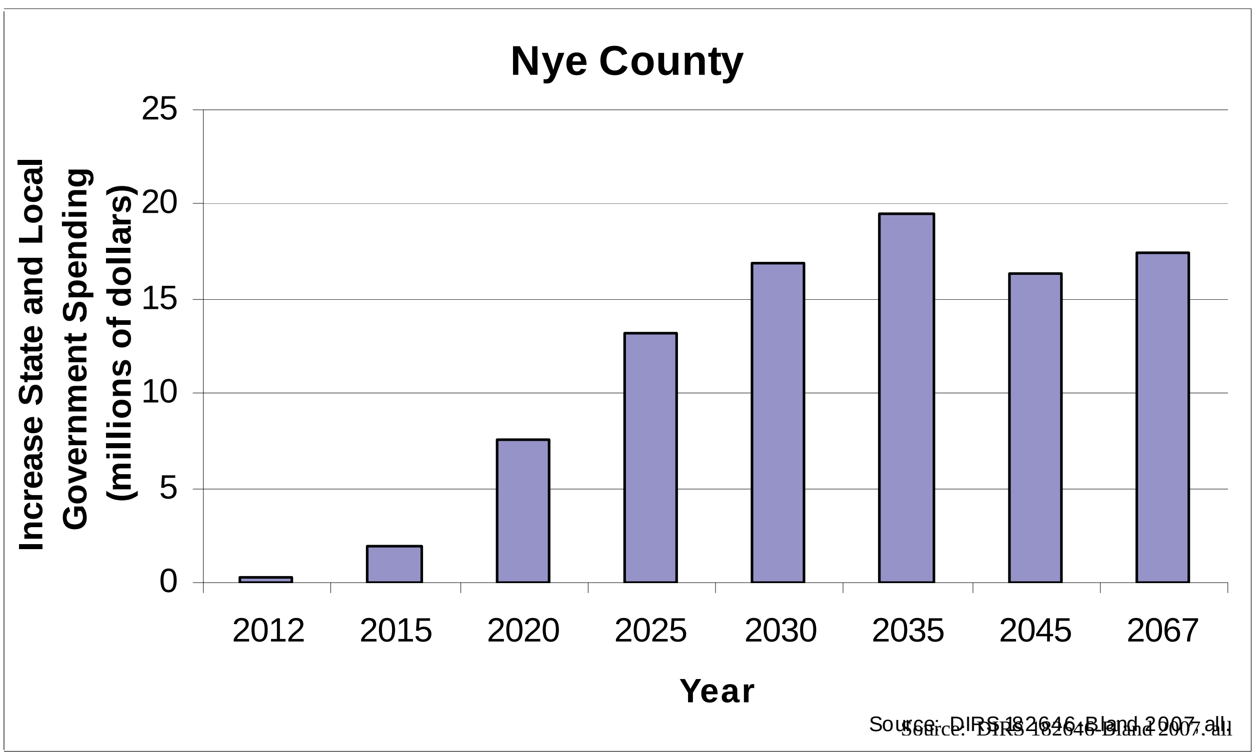


Figure A-8. Changes in spending by state and local governments in Nye County from repository activities during construction and operations periods, 2012 to 2067.

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 periods under the 80-percent assumption.

During the operations 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 the 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 form $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 the 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 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 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 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 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 right 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. Southern Nye County, particularly Pahrump, would experience increased demand for public services. However, because the changes in population in the county would occur steadily over a long period and result in increases in government revenues, the county would be able to plan for and absorb increased demands for education and public safety services such as law enforcement and fire protection. These services are currently at capacity. Nye County communities are geographically widely separated from one another, and communities cannot readily share public services. If the incremental population increases reflected the current patterns in Nevada (rather than Nye County, which has a large retirement-age population), about 21 percent of new residents in a year would be school-age children. Schools in Nye County are at capacity, and the county is widely reliant on portable units at present. The county and the communities in the county would continue to provide services as the government revenue base grew.

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 county-wide 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 serve to alleviate the scarcity of medial 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 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 would live in temporary work camps and return to out-of-county homes on the weekends. 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 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 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 Repository SEIS, monitoring and closure activities would end 50 years after the emplacement of the last waste package. The 10-year closure 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 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 been rendered and, to the extent necessary, mitigated. 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 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 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 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 is 7,400 cubic meters (6 acre-feet) per year and would be unlikely to change during the extended monitoring period.

A.5.1.4 Biological Resources and Soils

The potential impacts to biological resources and soils due to an extended monitoring 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 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 member 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 to 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 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		Impact for additional 200-year monitoring period[a]
Public, Radiological		
MEI (probability of an LCF)	0.00029	No change
Population (LCFs)	8	18
Fatalities due to emissions		
Workers (involved and noninvolved)		
Radiological (LCFs)	4.4	2.8

a. Additional impacts were obtained by multiplying the 40-year monitoring impacts by a factor of 5 (200 divided by 40).
LCF = Latent cancer fatality.

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

The extended monitoring 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 (55.5 million gallons) additional
Oils and lubricants:	44 million liters (11.6 million gallons) additional

A.5.1.7 Waste and Hazardous Materials

During the extended monitoring 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:	52,000 cubic meters (68,000 cubic yards)

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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 DOE used to estimate impacts from potential nonradiological releases to air for this *Draft 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-S1D) (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, the U.S. Department of Energy (DOE or the Department) 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 (about 0.0001 inch), and $PM_{10,}$ particulate matter with an aerodynamic diameter of 10 micrometers or less (about 0.0004 inch). 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. Erionite, an uncommon zeolite mineral, could be encountered 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 was not considered 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 proposed repository construction period, the operations and monitoring periods, and the closure period, respectively. 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.

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. Because there would be no sources of airborne lead at the repository, the analysis did not consider that

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		150	Same
$PM_{2.5}$	Annual		15	None
	24-hour[c]		35	None
Ozone	8-hour	0.08		None
	1-hour[d]	0.12	235	Same
Lead	Quarterly		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. Effective December 17, 2006.

d. 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.

pollutant. 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 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 to 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 land withdrawal area from the burning of fossil fuels during the operations 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 on-road automobile and truck sources. Emissions of nitrogen dioxide due to the Proposed Action would be relatively small in comparison to 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.

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* (Yucca Mountain FEIS) (DIRS 155970-DOE 2002, pp. F-12 to F-14), 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 four 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.

Exposure to cristobalite to members of the public and to surface workers could occur. The sources of cristobalite releases would include fugitive dust from the excavated rock pile and dust emission 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 ISC model (40 CFR Part 51, Appendix W). The AERMOD provides better characterization of plume dispersion than the ISC model. 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 project 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 that DOE established in Appendix G of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-1 to G-44) 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 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 traveled 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.

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

Direction	From North Portal	From South Portal
Northwest	14	15
West-northwest	12	12
West	11	11
West-southwest	14	12
Southwest	18	16
South-southwest	23	19
South	21	18
South-southeast	21	19
Southeast	22	24

Source: Derived from DIRS 104493-YMP 1997, all, and DIRS 153849-DOE 2001, p. 1-21.
Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.

The potential location of the maximally exposed individual member of the public for surface construction outside the analyzed land withdrawal 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 (YMP 5). DOE used meteorological station YMP5 for AERMOD simulation 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 would best represent the meteorological data for all activities within 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. Those data were preprocessed with AERMET for input into AERMOD.

Desert Rock is near Mercury, Nevada, approximately 44 kilometers (27 miles) east-southeast of the proposed North Portal surface 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 the 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 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

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	0.087

Note: Numbers are rounded to two significant figures.

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 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.

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

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 location for receptors 100 meters from surface construction activities outside the 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	50
3-hour (9 a.m. to 12 p.m.) – 24-hour average concentration	South	4.7

Note: Numbers are rounded to two significant figures.

conservative approach. The following sections describe the source release rates and impacts for each period of activity.

B.5 Construction Period

This section describes the methods DOE used to estimate air quality impacts during the construction 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 earth moving 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, earth moving and truck traffic) and calculate the emissions for each component. However, information to that detail was not available for the construction 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 suppression 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 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 disturb the entire land area required for excavated rock storage (for both the construction and operations 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

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

Operations area	Disturbed land
North and South Portal areas	
North Portal site	2.8
Topsoil storage location near North Portal site	0.061
North Portal site ancillary support facilities	0.14
North Portal site protective forces administrative facility	0.081
Aging pads	0.57
Subsurface intake/exhaust shafts (and access roads)	0.24
South Portal area	0.081
Muck storage (excavated rock pile)	0.81
Rail Equipment Maintenance Yard and associated rail facilities	0.4
Other—in land withdrawal area	
Main access road	2.3
Gate 510 security complex	0.11
Primary roads	0.4
Aggregate quarry/engineered fill quarry	0.4
Infrastructure: Power lines	0.12
Other—outside land withdrawal area	
Intersection at U.S. Highway 95	0.11
Disturbed land outside the land withdrawal area	0.26
Infrastructure: Offsite Sample Management Facility	0.012
Total land disturbance	8.8
Area disturbed per year	1.8

Source: DIRS 182827-Morton 2007, all.
Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures; therefore, totals might differ from sums.

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 be emitted 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 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.

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

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

Period	Pollutant emission (kilograms)	Emission rate (grams per second)
North and South Portal areas		
Annual[a]	230,000	7.2
24-hour	910	31[b]
Other–in land withdrawal area		
Annual[a]	150,000	4.6
24-hour	580	20[b]
Other–outside land withdrawal area		
Annual[a]	17,000	0.54
24-hour	68	2.4[b]
Total		
Annual[a]	390,000	12
24-hour	1,600	54[b]

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures; therefore, totals might differ from sums.
a. 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.
b. Based on an 8-hour release period.
NAAQS = National Ambient Air Quality Standards.

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.

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 from construction activity	24-hour	64	150	43

Note: Conversion factors are on the inside back cover of this Repository SEIS.
a. Numbers are rounded to two significant figures.

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

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

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.
Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.
a. Based on a 24-hour release period.

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

	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).

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 in the Yucca Mountain FEIS 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 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 (0.1 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 kilometers (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 grams (0.018 ounces) 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 period.

	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 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 to 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-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.

B.5.5 FUGITIVE DUST FROM EXCAVATED ROCK REMOVAL

Excavated rock from construction of the Exploratory Studies Facilities 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

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

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: Conversion factors are on the inside back cover of this Repository SEIS. EPA updated emission rates in June 2006. Numbers are rounded to two significant figures.
EPA = U.S. Environmental Protection Agency.

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

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

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.
a. NAAQS annual PM_{10} regulatory limit revoked December 17, 2006; therefore, DOE did not calculate annual PM_{10} impacts. The annual pollutant emission is shown here for comparison purposes only.
b. Based on a 3-hour release period.
DOE = U.S. Department of Energy.
NAAQS = National Ambient Air Quality Standards.

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:

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

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 kilograms of PM_{10} per kilogram of transferred material (0.41 pounds 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 releases from excavated rock pile removal (PM_{10}).

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

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.

a. Assume 1 cubic meter of packed earth weighs 1,522 kilograms.

b. 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.

c. Based on 250 working days per year.

d. 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 period.

	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 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 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) for construction equipment.

Pollutant	Estimated emission	
	Diesel[a]	Gasoline[b]
Carbon monoxide	0.7475	37.1
Nitrogen dioxide	2.5	4.
Sulfur dioxide	0.005	0.11
PM_{10}	0.15	0.16
$PM_{2.5}$	0.1455	0.16[c]
Hydrocarbons	0.1836	1.9

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Assume the horsepower rating for construction equipment is between 176 and 300 horsepower.

a. Source: DIRS 174089-EPA 2004, p. A6.

b. Source: DIRS 182387-EPA 1997, all; DIRS 103679-EPA 1991, pp. II-7-1 and II-7-7.

c. Assume PM_{10} is 100 percent $PM_{2.5}$.

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

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

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers rounded to two significant figures; therefore, totals might differ from sums.

a. DOE estimated the amount of fuel use in and outside the LWA by multiplying the percentage of disturbed land in or outside the area by the total amount of fuel use during the construction period.

hp-hr = horsepower-hour.

LWA = Land withdrawal area.

B.6 Operations and Monitoring Periods

This section describes the methods DOE used to estimate air quality impacts during the operations and monitoring periods. The operations period would begin upon 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.

The monitoring period would begin at the completion of the operations 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, continuing 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 analysis 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 periods: three 8-hour shifts a day, 5 days a week, 50 weeks a year.

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

Pollutant	Period	Mass of pollutant per averaging period (kilograms)	Emission rate[a] (grams per second)
Construction in land withdrawal area			
Nitrogen dioxide	Annual	51,000	1.6
Sulfur dioxide	Annual	190	0.006
	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	0.29
PM_{10}	24-hour	0.55	0.019
$PM_{2.5}$	Annual	130	0.0042
	24-hour	0.53	0.018

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.

a. Based on an 8-hour release for averaging periods of 24 hours or less.

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 to regulatory limits.

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 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 period would emit fugitive dust. For this analysis and consistent with the **Table**

B-18. Air quality impacts from construction equipment during the construction 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	0.0002
	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 from construction activity)				
Nitrogen dioxide	Annual	1	100	1.0
Sulfur dioxide	Annual	0.004	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

Note: Conversion factors are on the inside back cover of this Repository SEIS.

a. Numbers are rounded to two significant figures.

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

Description	Total disturbed land	Percent disturbed during operations period	Land disturbed during operations period	Land disturbed per year during operations period[a]
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[b]			1.9	0.38

Note: Conversion factors are on the inside back cover of this Repository SEIS.

a. 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.

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

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 that had previously been disturbed during the construction period.

The analysis assumed the disturbance of an equal amount of land every year during the 5 years of surface construction in the operations 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 period.

The estimated PM_{10} emissions and emission rates during the operations period would be 21 percent of the total during the construction 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 analysis 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 would use during construction would operate during the first 4 years of the operations period. The Proposed Action would require an average of 42,000 cubic meters (55,000 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 operation period would be about 64 percent of those for the construction 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 period would be similar to the activity during the construction 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 period. For this analysis 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 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 that for the construction period (Section B.5.3). The PM_{10} concentration would be 0.26 percent of the regulatory limit, and the cristobalite concentration would be 0.018 percent of the regulatory limit.

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 period, while construction activities were occurring, the annual diesel fuel use would be 101 percent of that during the construction 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 period. Gasoline use would not decrease in comparison to 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 period, when the total amount of release would be greatest. Table B-21 lists the air quality impacts from surface equipment emissions.

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

Pollutant	Period	Mass of pollutant per averaging period (kilograms)	Emission rate[a] (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

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.

a. Based on an 8-hour release for averaging periods of 24 hours or less.

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.

B.6.6 EXHAUST EMISSIONS FROM BOILERS AND GENERATORS

Diesel plant heating boilers in the North Portal operations area would emit carbon monoxide, nitrogen dioxide, sulfur dioxide, and particulate matter. The basis for the emission calculations would be fuel

Table B-21. Air quality impacts from surface equipment during the first 5 years of the operations 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	0.07	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	0.003	15	0.02
	24-hour	0.43	35	1.2

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

consumption during the 5-year period of increasing operations activities when the annual total emissions would be greatest for the operations 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 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.

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	0.6
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).

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. The impact from boiler emissions was thus combined with impacts from the 5-year period of surface construction in order to calculate the most conservative combined impact.

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

Pollutant	Period	Mass of pollutant per averaging period (kilograms)	Emission rate[a] (grams per second)
Nitrogen dioxide	Annual	31,000	1
Sulfur dioxide	Annual	330	0.01
	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	0.1
	24-hour	13	0.46

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.
a. Based on an 8-hour release for averaging periods of 24 hours or less.

Table B-24. Air quality impacts from diesel boilers during the first 5 years of the operations 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 period. It is assumed that annual diesel fuel usage 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 period. Approximately 850 kilograms (1,900 pounds) of volatile organic compounds would also be released annually by the generators.

Table B-28 lists the air quality impacts from diesel generator emissions.

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

Pollutant	Estimated emissions: Pounds per 1,000 gallons diesel burned[a]	Estimated emissions: 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).

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

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

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

Pollutant	Period	Mass of pollutant per averaging period (kilograms)	Emission rate[a] (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

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.
a. Based on an 8-hour release for averaging periods of 24 hours or less.

B.7 Closure Period

This section describes the methods DOE used to estimate air quality impacts during the closure period at the proposed repository. The closure period would last 10 years and would overlap the last 10 years of the monitoring 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.

Table B-28. Air quality impacts from diesel generators during the operations 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.

For this analysis 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 to 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 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 in order 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 period. DOE assumed that 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 (0.039 ounces) 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 CONCRETE BATCH FACILITY

This section summarizes, incorporates by reference, and updates Section G.1.6.2 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. G-22 and G-23). DOE assumed that the concrete batch facility to be used during the closure period would be similar to those used during the construction and operations periods (Sections B.5.4 and B.6.2, respectively). The primary difference would be that the plant would run only 10 3-hour shifts a year for each concrete seal. The analysis assumed that the plant would produce two seals per year. Consistent with Table G-33 in the Yucca Mountain FEIS, DOE assumed the PM_{10} release amount would be 120 kilograms (265 pounds) per 24-hour period, with an emission rate of 11 grams (0.39 ounce) per second over a 3-hour release period. The maximum concentration of PM_{10} would be 4.2 micrograms per cubic meter, which is 2.8 percent of the regulatory limit. The fugitive dust from concrete batch facilities would be less then that during the construction 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.

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

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

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures. Assumes 10 years for closure.

a. 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.

b. Based on an 8-hour release period.

DOE = U.S. Department of Energy.

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 to 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-37 in the FEIS, DOE assumed the PM_{10} release amount would be 30 kilograms (66 pounds) per 24-hour period, with an emission rate of 0.35 gram (0.012 ounce) 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 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 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 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 period.

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

Pollutant	Period	Mass of pollutant per averaging period (kilograms)[a]	Emission rate[b] (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

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.

a. Mass of pollutant was calculated by using diesel emission factors from Table B-15.

b. 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 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.7.6 RAIL CONSTRUCTION: FUGITIVE DUST EMISSIONS DURING CONSTRUCTION PERIOD

Activities associated with constructing the rail line would generate fugitive dust. Crystalline silica might be present in the rock DOE could use as ballast, and thus crystalline silica might be present 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 work day with 250 working days per year. The estimated PM_{10} releases within the land withdrawal area from track construction would be about 150,000 kilograms (330,000 pounds) per year, or 610 kilograms (1,300 pounds) per day. The daily emission rate would be about 14 grams (0.49 ounce) per second. The maximum concentration of PM_{10} at the boundary of the land withdrawal area would be about 54 micrograms per cubic meter, which would be about 36 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 located at the receptor 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.

B.7.7 RAIL CONSTRUCTION: EXHAUST EMISSIONS DURING CONSTRUCTION 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 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 located at the receptor 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.

B.7.8 RAIL FACILITY CONSTRUCTION: EXHAUST EMISSIONS DURING CONSTRUCTION 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-33. Rail construction pollutant release rates in the land withdrawal area from surface equipment during the construction period.

Pollutant	Period	Mass of pollutant per averaging period (kilograms)	Emission rate[a] (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
PM_{10}	24-hour	140	3.2
$PM_{2.5}$	Annual	34,000	1.1
	24-hour	140	3.1

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.
a. 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 land withdrawal area during the construction 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.

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

Pollutant	Period	Mass of pollutant per averaging period (kilograms)	Emission rate[a] (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
PM_{10}	24-hour	23	0.81
$PM_{2.5}$	Annual	5,700	0.18
	24-hour	23	0.79

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.

a. 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 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	0.7	150	0.47
$PM_{2.5}$	Annual	0.0048	15	0.032
	24-hour	0.68	35	1.9

Note: Receptors at boundary of land withdrawal area.

a. Numbers are rounded to two significant figures.

B.7.9 RAIL FACILITY EMISSIONS DURING OPERATIONS 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) from rail facilities and activities during the operations 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	300	0	1,100
Carbon monoxide	10,000	190	150,000	0	160,000
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	37,000	150	42,000

Source: Rail Alignment EIS.
Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures; therefore, totals might differ from sums.

Table B-38. Air quality impacts from rail facilities and activities during the operations 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	0.00093	80	0.0012
	24-hour	0.13	365	0.036
	3-hour	1.1	1,300	0.081
Carbon monoxide	8-hour	57	10,000	0.57
	1-hour	460	40,000	1.1
PM_{10}	24-hour	1.4	150	0.94
$PM_{2.5}$	Annual	0.009	15	0.06
	24-hour	1.3	35	3.6

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

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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 and 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, 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.

In 1982, Congress enacted the *Nuclear Waste Policy Act*, as amended (NWPA; 42 U.S.C. 10101 et seq.) in recognition of the national problem created by 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 by redirecting DOE to determine the suitability of only Yucca Mountain in southern Nevada.

In 1989, DOE published a 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 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 a 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 a *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-0250; 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 unlikely to increase the risk of future flood damage, increase the impact of floods on human health and safety, or 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, the *Yucca Mountain Development Act*, 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 a December 29, 2003, "Notice of Preferred Nevada Rail Corridor" (68 FR 74951), 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 a "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 the this *Draft Supplemental Environmental Impact Statement for a Geological Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DOE/EIS-0250F-S1D) (Repository SEIS) is to address changes in the design and operation of the repository since the completion of the Yucca Mountain FEIS. This supplemental EIS will assist the NRC in the adoption, to the extent practicable, of any EIS prepared pursuant to Section 114(f)(4) of the NWPA.

This floodplain/wetlands assessment updates the floodplain and wetlands assessment that DOE included with the Yucca Mountain FEIS by addressing changes to the repository design and operational plans since 2002. Specifically, this assessment addresses potential effects of two elements: (1) the current 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 feels are 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). Appendix A of the Draft Environmental Assessment was a *Floodplain and Wetlands Assessment for the Proposed Infrastructure Improvements for the Yucca Mountain Project, Nevada*, which DOE has incorporated 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 snow.

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 and high-level radioactive waste across washes and manage them at facilities adjacent to washes.

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

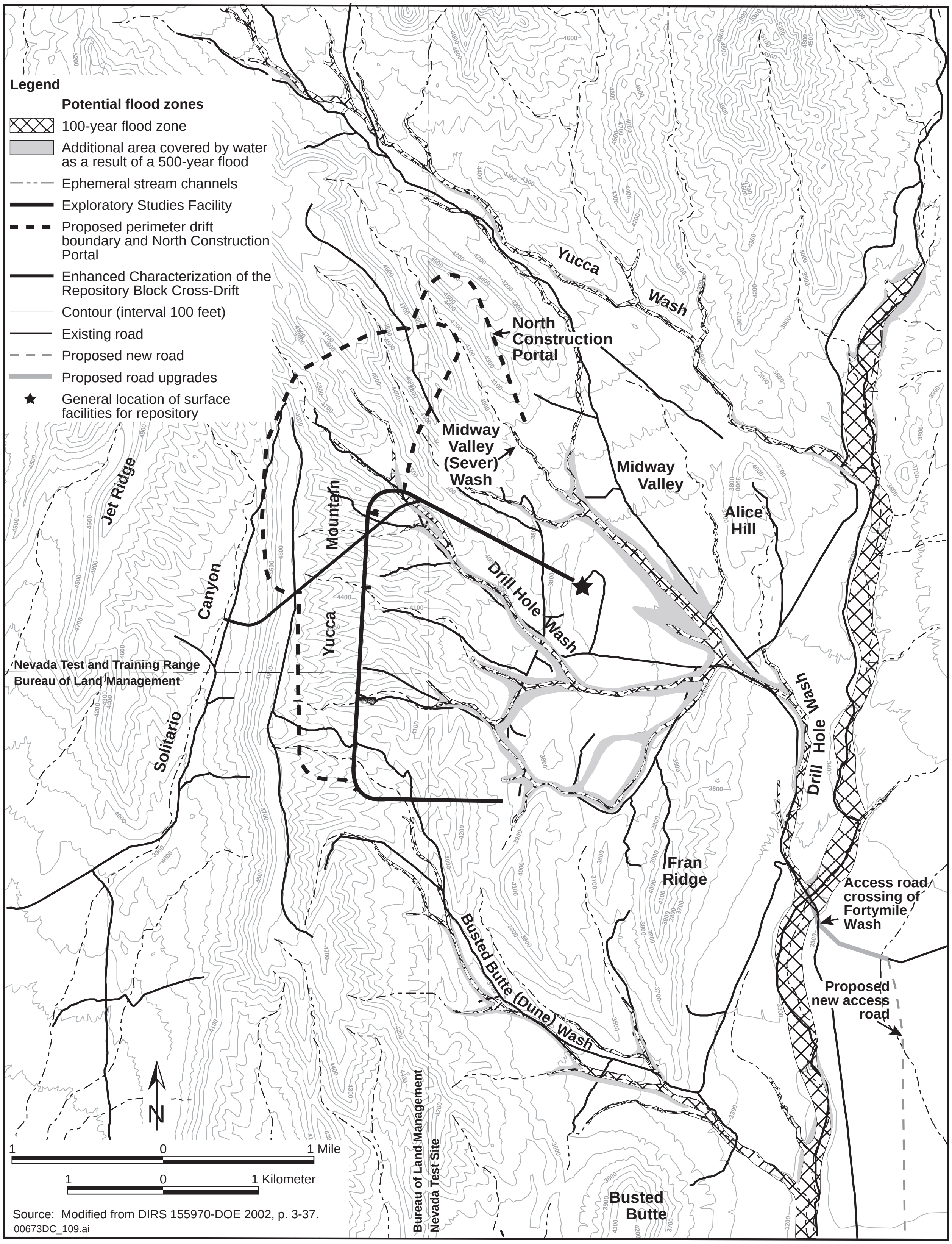

Figure C-1. Yucca Mountain site topography, drainage channels, and floodplains.

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 102783-Squires and Young 1984, Plate 1). Figure C-1 shows the lateral extents of 100- and 500-year floods within these drainages.

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.

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 180914-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. A water of the United States determination simply means that an additional regulatory requirement applies to the applicable channel in that DOE would require a permit from the U.S. Army Corps of Engineers before any fill or excavation could occur in the channel.

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 180914-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 report (DIRS 180914-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.

On June 5, 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*. 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 that 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 proposed repository construction area.

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 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 storm water detention ponds in the low eastern and southern sides of the diked area would control runoff. Culverts in the dikes would allow storm water 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 (ditches, improved channels, storm water ponds, etc.) 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 constructed 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 potential 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.

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)	100-year flood peak discharge (cubic meters per second)	500-year flood peak discharge (cubic meters per second)
Fortymile Wash	810	340	1,600
Busted Butte Wash	17	40	180
Drill Hole Wash[a]	40	65	280

Source: DIRS 102783-Squires and Young 1984, p. 2.

Note: Conversion factors are on the inside back cover of this Repository SEIS.

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. Glancy and Beck (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 to 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, p. 2-8), which summarizes 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 (see Section 3.1.51.3 of this Repository SEIS). These species can occur in the floodplains at and near Yucca Mountain, but they 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 1022.12(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 will 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, and 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 storm water 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 storm water 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 to 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, the DOE consulted with the U.S. Fish and Wildlife Service about the effects on the desert tortoise from construction, operations and monitoring, and 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, is applicable to the Proposed Action and its alternatives. 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 resource 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 or, if that was not possible, 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 have 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, to 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 groundwater table about 300 meters (980 feet) below the surface at Fortymile Wash.

DOE would control the use of petroleum fuels, oil, 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 for the Yucca Mountain project in comparison to 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 1022.12(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, would 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 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 that 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 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 1022.12(a)(3), DOE must consider alternatives to its 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 NWPA, which specified that DOE will dispose of spent nuclear fuel and high-level radioactive waste underground in deep geologic repositories. In the 1987 amendments to the NWPA, 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 will 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 ACTION

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 preliminary Floodplain Statement of Findings for those actions at the Yucca Mountain site and for the infrastructure actions that would affect only Fortymile Wash. Pending results of the public comment period for the Draft Repository SEIS, DOE intends to finalize the Statement of Findings below, or a similar one, in the Final Repository SEIS.

C.7.1 PRELIMINARY 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). Since Yucca Mountain has been designated as the site for development of a geologic repository, DOE believes 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 will 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, topsoil salvage, preservation of natural contours, and surface erosion or runoff control; reclamation and revegetation of disturbed areas; and use of established guidelines for hazardous materials storage and response to a spill.

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 PRELIMINARY 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, topsoil salvage, preserving natural contours, and controlling surface erosion and runoff; reclaiming and revegetating disturbed areas; and use of established guidelines for hazardous materials storage and response to accidental spills.

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 *Draft 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-S1D) (Repository SEIS). Preclosure impacts would occur during construction, operations and 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 this 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. We 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. Shielding from neutrons, which are also highly penetrating, requires materials that contain light elements such as hydrogen.

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 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 affects 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 the quantification of 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 plus 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 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. In addition, there is a potential for acute exposures in the event of an accident at an operating nuclear power plant, although U.S. Nuclear Regulatory Commission (NRC) regulations require plant designs that make such events extremely unlikely. Such exposures could occur only if a highly unlikely failure of the containment vessel around the nuclear reactor occurred with a large release of fission products.

In contrast, accidents during the shipment of spent nuclear fuel or high-level radioactive waste do not have the potential to release sufficient fission products to cause acute exposures that could immediately threaten the life of workers or the public. The fission product source term in the spent nuclear fuel would have decayed by a factor of 10,000 or more by the time the U.S. Department of Energy (DOE or the Department) shipped the material to the proposed repository. Therefore, there would not be sufficient energy in the fission products in the spent nuclear fuel and high-level radioactive waste to melt the fuel elements and vaporize fission products, as NRC has postulated for an accident at an operating nuclear power plant.

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 current 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 dose-to-health effect conversion factor recommended by the Interagency Steering Committee on Radiation Standards is higher than that in the analysis for 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). The FEIS used 0.0004 latent cancer fatality per person-rem for workers and 0.0005 latent cancer fatality per person-rem for individuals among the general population (DIRS 155970-DOE 2002, p. 3-97). The recommended 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 182836-ICRP 2007, p. 25). 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 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 to 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 182836-ICRP 2007, p. 25).

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}	6.0×10^{-4}
Adults	4.1×10^{-4}	1×10^{-5}	4.0×10^{-4}

Source: DIRS 182836-ICRP 2007, p. 25.
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, 0.0004 per person-rem for workers and 0.0006 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 18125-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. 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 periods (Section D.2.1). Handling and transfer of commercial spent nuclear fuel in the surface Wet Handling 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 and release of neutron-activated rock dust to the air from the emplacement drift walls (Section D.2.3). As indicated in Section D.5.1, almost all (99.9 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 period and 22 years of construction and operations at the beginning of the 50-year operations 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 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 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.

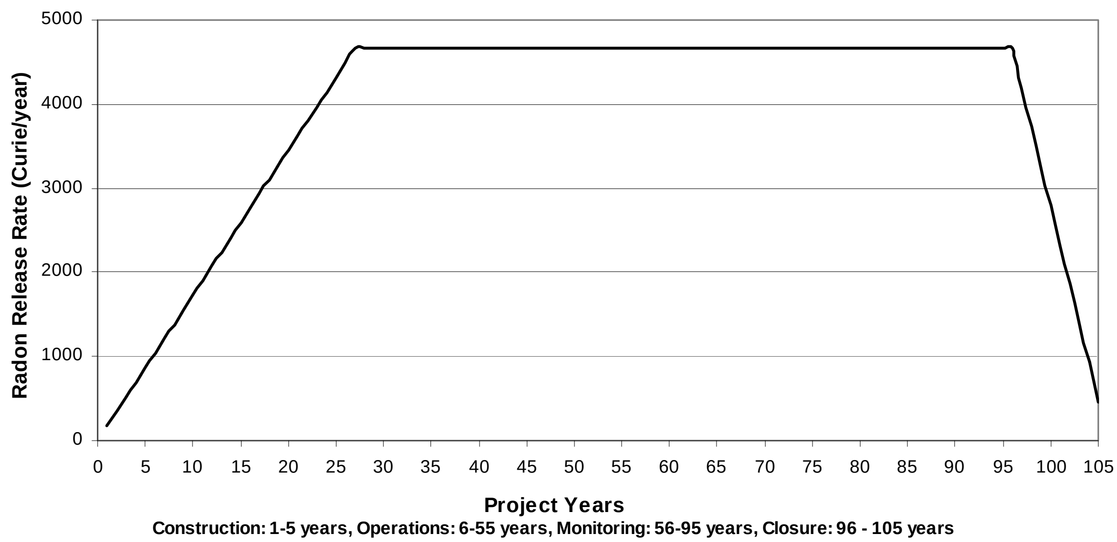

Figure D-1. Radon release rate as a function of time.

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 transportation, aging, and disposal (TAD) canisters. Most DOE spent nuclear fuel and high-level radioactive waste would arrive in disposable canisters. The only exception would be DOE spent nuclear fuel of commercial origin, which could arrive uncanistered. 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 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 these 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 gasses and a small percentage of the particulates in the canisters or shipping containers would be released to the atmosphere under normal operating conditions.

The Wet Handling Facility, which would process about 10 percent of the commercial spent nuclear fuel, would be the only surface facility with the potential to release radioactive materials to the environment during normal operations. The other surface facilities would handle only sealed canisters, and therefore would not release airborne radioactive materials under normal operating conditions. The sources of radioactive materials from the Wet Handling Facility would include cask venting and fuel failures during handling and temporary staging. The following sections describe the assumptions and methods for estimation of these releases.

D.2.2.1 Characteristics of Spent Nuclear Fuel

Airborne releases during normal operations would only occur in the Wet Handling Facility during processing of uncanistered fuel and fuel from dual-purpose canisters. Because 90 percent of the waste stream would be received in TAD canisters, potential airborne releases would be only from the remaining portion of the waste stream. To estimate the magnitude of the radioactive releases from the Wet Handling Facility, the analysis conservatively assumed that all commercial spent nuclear fuel would consist of the same composition of radionuclides. This composition represents the design-basis fuel characteristics, and the analysis based it on a 4-percent (maximum) initial enrichment of uranium-235 in a large pressurized-water reactor with burnup of 60,000 megawatt-days per metric ton of heavy metal (MTHM) and a cooling (or aging) time of 10 years after removal from the reactor (DIRS 161120-BSC 2002, Section 5.5). The radiation intensity of this fuel bounds approximately 97 percent of the fuel that DOE would dispose of at the proposed repository (DIRS 161120-BSC 2002, Section 5.5.1). Use of the design-basis fuel characteristics also bounds the representative commercial spent nuclear fuel characteristics developed for repository preclosure normal operations radiological impact analysis; which based it on a pressurized-water reactor fuel assembly with 4.2-percent initial enrichment, 50,000 megawatt-days per MTHM burnup rate, and 10 years cooling time (DIRS 180185-BSC 2007, Section 7). The radionuclide composition of this design-basis fuel, therefore, represents a conservative approach for estimation of the potential release source terms during normal operations.

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

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	0.00005

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.

commercial spent nuclear fuel rod; the fractions are applicable only to the failed fuel rods in a fuel assembly.

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 that handling activities would release to the environment as a result of normal operations. Table D-3 lists the radionuclide releases for an annual throughput of 3,000 MTHM of commercial spent nuclear fuel; 10 percent of this amount (300 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 99.7 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, two potential mechanisms could generate additional airborne releases of radioactive materials: neutron activation of ventilation air in the emplacement drifts and release of neutron activated rock dust to the air from the emplacement drift walls. Table D-3 lists the estimated annual releases of radionuclides from the subsurface facility under normal operating conditions (DIRS 172487-BSC 2005, pp. 33 to 35).

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 after redistribution to other organs of the body, and (3) ingestion of radioactivity in foodstuffs for offsite members of the public.

Table D-3. Annual releases from normal operations.[a,b]

Radionuclide	Curies per year
Subsurface facility releases	
Activated air[c]	
Nitrogen-16	3.4×10^{-2}
Argon-41	2.0×10^{1}
Activated dust[c]	
Sodium-24	3.7×10^{-3}
Aluminum-28	1.6×10^{-3}
Silicon-31	5.2×10^{-4}
Potassium-42	8.0×10^{-4}
Iron-55	8.2×10^{-5}
Naturally occurring radioactivity[d]	
Radon-222	4.7×10^{3}
Surface facility releases	
Wet Handling Facility releases	
Hydrogen-3	5.8×10^{2}
Carbon-14	8.2×10^{-1}
Chlorine-36	1.7×10^{-2}
Krypton-85	6.4×10^{3}
Iodine-129	5.2×10^{-2}
Cesium-134	6.9×10^{-4}
Cesium-137	9.0×10^{-3}
Surface facility releases	
Wet Handling Facility releases (continued)	
Barium-137m	8.5×10^{-3}
Crud (cobalt-60)	1.6×10^{-2}
Crud (iron-55)	2.0×10^{-1}
Fuel (cobalt-60)	4.1×10^{-5}
Nickel-63	5.0×10^{-6}
Strontium-90	8.6×10^{-4}
Yttrium-90	8.6×10^{-4}
Promethium-147	1.2×10^{-4}
Samarium-151	4.9×10^{-6}
Europium-154	5.7×10^{-5}
Europium-155	1.2×10^{-5}
Plutonium-238	7.5×10^{-5}
Plutonium-239	3.3×10^{-6}
Neptunium-239	7.4×10^{-6}
Plutonium-240	6.9×10^{-6}
Americium-241	2.4×10^{-5}
Plutonium-241	1.1×10^{-3}
Americium-243	7.4×10^{-7}
Curium-243	4.7×10^{-3}
Curium-244	1.1×10^{-4}

a. The listed source term nuclides contribute more than 99.9% of the total dose to the maximally exposed offsite member of the public.
b. Based on Wet Handling Facility throughput of 300 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.
c. Source: DIRS 172487-BSC 2005, Table 13.
d. Assumes a fully excavated repository; Source: DIRS 167021-BSC 2003, p. 37.

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 80 kilometers (50 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, noninvolved worker population doses from radon releases apply to involved and noninvolved workers.

D.3.1 PUBLIC

The location of the maximally exposed member of the public would be at the southeastern boundary of the analyzed land withdrawal area in the prevailing downwind direction (southeast and south-southeast) from the release points. DOE determined this to be the location of unrestricted public access that would receive the highest radiation exposure. The release points for radon and other subsurface facility releases include the South Portal and one to six exhaust ventilation shafts. Normal operations releases of manmade radionuclides would occur only from the Wet Handling Facility, near the North Portal. The analysis used 22 kilometers (14 miles) in the south-southeast direction as a representative distance to the exposed individual location for releases from the Wet Handling Facility and 21 kilometers (13 miles) in the southeast direction for releases from subsurface facilities.

Table D-4 lists the estimated average population for 2067 of about 117,000 within 80 kilometers (50 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 80-kilometer region. The analysis calculated both annual population dose and cumulative dose for the Proposed Action duration of 105 years, which would consist of 5 years of construction, 50 years of operations, 50 years of monitoring, and 10 years of closure, which overlaps the final 10 years of the monitoring period.

Table D-4. Projected 2067 population distribution within 80 kilometers of repository site.

Direction	Distance (kilometers)									
	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

Note: Conversion factors are on the inside back cover of this Repository SEIS.

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 analyzed 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 period and the development of the first two emplacement panels during initial operations, 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 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 (30 miles) east-southeast of the proposed repository.

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

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 period of the project (construction, operations, monitoring, and closure). Radiological impacts to workers include those from naturally occurring and manmade radiation and from radioactive materials in the workplace. Radiological impacts to members of the public include those from potential exposure to airborne releases of naturally occurring radiation and manmade radionuclides.

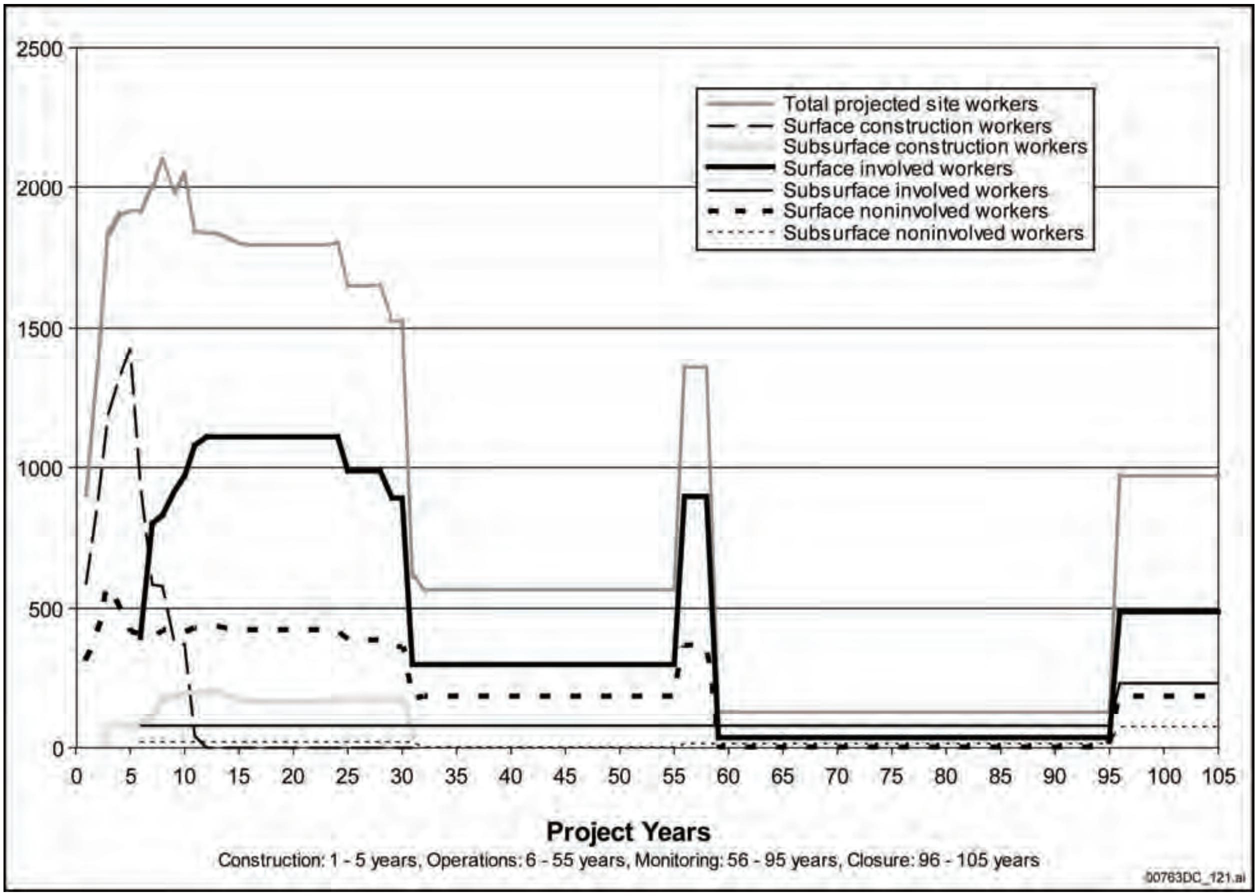

Figure D-2. Projected worker population for radiological impact assessment.

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 80 kilometers (50 miles) of the repository.

D.4.1 ESTIMATED PUBLIC AND NONINVOLVED WORKER DOSES

D.4.1.1 Estimated Doses from Atmospheric Releases

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, to calculate estimated population dose to the public and the dose to the maximally exposed workers and member of the public. 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 177510-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 (1) curie of radon-222. Factors for 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 analyzed activity period.

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

Source/facility	Maximally exposed individuals[a]				Population dose within 80 kilometers (person-rem)
	Offsite public (millirem)	Noninvolved subsurface worker (millirem)	Noninvolved surface worker (millirem)	NTS worker (millirem)	
Subsurface facility per curie radon release	0.0015	0.0011	0.00097	0.000031	0.033
South Portal per curie radon release[b]		0.066			
Surface facility per MTHM SNF processed	0.0000048	0.0000024	0.0000012	0.000000023	0.000097
Subsurface facility per year operation (non-radon release)	0.0011	0.0023	0.0048	0.000023	0.025

Notes: Conversion factors are on the inside back cover of this Repository SEIS. 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 period.
NTS = Nevada Test Site.
MTHM = Metric tons of heavy metal.
SNF = Spent nuclear fuel.

The analysis calculated individual and population doses for every year of the analyzed project period from the beginning of construction to the end of closure. To estimate the maximum annual doses, the analysis assumed that the proposed repository would receive and process spent nuclear fuel and high-level radioactive waste at the design capacity. Multiplying the unit dose factors in Table D-5 by the projected annual spent fuel processing rate for the repository yielded the 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 the noninvolved workers as a function of time predicted for each year using the 105-year analysis period.

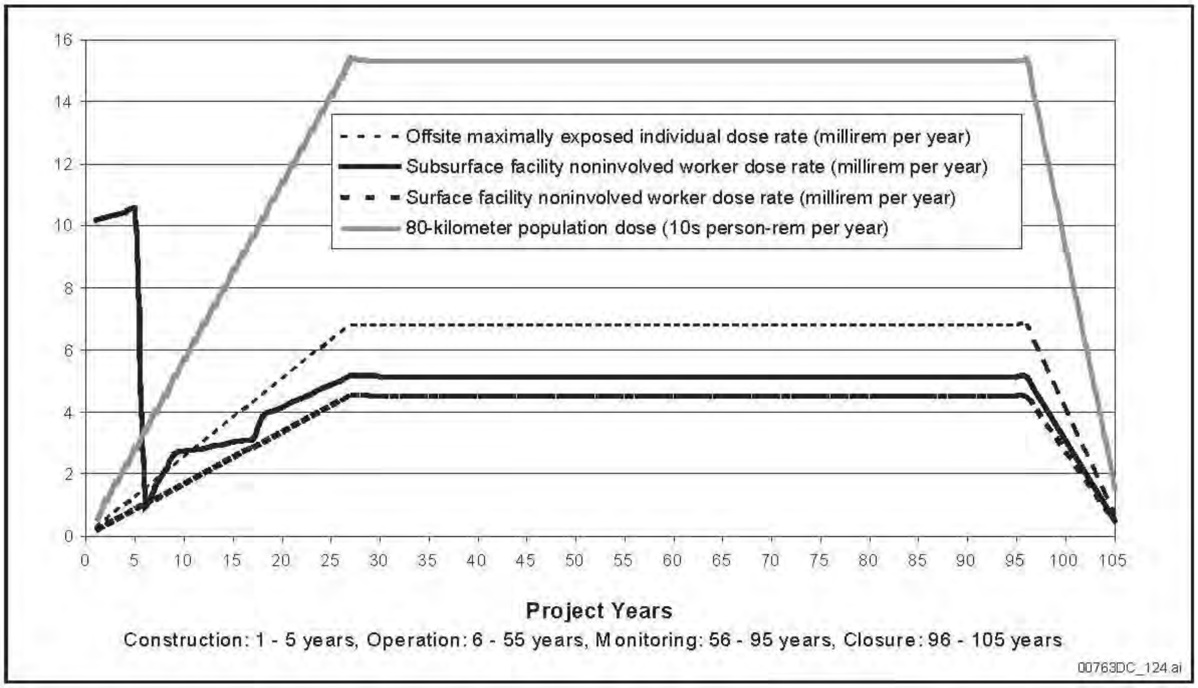

Figure D-3. Estimated individual and population doses from normal operations releases.

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

The analysis used dose rate versus distance information (DIRS 172729-BSC 2005, Table 4) and relative distances of the worker locations from the cask staging area to calculate dose rates at worker locations from exposure to external radiation from this source. It used dose rate-versus-distance information based on an aging overpack surface dose rate of 40 millirem per hour (DIRS 180131-BSC 2007, Figure 18) 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 at designated staging and aging areas. For conservatism, the analysis did not consider radiation shielding from construction materials and temporary shielding that DOE would provide for construction and operations activities. The calculated maximum annual dose and total dose for the entire operations period to a full-time noninvolved worker would be 7 millirem per year and 130 millirem, respectively. The total population dose to noninvolved workers over the entire operations period would be 63 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 analyzed activity period from normal operations.

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

Impact category	Construction	Operations	Monitoring	Closure
Maximum individual annual dose (millirem per year)				
Member of the public[b]	1.3	6.8	6.8	6.8
Noninvolved surface facility worker	0.83	11	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]	3.8	280	270	37
Noninvolved surface facility worker	2.5	320	180	25
Noninvolved subsurface facility worker	52	220	210	28
Population dose (person-rem)				
Exposed 80-kilometer population[c]	85	6,400	6,100	840
Noninvolved onsite population	4.7	230	26	18
Noninvolved Nevada Test Site population	0.12	9.2	8.9	1.2

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures.

a. About 99.9 percent of the dose and impact would be from naturally occurring radon-222 and its decay products.

b. At the southeastern boundary of the analyzed land withdrawal area.

c. The projected population would include about 117,000 individuals within 80 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 50 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 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 units 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 analyzed activity 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 analyzed 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 and ambient radiation.

D.4.2.2 Estimated Doses from Airborne Releases

The analysis used the calculated annual average atmospheric dispersion factors (DIRS 180308-BSC 2007, Table 7), 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 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 and 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. They would be the only airborne releases of manmade radionuclides during the monitoring and closure periods because the Wet Handling Facility would no longer be operating.

Table D-8 lists estimated radiological doses to involved workers from potential normal operational releases for each analyzed activity 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 project activity period.[a,b]

Impact category	Operations	Monitoring	Closure
Maximum individual annual dose (millirem per year)			
Surface facility	0.35	0.0048	0.0048
Subsurface facility	0.054	0.047	0.047
Maximum individual period total dose (millirem)			
Surface facility	7.4	0.19	0.026
Subsurface facility	2.5	1.9	0.26
Total worker population dose (person-rem)	1.8	0.17	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 period.

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 pads, and
- Low-Level Waste Facility.

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 analysis based dose assessments by worker group on job function and used time-motion inputs and calculated dose rates at worker locations. For cask processing facilities, the analysis based dose rates estimated for the design-basis commercial spent nuclear fuel. 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 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 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 4,300 person-rem and 510 person-rem, respectively. The largest contributions to individual and population doses would be preparation of casks and the transferal of casks to waste processing and storage areas in surface facilities.

Table D-9. Estimated radiation doses to involved surface workers from manmade external radiation during operations 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)	850
Initial Handling Facility	Maximum annual individual dose (rem/year)	0.81
	Total individual dose (rem)	19
	Total population dose (person-rem)	110
Wet Handling Facility	Maximum annual individual dose (rem/year)	0.96
	Total individual dose (rem)	22
	Total population dose (person-rem)	810
Canister Receipt and Closure Facilities	Maximum annual individual dose (rem/year)	0.29
	Total individual dose (rem)	6.8
	Total population dose (person-rem)	580
Aging pads	Maximum annual individual dose (rem/year)	0.89
	Total individual dose (rem)	21
	Total population dose (person-rem)	1,600
Low-Level Waste Facility	Maximum annual individual dose (rem/year)	0.95
	Total individual dose (rem)	22
	Total population dose (person-rem)	370
Total surface repository operations	Population dose (person-rem)	4,300

Source: (DIRS 182604-Darling 2007, Attachment I).
Note: Numbers are rounded to two significant figures.
a. Annual doses based on process of 3,000 MTHM commercial spent nuclear fuel throughput per year or about 500 casks per year.
b. Total doses based on process of a total waste throughput of 70,000 MTHM.
MTHM = Metric tons of heavy metal.

These conservative estimates of involved worker doses do not take credit for the application of administrative limits to reduce individual exposures. The Department would apply additional measures to ensure that radiation exposures to workers were as low as reasonably achievable.

Table D-10. Estimated radiation doses to involved subsurface workers from manmade external radiation during each project activity 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 182715-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 period.
c. Doses incurred from backfill operations.

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 period, 50-year operations period, 50-year monitoring period, and 10-year closure period, which would overlap the last 10 years of the monitoring period.

Table D-11 summarizes estimates of radiological doses to the public for each activity 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 99.9 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 6.8 millirem, which is less than 4 percent of the annual average 200- millirem dose to members of the public from ambient 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 80 kilometers (50 miles) 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 in the projected population in 2067 of about 117,000 persons within 80 kilometers of the repository would receive from natural background radon exposure.

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

Impact category	Construction	Operations	Monitoring[b]	Closure	Entire project
Maximally exposed member of the public[c]					
Maximum annual dose (millirem per year)	1.3	6.8	6.8	6.8	6.8
Total dose (millirem)	3.8	280	270	37	480[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. About 99.9 percent of the dose and impact would be from naturally occurring radon-222 and its decay products.
b. Doses are for monitoring period under active ventilation operating mode.
c. At the southeastern boundary of the analyzed land withdrawal area.
d. Based on a 70-year exposure of the maximally exposed individual.
e. The projected population includes about 117,000 individuals within 80 kilometers of the repository.

Table D-12 lists estimates of radiological doses to workers for each repository activity 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

Table D-12. Estimated radiation doses to workers during each activity 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	0.011	0.0052	0.0052	0.011
NTS noninvolved	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	0.32	0.21	0.028	0.32
NTS noninvolved	0.000079	0.0059	0.0057	0.00078	0.0059
Population dose (person-rem)					
Surface facility involved worker	--	4,300	0.019	0.023	4,300
Subsurface facility involved worker	33	1,400	890	400	2,700
Onsite noninvolved worker	4.7	230	26	18	280
NTS noninvolved	0.12	9.2	8.9	1.2	19
Total worker population	38	6,000	930	420	7,400

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 monitoring period under active ventilation operating mode.
NTS = Nevada Test Site.

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 years would be 7,400 person-rem. About 80 percent of the dose would occur during the operations 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 22 percent of the total worker dose. Inhalation of radon-222 and its decay products by subsurface workers would contribute 13 percent of the total dose, and ambient radiation exposure to subsurface workers would contribute 9 percent. To put the 7,400-worker person-rem occupational risk in perspective, the estimated worker population year of about 86,000 number of full-time equivalent worker years would receive 29,000 person-rem from natural background radiation exposure of 340 millirem per year (Chapter 3, Section 3.1.8.1) over the entire project period of 105 years. Therefore, the addition of 7,400 person-rem would represent a 25-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 with the exposed population. 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 activity 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 repository activity period and entire project duration.[a]

Health impact	Construction	Operations	Monitoring[b]	Closure	Entire project
Maximally exposed member of the public[c]					
Increase in probability of latent cancer fatality	0.0000023	0.00017	0.00016	0.000022	0.00029
Exposed 80-kilometer population[d]					
Number of latent cancer fatalities	0.051	3.8	3.7	0.51	8

Notes: Conversion factors are on the inside back cover of this Repository SEIS. Numbers are rounded to two significant figures; therefore, totals might differ from sums.

a. About 99.9 percent of the dose and impact would be from naturally occurring radon-222 and decay products.
b. Doses are for monitoring period under active ventilation operating mode.
c. At the southeastern boundary of the land withdrawal area.
d. The projected population includes about 117,000 individuals within 80 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 80 kilometers (50 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 periods. 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). The comparison indicates that over the 105-year project duration the incremental chance of latent cancer fatalities among the projected population of about 117,000 would be about 2 in 10,000.

D.5.2 ESTIMATED HEALTH IMPACTS TO WORKERS

Table D-14 summarizes estimates of radiological health impacts to workers for each repository activity 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.

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 4.4. This can be compared to 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).

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

Health impact/ worker group	Construction	Operations	Monitoring[a]	Closure	Entire project
Increase in probability of latent cancer fatality for the maximally exposed worker					
Involved	0.00029	0.018	0.0078	0.0010	0.018
Noninvolved	0.000031	0.00019	0.00012	0.000017	0.00019
Number of latent cancer fatalities in worker population					
Involved	0.02	3.5	0.54	0.24	4.2
Noninvolved	0.0028	0.14	0.016	0.011	0.17
Nevada Test Site noninvolved	0.000074	0.0055	0.0053	0.00073	0.012
Total	0.023	3.6	0.56	0.25	4.4

Note: Numbers are rounded to two significant figures; therefore, totals might differ from sums.
a. Health effects are for monitoring period under active ventilation operating mode.

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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 *Draft 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-S1D) (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 Department has modified the design and operating philosophy for the repository. 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 based on the current design and operating philosophy (1) to evaluate their 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. To meet licensing requirements, the results from the accident analysis will be more specific and comprehensive than those in this appendix and they will reflect a more fully developed repository design and operational details. To be consistent with the current design and operating philosophy, DOE revised the Yucca Mountain FEIS accident analyses, which now reflect the data and accident modeling changes.

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 FEIS to account for the current design and 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 reasonable 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 current design in *Yucca Mountain Project Critical Decision-1, Preliminary Hazards Analysis* (DIRS 176678-DOE 2006, 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, the *Yucca Mountain Project Critical Decision-1 Preliminary Hazards Analysis* (DIRS 176678-DOE 2006, all) provides the most recent repository accident scenario analysis for internal and external events that involved receipt, handling, or emplacement of spent nuclear fuel and high-level radioactive waste. That document addressed U.S. Nuclear Regulatory Commission (NRC) requirements in 10 CFR 63.112 and preclosure performance objectives in 10 CFR 63.111. The analysis was 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 a frequency of less than 1 in 1 million 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 only during these operations, the average annual probability 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 probability threshold for events that could occur anytime before permanent closure would be 0.01 per year. Similarly, the Category 2 event threshold is 2×10^{-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 (DIRS 176678-DOE 2006, Section 4.4.5 and Appendix A). However, DOE NEPA guidance recommends consideration of these events for evaluation if (1) they have an annual frequency above 1×10^{-7} per year, and (2) the consequences could be very large (DIRS 178579-DOE 2004, p. 28). As discussed in Section E.2.1.1.8, 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 evaluation that identified the internal accident scenarios (DIRS 176678-DOE 2006, all) did not quantitatively evaluate criticality events. DOE will address the means to prevent and control criticality as part of the Yucca Mountain preclosure safety analysis required for compliance with 10 CFR Part 63, in which the preclosure period covers the time before and during, permanent closure activities. The criticality objective of the preclosure safety analysis as stated in 10 CFR 63 is to perform:

> "…an analysis of the performance of the structures, systems, and components to identify those that are important to safety. This analysis identifies and describes the controls that are relied on to limit or prevent potential event sequences or mitigate their consequences. This analysis also identifies measures taken to ensure the availability of safety systems. The analysis required in this paragraph must include, but not necessarily be limited to, consideration of…(6) Means to prevent and control criticality…" [10 CFR Part 63, Subpart E, Section 112(e)].

To comply with this requirement, DOE has developed a process that it would use to demonstrate subcriticality for all preclosure operations with fissile materials for normal operations and for Category 1 and Category 2 event sequences. Subcriticality is defined as an end-state configuration with a maximum k_{eff} of less than an upper subcritical limit, which includes allowance for calculational bias and for administrative margin. Maintaining k_{eff} less than an upper subcritical limit prevents the occurrence of criticality. A demonstration of subcriticality is based on passive engineered systems (for example, fixed and soluble neutron absorbers, moderator control) with minimal reliance on administrative controls or operator intervention.

Even though it will be quantitatively demonstrated that no event sequence with a mean probability of occurrence greater than 1×10^{-4} during the preclosure period would result in a configuration that violated the upper subcritical limit, the actual likelihood of a criticality accident is significantly lower given the following conservatisms in the analysis:

- Commercial spent nuclear fuel represented as fresh (that is, no burnup credit), with an enrichment of 5 weight-percent uranium-235. This is conservative because the commercial spent nuclear fuel received at the repository will have a range of enrichments below 5 weight-percent uranium-235 and reduced fissionable material compared to fresh fuel due to burnup in a commercial reactor.

- Evaluation of the most reactive fuel state for DOE spent nuclear fuel (that is, fresh fuel for non-breeder reactor fuel, or calculated most reactive state for breeder reactor fuel). This is conservative because evaluation of the most reactive fuel state puts an upper bound on the potential for criticality.
- No credit for the presence of uranium-234 and -236. These isotopes are neutron absorbers that reduce the potential for criticality.
- Credit for only 75 percent of the neutron absorber loading. This assumption reduces the neutron absorber effect and thus increases the availability of neutrons that can cause fission.
- No credit for fixed neutron absorbers in moderator control facilities. This assumption also reduces the neutron absorber effect and thus increases the availability of neutrons that can cause fission.
- An administrative margin between the criterion to determine subcriticality and the actual critical calculated state.

Criticality could occur if the commercial spent nuclear fuel was moderated with water and had sufficient fissionable material in a configuration to allow criticality. However, the only place that DOE would store spent nuclear fuel in water would be the Wet Handling Facility storage pool. The water in this pool would be borated to prevent criticality. For DOE spent nuclear fuel that could be self-moderated (for example, spent nuclear fuel from training, research, isotopes General Atomics reactors or fast reactor fuel), DOE would use robust canisters, fixed neutron absorbers, and basket designs that provided fuel geometry controls to control criticality.

Considering these factors, DOE has determined that criticality would not be a credible event.

Table E-1 lists the accident scenarios that internal events could initiate that DOE included in the analysis. The table lists the Category 2 accident scenarios (DIRS 176678-DOE 2006, Section 4.4.4). The analysis did not identify any Category 1 scenarios. In addition, DOE performed a qualitative evaluation of beyond-Category-2 accident scenarios (Section E.2.1.1.8).

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.

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 other than naval 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 other than naval spent nuclear fuel (DIRS 176678-DOE 2006, p. A-1).

DOE selected fuel from pressurized-water reactors 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

Table E-1. Evaluated accident scenarios with internal initiators.

Scenario number	Location	Description	Material at risk	Expected occurrences over preclosure period (annual frequency)[a]
1	Initial Handling Facility, Canister Receipt and Closure Facilities	Breach of naval canister	1 naval canister	1.7×10^{-2} (3.4×10^{-4})
2	Initial Handling Facility, Canister Receipt and Closure Facility	Drop and breach of HLW canister in transportation cask	5 HLW canisters	2.1×10^{-2} (4.2×10^{-4})
3	Initial Handling Facility, Canister Receipt and Closure Facility	Breach of HLW canister in unsealed waste package or drop of equipment on HLW causing breach while in transportation cask or waste package	5 HLW canisters	9.8×10^{-2} ($2.0 \times 10^{-3)}$
4	Initial Handling Facility, Canister Receipt and Closure Facility	Drop with breach of HLW canister during transfer	2 HLW canisters	2.1×10^{-1} (4.2×10^{-3})
5	Wet Handling Facility	Drop of truck transportation cask without impact limiters causing breach	4 PWR or 9 BWR fuel assemblies	8.7×10^{-2} (1.7×10^{-3})
6	Wet Handling Facility	Drop of inner lid of truck transportation cask onto fuel assemblies in cask under water	4 PWR or 9 BWR fuel assemblies	4.4×10^{-2} (8.8×10^{-4})
7	Receipt Facility, Wet Handling Facility	Breach of DPC from drop or equipment impact	36 PWR or 74 BWR fuel assemblies	5.7×10^{-2} (1.1×10^{-3})
8	Wet Handling Facility	Breach of DPC under water from drop or equipment impact	36 PWR or 74 BWR fuel assemblies	2.1×10^{-2} (4.2×10^{-4})
9	Receipt Facility, Wet Handling Facility, Canister Receipt and Closure Facility	Drop and breach of TAD canister during handling operations	21 PWR or 44 BWR fuel assemblies	5.0×10^{-1} (1.0×10^{-2})
10	Wet Handling Facility	Drop of TAD canister lid onto fuel assemblies under water	21 PWR or 44 BWR fuel assemblies	1.7×10^{-2} (3.3×10^{-4})
11	Wet Handling Facility	Drop of one fuel assembly on another with breach under water	2 PWR or 2 BWR fuel assemblies	4.8×10^{-1} (9.6×10^{-3})
12	Wet Handling Facility	Drop of equipment on fuel assembly with breach under water	1 PWR or 1 BWR fuel assemblies	4.8×10^{-1} (9.6×10^{-3})
13	Low-Level Waste Facility	Fire involving low-level radioactive waste	Filters, spent resin, dry active waste, liquid waste	5.0×10^{-1} ($1.0 \times 10^{-2)}$

a. Annual frequency is estimated by dividing the expected number of occurrences over the preclosure period by the preclosure operating interval of 50 years.

BWR = Boiling-water reactor.
DPC = Dual-purpose canister.
HLW = High-level radioactive waste.
PWR = Pressurized-water reactor.
TAD = Transportation, aging, and disposal (canister).

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 into waste packages. The Initial Handling Facility would receive, package, and support placement of waste. Waste transfer operations would occur inside concrete enclosures.

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 receiving yard,
- 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.

The preliminary hazards analysis report (DIRS 176678-DOE 2006, all) did not consider accidents in the Initial Handling Facility because the facility was not yet part of the design. However, the Initial Handling Facility operations would be similar to the handling of high-level radioactive waste in the Canister Receipt and Closure Facility. Therefore, DOE assumes the same accident scenarios would apply. Because the number of canisters would remain the same, the number of handling operations would also be the same. Therefore, the accident frequencies in Table E-1 would be valid for the two facilities. DOE identified Scenarios 2 to 4 involving high-level radioactive waste (Table E-1) for the Canister Receipt and Closure Facility. These accident scenarios would apply to the Initial Handling Facility.

E.2.1.1.2 *Receipt Facility*

The functions of the Receipt Facility would be to (1) receive loaded transportation casks, (2) remove personnel barriers and impact limiters from the casks, and (3) transfer the TAD or dual-purpose canister from the transportation cask to a shielded transfer cask for movement to the Wet Handling Facility, one of the Canister Receipt and Closure Facilities, or an aging pad after placement in an aging overpack. Because the Canister Receipt and Closure Facilities would also directly receive TAD canisters in transportation casks, the primary function of the Receipt Facility would be to transfer TAD canisters and dual-purpose canisters from transportation casks to aging overpacks. In addition, the Receipt Facility would transfer TAD canisters from shielded transfer casks to aging overpacks, and transfer dual-purpose canisters from aging overpacks to shielded transfer casks, for movement to and from the Wet Handling Facility.

The Receipt Facility would receive only rail carriers directly. No uncanistered spent nuclear fuel would be handled in the Receipt Facility, and no canisters would be opened inside. There would be direct rail access to the Receipt Facility with a trench in the operating floor to position the deck of the railcar even with the operating floor.

The facility would consist of a multipurpose cell for shielded handling of TAD and dual-purpose canisters, as well as the aging overpacks and shielded transfer casks that held the canisters. The facility would accommodate the cask transporter for movement of the loaded aging overpacks and transfer casks to aging pads and to a Canister Receipt and Closure Facility, respectively. The cask transporter would move dual-purpose canisters in shielded transfer casks to the Wet Handling Facility and vertical dual-purpose canisters in aging overpacks to an aging pad. Casks containing horizontal dual-purpose canisters would be 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 shielded transfer casks and aging overpacks would utilize the vertical transfer method in *Yucca Mountain Project Critical Decision-1 Preliminary Hazards Analysis* (DIRS 176678-DOE 2006, p. 3-18, and Section 3.2.2.6.1.). Casks containing horizontal dual-purpose canisters would be transferred 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 via 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 the potential hazards of operations in the Receipt Facility, DOE identified two general accident scenarios with the potential to release radioactive material (DIRS 176678-DOE 2006, Table 4-5): Scenario 7, breach of a dual-purpose canister from drop or equipment impact, and Scenario 9, drop and breach of a TAD canister. These scenarios represent accidents that could occur during operations at the Receipt Facility that involved moving or lifting the dual-purpose and TAD canisters or that involved handling equipment over a dual-purpose canister when the canister was vulnerable to an equipment drop or fall. The estimated frequency of this accident takes into account the number of dual-purpose and TAD canisters the facility would handle and the number of operations for each canister. The analysis retained these scenarios for calculation of consequences.

E.2.1.1.3 *Aging Pads*

DOE would place TAD canisters into aging overpacks at the Wet Handling Facility, the Receipt Facility, and the Canister Receipt and Closure Facility. The aging overpacks would then be transferred to the aging pads to age the waste until it was ready for emplacement or repackaging. Vertical dual-purpose canisters could be placed into aging overpacks at the Receipt Facility and Canister Receipt and Closure Facility and also transferred to the aging pads. Casks containing horizontal dual-purpose canisters could be placed on a transfer trailer and moved to the aging pad where the dual-purpose canisters were pushed into an aging overpack. There would be two aging pads with 2,500 spaces for storage of up to 21,000 metric tons of heavy metal of waste. Chapter 2 provides a detailed description of aging operations. In evaluating these operations, DOE did not identify any Category 2 accident scenarios resulting in a release of radioactive materials.

E.2.1.1.4 *Wet Handling Facility*

The Wet Handling Facility would:

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

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

- 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.

- 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 transportation cask would be inspected and surveyed for contamination before transport back to the truck or rail buffer area.

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

- Close and seal-weld the loaded TAD canister and transfer it in a shielded transfer cask to a TAD 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 a shielded transfer cask for loading in a waste package or an aging overpack.

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

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

The Wet Handling Facility would handle commercial spent nuclear fuel, uncanistered and in dual-purpose 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. 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 the 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 under water 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 drops.

DOE identified Scenarios 5 through 12 (Table E-1) as accident scenarios applicable to operations in the Wet Handling Facility (DIRS 176678-DOE 2006, all).

E.2.1.1.5 *Canister Receipt and Closure Facilities*

The Canister Receipt and Closure Facilities would:

- Receive transportation casks with spent nuclear fuel and high-level radioactive waste in disposable canisters (TAD, dual-purpose, 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 shielded transfer casks with TAD canisters from the Wet Handling Facility and aging overpacks with TAD and dual-purpose canisters from aging pads.

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

- Transfer the contents of the transportation casks, shielded transfer casks, and aging overpacks into waste packages.

- Transfer TAD and dual-purpose canisters from transportation casks to shielded transfer casks or aging overpacks and transfer them between shielded transfer casks and aging overpacks.

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

- 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.

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

- Transfer the waste package to the Transport Emplacement Vehicle.

Each Canister Receipt and Closure Facility would house two shielded, remote canister-handling cells where DOE would transfer TAD canisters from shielded transfer casks or 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 or DOE canisters. All transportation casks with high-level radioactive waste and DOE and commercial spent nuclear fuel would move on rail cars directly from the rail buffer area to a Canister Receipt and Closure 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 overhead trolley. 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 2 through 4 and 9 (Table E-1) as Category 2 accident scenarios applicable to operations in the Canister Receipt and Closure Facility (DIRS 176678-DOE 2006, all).

E.2.1.1.6 *Low-Level Waste Handling Facility*

The Low-Level Waste Facility would accept, manage, and store solid low-level radioactive waste and liquid low-level radioactive waste until shipment off-site for processing. DOE would use standard vehicular transport, such as open flatbed trucks, to move the low-level 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. The waste 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 identified Scenario 13 (Table E-1) as applicable to the Low-Level Waste Facility (DIRS 176678-DOE 2006, all).

E.2.1.1.7 *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 and the vehicle would then descend the North Ramp and proceed to the predetermined emplacement drift. A third-rail electrical system would power the Transport and Emplacement Vehicle. In addition, the transport locomotive has a battery for secondary power. DOE did not identify any accident scenarios for waste emplacement operations (DIRS 176678-DOE 2006, all). 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.2.1.1.8 discusses this accident scenario.

E.2.1.1.8 *Beyond-Category-2 Accident Scenarios*

As noted above, DOE evaluated accident scenarios with probabilities 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 probabilities of 1×10^{-7} per year or higher if the impacts 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 that the Yucca Mountain FEIS evaluated with a different vehicle, the Transport and Emplacement Vehicle. DOE determined that the probability of a runaway event involving the Transport and Emplacement Vehicle would now be less than 1×10^{-7} per year, which is less than the threshold guidance provided by DOE for reasonably foreseeable events (DIRS 180101-BSC 2007, all). Other beyond-Category-2 events could also occur at the repository. The preliminary hazards analysis lists 26 potential beyond-Category-2 internal events (DIRS 176678-DOE 2006, Table 4-7). However, DOE determined that none of these events would be likely to cause very large offsite consequences because most of the events could occur only in waste handling buildings that have high-efficiency particulate air filtration systems that 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 be deposited 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.

E.2.1.2 Externally Initiated Events

Externally initiated events result either from causes external to the repository (earthquakes, high winds, etc.) or from natural processes that occur over a long period within the repository (corrosion, erosion, etc.). 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 current repository design, a recent DOE analysis determined that an aircraft crash into repository surface facilities would have a frequency of 7.9×10^{-7} per year (DIRS 178581-BSC 2006, p. 61). While this probability is below the probability threshold of 2×10^{-6} per year and DOE need not 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 178581-BSC 2006, 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 on the aging pads 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 building walls are 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 Pads*. The aging pads would be concrete pads on which DOE would place TAD and dual-purpose canister aging overpacks. The specification for these aging overpacks (DIRS 182282-DOE

2006, page 20) specifies that 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 150 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 for this Repository SEIS reduced the building footprints to include only those areas that would handle spent nuclear fuel and high-level radioactive waste based on floor plans from current design drawings to areas shown to be vulnerable (DIRS 178180-BSC 2006, all; DIRS 178288-BSC 2006, all; DIRS 180278-BSC 2007, all; DIRS 180989-BSC 2007, all). Table E-2 lists the dimension changes.

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

Building	DOE frequency analysis[a]		Repository SEIS frequency analysis	
	Length	Width	Length	Width
Initial Handling Facility	93	52	67	21
Canister Receipt and Closure Facility	130	99	100	30
Receipt Facility	99	87	61	21
Wet Handling Facility	120	82	82	32

Note: Conversion factors are on the inside back cover of this Repository SEIS.
a. Source: DIRS 178581-BSC 2006, p. 26.

- *Concrete walls*. The concrete walls of the buildings would vary in thickness from 1.5 to 1.8 meters (5 to 6 feet) (DIRS 180989-BSC 2007, all; DIRS 180278-BSC 2007, all; DIRS 178180-BSC 2006, all; DIRS 178288-BSC 2006, all). The DOE standard for accident analysis for aircraft crash into Hazardous Facilities (DIRS 101810-DOE 1996, p. 58) evaluated the potential for aircraft parts to penetrate concrete and recommends the following concrete penetration formula:

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

where

t = 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 (meters)
U = reference velocity [61 meters (200 feet) per second (DIRS 101810-DOE 1996, p. 68)]
V = missile impact velocity (aircraft impact velocity) (meters per second)
M = mass of the missile or the weight (kilograms) divided by gravitational acceleration [9.8 meters (32 feet) per square second)]
D = missile diameter (meters)
f_c = ultimate compressive strength of the concrete (kilograms per square meter)

Small military aircraft from Nellis Air Force Base dominate the probability for aircraft crash (DIRS 178581-BSC 2006, Section 7), 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 relevant characteristics of these engine parts that are relevant to Equation E-1 are an engine mass of about 59 kilograms (130 pounds), an engine diameter of about 1 meter (36 inches), an engine shaft mass of about 0.78 kilogram (1.70 pounds), and an engine shaft diameter of about 7.6 centimeters (3 inches). The ultimate compressive strength of reinforced concrete is

3.5 million kilograms per square meter (720,000 pounds per square foot) (DIRS 101910-Poe 1998, p. 1-4). The assumed impact velocity would be 150 meters (500 feet) per second based on *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 150 meters 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 150 meters per second, the F-15 or F-16 jet engine would penetrate about 1 meter (33 inches) of concrete, far less than the 1.5- to 1.8-meter (5- to 6-foot) wall thickness in the current design for 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 state in *Frequency Analysis of Aircraft Hazards for License Application* (DIRS 178581-BSC 2006, 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 181890-Ashley 2007, 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 178581-BSC 2006, p. 22), carrying ordnance over the flight-restricted airspace around the repository would be prohibited. Therefore, DOE considers this hazard as negligible or nonexistent (DIRS 178581-BSC 2006, p. 61).

Despite this result, and consistent with the Yucca Mountain FEIS, DOE analyzed a scenario in which a jet aircraft impacted and penetrated 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 was assumed to cause the Waste Handling Building to collapse. DOE based the FEIS 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). For the current design, 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 the potential consequences of seismic events. The Department intends to demonstrate seismic margins 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. However, DOE has determined (DIRS 174261-BSC 2005, Section 6.1.4.4) that a bounding credible seismic event could occur that could cause (1) failure of the high-efficiency particulate air filters and associated ducting and dampers in the waste handling facilities leading to release of accumulated radioactive material, and (2) failure of confinements for the solid and liquid low-level radioactive waste inventories in the Low-Level Waste Handling Facility leading to a release of radioactive material from the low-level radioactive waste.

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, hydrocarbons (including fuels, oils, and lubricants), and various 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. These plans and procedures would ensure the proper management and remediation of spills. Therefore, the generation, storage, packaging, and shipment off the site of solid and liquid hazardous waste 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), 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 pass through high-efficiency particulate air filters, DOE assumed a retention factor of 0.99 for each filter stage for particulates and cesium (DIRS 174261-BSC 2005, p. 37). Therefore, for the two-stage filter systems in the Initial Handling Facility, Wet Handling Facility, Canister Receipt and Closure Facility, and Receipt Facility, the filters would reduce airborne particulates by a factor of 10,000.

E.3.1 NAVAL SPENT NUCLEAR FUEL

The drop (or fall) of a naval spent nuclear fuel canister (Scenario 1, Table E-1) could result in a breach of the canister and release of radionuclides contained with the spent fuel in the canister. Table E-3 lists the total airborne activity estimated by the Navy to be released (DIRS 182094-McKenzie 2007, Table 1.8-9). The Navy also estimated the fraction of the total release that would be respirable. However, for conservatism, the analysis assumed that all of the airborne release would be respirable, consistent with the assumption in *Preclosure Consequence Analyses for License Application* (DIRS 174261-BSC 2005, p. 69).

Table E-3. Total airborne activity release by radionuclide for drop of naval canister (curies).

Radionuclide	Total airborne activity release	Radionuclide	Total airborne activity release
Actinium-227	1.2×10^{-9}	Niobium-94	2.6×10^{-3}
Americium-241	4.2×10^{-4}	Palladium-107	5.2×10^{-7}
Americium-242m	4.7×10^{-6}	Plutonium-238	1.1×10^{-1}
Americium-243	6.1×10^{-6}	Plutonium-239	1.1×10^{-4}
Antimony-125	2.5×10^{-2}	Plutonium-240	1.4×10^{-4}
Barium-137m	3.6×10^{0}	Plutonium-241	3.3×10^{-2}
Cadmium-113m	3.2×10^{-4}	Plutonium-242	7.8×10^{-7}
Californium-252	6.6×10^{-11}	Promethium-147	1.2×10^{0}
Carbon-14	1.7×10^{-1}	Protactinium-231	5.9×10^{-9}
Cesium-134	4.8×10^{0}	Radon-226	6.3×10^{-11}
Cesium-135	2.7×10^{-4}	Radon-228	4.6×10^{-15}
Cesium-137	2.5×10^{1}	Rhodium-102	2.1×10^{-7}
Cobalt-60	5.3×10^{-1}	Ruthenium-106	2.9×10^{-1}
Curium-242	1.3×10^{-5}	Samarium-147	2.7×10^{-10}
Curium-243	7.0×10^{-6}	Samarium-151	1.1×10^{-2}
Curium-244	5.8×10^{-4}	Scandium-79	2.9×10^{-6}
Curium-245	4.7×10^{-8}	Strontium-90	3.7×10^{0}
Curium-246	1.2×10^{-8}	Technetium-99	6.1×10^{-4}
Curium-247	3.1×10^{-13}	Thorium-229	7.3×10^{-11}
Curium-248	1.0×10^{-12}	Thorium-230	1.9×10^{-8}
Europium-154	9.2×10^{-2}	Thorium-232	9.8×10^{-13}
Europium-155	1.8×10^{-2}	Tin-126	1.0×10^{-5}
Hydrogen-3	1.1×10^{2}	Uranium-232	3.8×10^{-6}
Iodine-129	1.1×10^{-2}	Uranium-233	9.8×10^{-9}
Iron-55	5.7×10^{-1}	Uranium-234	1.4×10^{-4}
Krypton-85	3.0×10^{3}	Uranium-235	2.2×10^{-6}
Lead-210	1.0×10^{-11}	Uranium-236	2.2×10^{-5}
Neptunium-237	1.5×10^{-5}	Uranium-238	8.7×10^{-9}
Nickel-59	3.0×10^{-3}	Yttrium-90	3.7×10^{0}
Nickel-63	2.9×10^{-1}	Zirconium-93	6.5×10^{-5}
Niobium-93m	1.2×10^{-2}		

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 waste to represent the materials at risk because it would have the highest dose consequences (DIRS 174261-BSC 2005, p. 72). Table E-4 lists the materials at risk per canister.

Table E-4. Materials at risk for high-level radioactive waste canisters (curies).

Radionuclide	Total airborne activity release	Radionuclide	Total airborne activity release
Antimony-125	1.2×10^{2}	Plutonium-238	9.9×10^{2}
Americium-241	3.3×10^{2}	Plutonium-239	1.7×10^{1}
Americium-242m	7.8×10^{-2}	Plutonium-240	8.4×10^{0}
Americium-243	1.4×10^{0}	Plutonium-241	8.4×10^{2}
Barium-137m	5.3×10^{4}	Plutonium-242	2.1×10^{-2}
Cadmium-113	2.6×10^{-11}	Praseodymium-144m	3.8×10^{0}
Californium-249	2.3×10^{-2}	Promethium-147	2.2×10^{3}
Californium-251	1.9×10^{-2}	Ruthenium-106	4.4×10^{0}
Cerium-144	3.8×10^{0}	Samarium-151	1.6×10^{2}
Cesium-134	2.0×10^{2}	Selenium-79	5.3×10^{-1}
Cesium-135	2.2×10^{-1}	Strontium-90	3.4×10^{4}
Cesium-137	5.6×10^{4}	Technetium-99	9.2×10^{0}
Cobalt-60	1.9×10^{2}	Thorium-229	8.9×10^{-5}
Curium-243	4.2×10^{-1}	Thorium-230	8.0×10^{-6}
Curium-244	4.4×10^{2}	Thorium-232	1.4×10^{-3}
Curium-245	2.4×10^{-2}	Tin-121m	1.9×10^{0}
Curium-246	2.9×10^{-2}	Tin-126	7.8×10^{-1}
Curium-247	2.2×10^{-2}	Uranium-232	3.0×10^{-4}
Europium-154	4.2×10^{2}	Uranium-233	5.6×10^{-2}
Europium-155	6.8×10^{-1}	Uranium-234	4.5×10^{-2}
Iodine-129	3.2×10^{-4}	Uranium-235	6.6×10^{-4}
Neptunium-237	2.9×10^{-2}	Uranium-236	3.7×10^{-3}
Nickel-59	8.4×10^{-1}	Uranium-238	4.7×10^{-2}
Nickel-63	8.0×10^{1}	Yttrium-90	3.4×10^{4}
Niobium-93m	1.5×10^{-1}	Zirconium-93	3.9×10^{-1}
Palladium-107	1.3×10^{-3}		

Source: DIRS 181690-Ray 2007, Table 2, Column 3.

The analysis established the airborne release fraction of the materials at risk to calculate the doses to workers and members of the public based on the method described in Yucca Mountain Project correspondence (DIRS 182924-Wisenburg 2007, all). The high-level radioactive waste release fraction would consist of pulverized particles that would result 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:

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

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 $\times p \times g \times h$
where
p = density of the high-level radioactive waste, 2.75 gram per cubic centimeter (DIRS 174261-BSC 2005, all)
g = gravitational constant, 980.7 centimeters per square second (DIRS 174261-BSC 2005, all)
h = drop height in centimeters.

For the high-level radioactive waste drop (Scenario 3 from Table E-1), the drop height would be 1,138 centimeters (448 inches) (DIRS 174261-BSC 2005, p. 75). This drop height is conservative because the handling system design would have a maximum drop height for an unsealed waste package with high-level radioactive of 710 centimeters (276 inches) (DIRS 174261-BSC 2005, all). Using a drop height of 1,138 centimeters results in a respirable fraction of

$$PULF = (2 \times 10^{-4}) \times (1.0 \times 10^{-7}) \times 2.75 \times 980.7 \times 1{,}138 = 6.14 \times 10^{-5} \qquad \text{(Equation E-3)}$$

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

For the three accident scenarios that would involve high-level radioactive waste (Scenarios 2, 3, and 4, Table E-1), the analysis applied a leak path factor (DIRS 176678-DOE 2006, p. 4-33, footnote c). This factor accounts 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 and 4, 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:

Scenario 2 = 5 canisters × 0.01 × 7 × 10^{-5} = 3.5×10^{-6}
Scenario 3 = 5 canisters × 0.1 × 7 × 10^{-5} = 3.5×10^{-5}
Scenario 4 = 2 canisters × 0.1 × 7 × 10^{-5} = 1.4×10^{-5}

The analysis applied these values to the materials at risk radionuclide values from Table E-2 and used the results to calculate the consequences from the high-level radioactive waste drop scenario.

E.3.3 COMMERCIAL SPENT NUCLEAR FUEL DROP

Scenarios 5 to 12 would involve releases from commercial spent nuclear fuel assemblies when the assemblies were damaged during an accident. The releases would consist of fuel and crud. For the analysis in this Repository SEIS, DOE chose to use the maximum fuel characteristics. This selection 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.

Previous analyses determined that the consequences of accidents that involved pressurized-water reactor fuel assemblies would be higher than those that involved boiling-water reactor assemblies. For the maximum fuel, the preclosure consequence analysis (DIRS 174261-BSC 2005, p. 40) validates this conclusion.

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

BWR = Boiling-water reactor.
GWd = Gigawatt-day.
MTU = Metric ton of uranium.
SNF = Spent nuclear fuel.
PWR = Pressurized-water reactor.

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 the 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.

Table E-6. Inventory for maximum commercial spent nuclear fuel (curies per assembly).

Radionuclide	Total airborne activity release	Radionuclide	Total airborne activity release
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.

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 that happened 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 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 173261-BSC 2005, all). 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 182924-Wisenburg 2007, all). 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.

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[c]	Accident scenarios (Table E-1)[d]
Hydrogen-3	3.0×10^{-1}	3.0×10^{-1}	7.0×10^{-1}	5 to 12
Krypton-85	3.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	5 to 12
Iodine-129	3.0×10^{-1}	3.0×10^{-1}	3.0×10^{-1}	5 to 12
Cesium	2.0×10^{-3}	2.0×10^{-7}	2×10^{-7}	5, 7, 9
Strontium[e]	3.0×10^{-5}	3.0×10^{-9}	2×10^{-7}	5, 7, 9
Ruthenium	2.0×10^{-3}	2.0×10^{-7}	2×10^{-7}	5, 7, 9
Crud[e]	1.5×10^{-2}	1.5×10^{-6}	0	5, 7, 9
Fuel fines[e]	3.0×10^{-5}	3.0×10^{-9}	2×10^{-7}	5, 7, 9

a. Source: DIRS 182924-Wisenburg 2007, all
b. Factor of 1×10^{-4} applied per DIRS 176678-DOE 2006, p. B-12.
c. Factor of 1×10^{-4} applied per DIRS 176678-DOE 2006, p. B-9.
d. These scenarios would occur where HEPA filtration was operating.
e. See Section E.3.3.2 for crud component.
HEPA = High-efficiency particulate air (filter).
RARF = Respirable airborne release fraction.

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 (5, 7, and 9).

For accident scenarios that would occur in the pool of the Wet Handling Facility (8, 10, 11, and 12), the analysis assumed release of only gaseous radionuclides because the particulates would be trapped by the water above the commercial spent nuclear fuel assemblies (DIRS 176678-DOE 2006, p. B-3) and would not be available for release. Consistent with the preliminary hazards analysis (DIRS 176678-DOE 2006, p. B-3), the analysis for this Repository SEIS assumed release fractions of 1.0 (100 percent) for krypton-85, hydrogen-3, carbon-14, and chlorine-36, and 0.005 for iodine-129. Absorption in the water would reduce the iodine-129 release.

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 provides the crud activity per assembly at the time of discharge from the reactor (DIRS 176678-DOE 2006, p. B-8), and after 5 years of decay (DIRS 174261-BSC 2005, Table 15). The analysis assumed that the fraction of crud release in a drop accident scenario would be 0.015 (DIRS 176678-DOE 2006, p. B-9), all of which would be respirable.

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

	Inventory					
	At discharge		At 5 years		Respirable amount (5-year-old fuel)	
Radionuclide	PWR	BWR	PWR	BWR	PWR	BWR
Iron-55	2.7×10^3	1.3×10^3	7.5×10^2	3.5×10^2	11	5.3
Cobalt-60	63	2.1×10^2	33	1.1×10^2	0.49	1.6

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

E.3.4 LOW-LEVEL 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 dry active waste stored at the Low-Level Waste Handling Facility. The source term for this scenario would be 0.034 curie per cubic meter (DIRS 182584-BSC 2007, Table 9). Table E-9 lists the distribution of radionuclides released from the fire event as developed in *Preclosure Consequence Analyses for License Application* (DIRS 174261-BSC 2005, all).

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

Radionuclide	Respirable airborne release (curies)
Cesium-134	2.0×10^{-3}
Cesium-137	2.2×10^{-3}
Cobalt-58	1.7×10^{-3}
Cobalt-60	4.4×10^{-3}
Manganese-54	2.3×10^{-4}

Source: DIRS 174261-BSC 2005, Appendix II, Table 5.

E.3.5 SEISMIC EVENT

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. Airborne release fractions for this event are based on values for free-fall spills (DIRS 174261-BSC 2005, all) taken from the DOE handbook on release fractions (DIRS 103756-DOE 1994, all). Free-fall spill release fractions are used for the seismic event 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 provided in DIRS 174261-BSC 2005, all.) 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 based the estimate of the amount of accumulated radiological material available for release on the basis of: (1) commercial spent nuclear fuel would be received at an average rate of 630 fuel assemblies per month (based on 3,600 metric tons per year with each fuel assembly equivalent to 475 kilograms), (2) 10 percent of these (36 per month) are assumed to be handled as uncanistered fuel assemblies (and thus are available to release radionuclides during normal operations), and (3) 1 percent are defective (resulting in a release that is accumulated on the ducts and filters). An airborne release fraction of 1.0×10^{-2} is applied to the accumulated inventory based on releases from unenclosed filter media during a seismic event sequence from *Analysis of Experimental Data*, *Volume 1* of *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DIRS 103756-DOE 1994, Section 5.4.4.). The fuel assumed for this event is the representative pressurized-water reactor fuel assembly developed for normal operations releases. Table E-10 lists the source term for this event. The table gives the radionuclide inventory for the representative fuel assembly in the second column. The fourth column gives the filter buildup rate, which was calculated by the product of the curies per spent fuel assembly in the second column multiplied by 36 fuel assemblies per month, the airborne release fraction in the third column, and a factor of 0.01 for the defective fuel fraction. The fifth column gives the buildup after 18 months, and the sixth column is the amount released from the filters (1 percent of the 18-month buildup quantity). The seismic event is also assumed to release radionuclides from the Low-Level Waste Handling Facility and include releases from high-integrity containers, drums, boxes, and tanks containing liquid low-level radioactive waste. Details of this release estimate are provided in Yucca Mountain Project correspondence (DIRS 182924-Wisenburg 2007, all). The Low-Level Waste Handling Facility respirable airborne release includes five radionuclides; their activity is listed in the seventh column in Table E-10. This activity and is added to the corresponding high-efficiency particulate air filter release t o provide the total respirable airborne release (last column).

E.4 Accident Scenario Consequences

E.4.1 GENERAL METHODOLOGY

The analysis calculated the 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 (180 feet) from the release point, and (3) members of the public who resided within about 80 kilometers (50 miles) of the proposed repository in 2067 (Chapter 3, Figure 3-16). The 60-meter distance for the noninvolved worker is less than the 100 meters (330 feet) DOE used in the Yucca Mountain FEIS because the current design places 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 1 through 12), operators would be in enclosed operating areas that would isolate them from a release; (2) for scenario 13 (fire involving low-level radioactive waste), the fire would cause the release to be lofted into the atmosphere such that workers close to the release would not be receive meaningful exposure; and (3) for scenario 14 (seismic event), workers inside the Low-Level

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

Radionuclide	Representative PWR (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
Americium-241	1.2×10^{3}	3.0×10^{-5}	2.2×10^{-2}	4.0×10^{-1}	4.0×10^{-3}	0	4.0×10^{-3}
Americium-242	7.3	3.0×10^{-5}	1.4×10^{-4}	2.5×10^{-3}	2.5×10^{-5}	0	2.5×10^{-4}
Americium-242m	7.3	3.0×10^{-5}	1.4×10^{-4}	2.5×10^{-4}	2.5×10^{-4}	0	2.5×10^{-4}
Americium-243	23	3.0×10^{-5}	4.4×10^{-4}	7.8×10^{-3}	7.8×10^{-5}	0	7.8×10^{-5}
Antimony-125	3.9×10^{2}	3.0×10^{-5}	7.4×10^{-3}	0.13	1.3×10^{-3}	0	1.3×10^{-3}
Barium-137m	5.7×10^{4}	2.0×10^{-4}	7.2	$1.3 \times 10^{+2}$	1.3	0	1.3
Cadmium-113m	14	3.0×10^{-5}	2.6×10^{-4}	4.7×10^{-3}	4.7×10^{-5}	0	4.7×10^{-5}
Carbon-14	0.42	0.3	0	0	0	0	0
Cerium-144	73	3.0×10^{-5}	1.4×10^{-3}	2.5×10^{-2}	2.5×10^{-4}	0	2.5×10^{-4}
Cesium-134	4.1×10^{3}	2.0×10^{-4}	0.52	9.3	9.3×10^{-2}	6.0	6.1
Cesium-135	0.37	2.0×10^{-4}	4.7×10^{-5}	8.5×10^{-4}	8.5×10^{-6}	0	8.5×10^{-6}
Cesium-137	6.0×10^{4}	2.0×10^{-4}	7.6	$1.4 \times 10^{+2}$	1.4	6.8	8.2
Chlorine-36	8.5×10^{-3}	0.3	0	0	0	0	0
Cobalt-58	0	-		0		2.8	2.8
Cobalt-60	17	1.5×10^{-2}	0.16	2.9	2.9×10^{-2}	7.2	7.2
Curium-242	6.0	3.0×10^{-5}	1.1×10^{-4}	2.1×10^{-3}	2.1×10^{-5}	0	2.1×10^{-5}
Curium-243	16	3.0×10^{-5}	3.0×10^{-4}	5.4×10^{-3}	5.4×10^{-5}	0	5.4×10^{-5}
Curium-244	2.6×10^{3}	3.0×10^{-5}	4.9×10^{-2}	8.8×10^{-1}	8.8×10^{-3}	0	8.8×10^{-3}
Curium-245	0.34	3.0×10^{-5}	6.4×10^{-6}	1.2×10^{-4}	1.2×10^{-6}	0	1.2×10^{-6}
Curium-246	0.12	3.0×10^{-5}	2.2×10^{-6}	4.0×10^{-5}	4.0×10^{-7}	0	4.0×10^{-7}
Europium-154	2.4×10^{3}	3.0×10^{-5}	4.5×10^{-2}	0.81	8.1×10^{-3}	0	8.1×10^{-3}
Europium-155	4.9×10^{2}	3.0×10^{-5}	9.4×10^{-3}	0.17	1.7×10^{-3}	0	1.7×10^{-3}
Hydrogen-3	2.4×10^{2}	0.3	0	0	0	0	0
Iodine-129	2.3×10^{-2}	0.3	0	0	0	0	0
Iron-55	2.1×10^{2}	1.5×10^{-2}	2.0	36	0.36	0	0.36
Krypton-85	3.1×10^{3}	0.3	0	0	0	0	0
Manganese-54	0	-		0	0	1.2	1.2
Neptunium-237	0.25	3.0×10^{-5}	4.8×10^{-6}	8.6×10^{-5}	8.6×10^{-7}	0	8.6×10^{-7}
Neptunium-239	23	3.0×10^{-5}	4.4×10^{-4}	7.8×10^{-3}	7.8×10^{-5}	0	7.8×10^{-5}
Niobium-93m	0.34	3.0×10^{-5}	6.5×10^{-6}	1.2×10^{-4}	1.2×10^{-6}	0	1.2×10^{-6}
Niobium-94	6.3×10^{-5}	3.0×10^{-5}	1.2×10^{-9}	2.2×10^{-8}	2.2×10^{-10}	0	2.2×10^{-10}
Paladium-107	8.7×10^{-2}	3.0×10^{-5}	1.6×10^{-6}	3.0×10^{-5}	3.0×10^{-7}	0	3.0×10^{-7}
Plutonium-238	2.8×10^{3}	3.0×10^{-5}	5.3×10^{-2}	0.95	9.5×10^{-3}	0	9.5×10^{-3}
Plutonium-239	1.8×10^{2}	3.0×10^{-5}	3.4×10^{-3}	6.1×10^{-2}	6.1×10^{-4}	0	6.1×10^{-4}
Plutonium-240	3.2×10^{2}	3.0×10^{-5}	6.1×10^{-3}	0.11	1.1×10^{-3}	0	1.1×10^{-3}
Plutonium-241	5.2×10^{4}	3.0×10^{-5}	0.99	18	0.18	0	0.18
Plutonium-242	1.7	3.0×10^{-5}	3.2×10^{-5}	5.7×10^{-4}	5.7×10^{-5}	0	5.7×10^{-6}
Praseodymium-144	73	3.0×10^{-5}	1.4×10^{-3}	2.5×10^{-2}	2.5×10^{-4}	0	2.5×10^{-4}
Promethium-147	6.4×10^{3}	3.0×10^{-5}	0.21	2.2	2.2×10^{-2}	0	2.2×10^{-2}
Protactinium-231	3.0×10^{-5}	3.0×10^{-5}	5.7×10^{-10}	1.0×10^{-8}	1.0×10^{-10}	0	1.0×10^{-10}
Ruthenium-106	3.4×10^{2}	2.0×10^{-4}	4.3×10^{-2}	0.77	7.7×10^{-3}	0	7.7×10^{-3}
Samarium-151	2.5×10^{2}	3.0×10^{-5}	4.6×10^{-3}	8.4×10^{-2}	8.4×10^{-4}	0	8.4×10^{-4}
Selenium-79	4.8×10^{-2}	3.0×10^{-5}	9.0×10^{-7}	1.6×10^{-5}	1.6×10^{-7}	0	1.6×10^{-7}
Strontium-90	4.1×10^{4}	3.0×10^{-5}	0.78	14	0.14	0	0.14
Technetium-99	9.3	3.0×10^{-5}	1.8×10^{-4}	3.2×10^{-3}	3.2×10^{-5}	0	3.2×10^{-5}
Thorium-230	6.5×10^{-5}	3.0×10^{-5}	1.2×10^{-9}	2.2×10^{-8}	2.2×10^{-10}	0	2.2×10^{-10}
Tin-126	0.40	3.0×10^{-5}	7.5×10^{-6}	1.4×10^{-4}	1.4×10^{-6}	0	1.4×10^{-6}
Uranium-232	2.4×10^{-2}	3.0×10^{-5}	4.6×10^{-7}	8.3×10^{-6}	8.3×10^{-8}	0	8.3×10^{-8}

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

Radionuclide	Representative PWR (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
Uranium-233	2.5×10^{-5}	3.0×10^{-5}	4.7×10^{-10}	8.4×10^{-9}	8.4×10^{-11}	0	8.4×10^{-11}
Uranium-234	0.60	3.0×10^{-5}	1.1×10^{-5}	2.1×10^{-4}	2.1×10^{-6}	0	2.1×10^{-6}
Uranium-235	7.7×10^{-3}	3.0×10^{-5}	1.5×10^{-7}	2.6×10^{-6}	2.6×10^{-8}	0	2.6×10^{-8}
Uranium-236	0.18	3.0×10^{-5}	3.4×10^{-6}	6.2×10^{-5}	6.2×10^{-7}	0	6.2×10^{-7}
Uranium-238	0.15	3.0×10^{-5}	2.8×10^{-6}	5.0×10^{-5}	5.0×10^{-7}	0	5.0×10^{-7}
Yttrium-90	4.1×10^{4}	3.0×10^{-5}	0.78	14	0.14	0	0.14
Zirconium-93	0.83	3.0×10^{-5}	1.6×10^{-5}	2.8×10^{-4}	2.8×10^{-6}	0	2.8×10^{-6}

ARF = Respirable airborne release fraction.
HEPA = High-efficiency particulate air (filter).
LLW = Low-level radioactive waste.
SFA = Spent fuel assembly.
PWR = Pressurized-water reactor.

Waste Handling Facility would likely 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 100953-Napier et al. 1988, 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 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 were used to represent areas affected by wind direction from the repository. Atmospheric dispersion factors (dilution of the plume as a function of weather and distance from the release point) were calculated with the methodology in *General Public Atmospheric Dispersion Factors* (DIRS 177510-BSC 2007, all) for site boundary doses and for 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 that are deposited on soil (groundshine), ingestion of food products that become contaminated with radionuclides deposited from the plume, and exposure from radionuclides that are resuspended from the ground. The dose calculations included all of these pathways for the site boundary and 80-kilometer (50-mile) population doses. For the noninvolved worker, the analysis conservatively assumed the worker would evacuate within 8 hours (DIRS 176678-DOE 2006, p. B-11), so only direct exposure, inhalation from the plume, and groundshine for 8 hours were factors. Site-Specific Input Files for Use with GENII Version 2 (DIRS 177751-BSC 2007, all) 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 of the repository and 7.8 kilometers (4.8 miles) east of the repository. These two locations were determined to be the locations producing the highest site boundary dose based on sector-specific meteorology.

For facilities with high-efficiency-particulate-air filtration systems, the analysis in this Repository SEIS credits the filtration provided during an accident. In some cases (Initial Handling Facility), the results are also presented to provide the consequences of the same accident if filtration were not credited. These results illustrate that some filtration systems may not be required to meet regulatory standards, however, since they are included in the current 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 was also assumed that the accident occurred during the fall of the year so that 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 80 kilometers (50 miles) of the repository, including pockets of people who would reside just beyond the 80-kilometer distance. DOE selected the south-southeast sector to compute population doses because this sector would contain the highest population out to 80 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 104441-YMP 1998, 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 in relation to the accident scenarios, but the Yucca Mountain FEIS did consider the 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 that existed 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, DOE expects costs 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 CATEGORY 2 ACCIDENT SCENARIO CONSEQUENCES

To calculate the potential consequences for the Category 2 accident scenarios (Table E-1), 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. Tables E-11 and E-12 list the results of the consequence calculations. Table E-11 provides the 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 the 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 80-kilometer (50-mile) population around the site. For selected individuals and populations, the tables list

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

Accident scenario	Expected occurrences over the preclosure period (annual frequency)	Maximally exposed offsite individual[a]		Population		Noninvolved worker	
		Dose (rem)	LCF_i[b]	Dose (person-rem)	LCF_p[b]	Dose (rem)	LCF_i[b]
1. Drop of, or equipment drop on, naval canister with breach	1.7×10^{-2} (3.4×10^{-4})	2.0×10^{-4} (4.6×10^{-2})[c]	1.2×10^{-7}	5.4×10^{0} (4.0×10^{2})[c]	3.2×10^{-3}	1.5×10^{-2} (5.6×10^{0})[c]	9.0×10^{-6}
2. Drop with breach of HLW canisters in transportation cask	2.1×10^{-2} (4.2×10^{-4})	2.4×10^{-5} (2.4×10^{-3})[c]	1.4×10^{-8}	2.0×10^{-1} (2.0×10^{1})[c]	1.2×10^{-4}	3.2×10^{-3} (3.2×10^{-1})[c]	1.9×10^{-6}
3. Drop of HLW in an unsealed waste package or drop of equipment on HLW with breach	9.8×10^{-2} (2.0×10^{-3})	2.4×10^{-4} (2.4×10^{-2})[c]	1.4×10^{-7}	2.0×10^{3} (2.0×10^{5})[c]	1.2×10^{-3}	3.2×10^{-2} (3.2×10^{4})[c]	2.0×10^{-5}
4. Drop with breach of HLW canister during transfer	2.1×10^{-1} (4.2×10^{-3})	9.6×10^{-5} (9.6×10^{-3})[c]	5.8×10^{-8}	8.0×10^{-1} (8.0×10^{1})[c]	4.8×10^{-4}	1.3×10^{-2} (1.3×10^{0})[c]	7.8×10^{-6}
5. Drop of transportation cask with breach of PWR assemblies	8.7×10^{-2} (1.7×10^{-3})	1.2×10^{-3}	7.2×10^{-7}	3.0×10^{-5}	1.8×10^{-2}	1.9×10^{-1}	1.1×10^{-4}
6. Drop of inner lid of transportation cask of PWR assemblies with breach in water	4.4×10^{-2} (8.8×10^{-4})	9.2×10^{-4}	5.5×10^{-7}	2.5×10^{1}	1.5×10^{-2}	5.2×10^{-2}	3.1×10^{-5}
7. Drop of, or drop of equipment on, DPC with breach	5.7×10^{-2} (1.1×10^{-3})	1.1×10^{-2}	6.6×10^{-6}	2.7×10^{2}	1.6×10^{-1}	1.7×10^{0}	1.0×10^{-3}
8. Drop of, or drop of equipment on, DPC with breach in water	2.1×10^{-2} (4.2×10^{-4})	8.2×10^{-3}	4.9×10^{-6}	2.3×10^{2}	1.4×10^{-1}	4.6×10^{-1}	2.8×10^{-4}
9. Drop with breach of TAD canister	5.0×10^{-1} (1.0×10^{-2})	6.4×10^{-3}	3.8×10^{-6}	1.6×10^{2}	9.6×10^{-2}	1.0×10^{0}	6.0×10^{-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	
		Dose (rem)	LCF_i[b]	Dose (person-rem)	LCF_p[b]	Dose (rem)	LCF_i[b]
10. Drop of lid into TAD canister in water with breach of fuel assemblies	1.7×10^{-2} (3.4×10^{-4})	4.8×10^{-3}	2.9×10^{-6}	1.3×10^{2}	7.8×10^{-2}	2.7×10^{-1}	1.6×10^{-4}
11. Drop with breach of fuel assembly in water	4.8×10^{-2} (9.6×10^{-3})	4.6×10^{-4}	2.7×10^{-7}	1.3×10^{1}	7.8×10^{-3}	2.6×10^{-2}	1.6×10^{-5}
12. Collision or drop of equipment on fuel assembly in water with breach	4.8×10^{-1} (9.6×10^{-3})	2.3×10^{-4}	1.4×10^{-7}	6.3×10^{0}	3.8×10^{-3}	1.3×10^{-2}	7.8×10^{-6}
13. Fire involving LLW	5.0×10^{-1} (1.0×10^{-2})	6.2×10^{-6}	3.7×10^{-9}	4.9×10^{-2}	2.9×10^{-5}	4.0×10^{-4}	2.4×10^{-7}
14. Seismic event involving failure of HEPA system and LLW confinement	$<1.0 \times 10^{-4}$ ($<2.0 \times 10^{-6}$)	2.3×10^{1}	1.4×10^{-8}	1.9×10^{2}	1.1×10^{-1}	2.3×10^{0}	1.4×10^{-3}

a. Assumed to be at the analyzed land withdrawal boundary either in the east sector (7.8 kilometers or 4.8 miles) or in the southeast sector (18.5 kilometers or 11 miles), whichever produces the highest site boundary dose. For accident scenarios 6, 8, 10, 11 and 12, DOE calculated the highest dose for the southeast sector. For all other accident scenarios, DOE calculated the highest dose for the east sector.

b. 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.

c. Unfiltered doses presented to illustrate that filtration systems might not be required in the Initial Handling Facility to meet regulatory standards.

DPC = Dual-purpose canister.
HEPA = High-efficiency particulate air.
HLW = High-level radioactive waste.
LCF = Latent cancer fatality.
LLW = low-level radioactive waste.
PWR = Pressurized-water reactor.
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.

Accident scenario	Expected occurrences over the preclosure period (annual frequency)	Maximally exposed offsite individual[a]		Population		Noninvolved worker	
		Dose (rem)	LCF_i[b]	Dose (person-rem)	LCF_p[b]	Dose (rem)	LCF_i[b]
1. Drop of, or equipment drop on, naval canister with breach	1.7×10^{-2} (3.4×10^{-4})	3.1×10^{-6}	1.9×10^{-9}	3.6×10^{-2}	2.2×10^{-5}	2.4×10^{-3}	1.4×10^{-6}
2. Drop with breach of HLW canisters in transportation cask	2.1×10^{-2} (4.2×10^{-4})	4.1×10^{-7}	2.5×10^{-10}	1.4×10^{-3}	8.4×10^{-7}	5.3×10^{-4}	3.2×10^{-7}
3. Drop of HLW canisters in an unsealed waste package or drop of equipment on HLW canisters with breach	9.8×10^{-2} (2.0×10^{-3})	4.1×10^{-6}	2.5×10^{-9}	1.4×10^{-2}	8.4×10^{-6}	5.3×10^{-3}	3.2×10^{-6}
4. Drop with breach of HLW canisters during transfer	2.1×10^{-1} (4.2×10^{-3})	1.6×10^{-6}	9.6×10^{-10}	5.6×10^{-3}	3.4×10^{-6}	2.1×10^{-3}	1.3×10^{-6}
5. Drop of transportation cask with breach of PWR assemblies	8.7×10^{-2} (1.7×10^{-3})	7.2×10^{-5}	4.3×10^{-8}	5.6×10^{-1}	3.4×10^{-4}	8.3×10^{-2}	5.0×10^{-5}
6. Drop of inner lid of transportation cask of PWR assemblies with breach in water	4.4×10^{-2} (8.8×10^{-4})	1.1×10^{-5}	6.6×10^{-9}	1.5×10^{-1}	9.0×10^{-5}	8.4×10^{-3}	5.0×10^{-6}
7. Drop of, or drop of equipment on, DPC with breach	5.7×10^{-2} (1.1×10^{-3})	6.5×10^{-4}	3.9×10^{-7}	5.0×10^{0}	3.0×10^{-3}	7.5×10^{-1}	4.5×10^{-4}
8. Drop of, or drop of equipment on, DPC with breach in water	2.1×10^{-2} (4.2×10^{-4})	1.0×10^{-4}	6.0×10^{-8}	1.4×10^{0}	8.4×10^{-4}	7.6×10^{-2}	4.6×10^{-5}
9. Drop with breach of TAD canister	5.0×10^{-1} (1.0×10^{-2})	3.8×10^{-4}	2.3×10^{-7}	2.9×10^{0}	1.7×10^{-3}	4.4×10^{-1}	2.6×10^{-4}

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	
		Dose (rem)	LCF_i[b]	Dose (person-rem)	LCF_p[b]	Dose (rem)	LCF_i[b]
10. Drop of lid into TAD canister in water with breach of fuel assemblies	1.7×10^{-2} (3.4×10^{-4})	5.9×10^{-5}	3.5×10^{-8}	7.7×10^{-1}	4.8×10^{-4}	4.4×10^{-2}	2.6×10^{-5}
11. Drop with breach of fuel assembly in water	4.8×10^{-2} (9.6×10^{-3})	5.6×10^{-6}	3.4×10^{-9}	7.4×10^{-2}	4.4×10^{-5}	4.2×10^{-3}	2.5×10^{-6}
12. Collision or drop of equipment on fuel assembly in water with breach	4.8×10^{-2} (9.6×10^{-3})	2.8×10^{-6}	1.7×10^{-9}	3.7×10^{-2}	2.3×10^{-5}	2.1×10^{-3}	1.3×10^{-6}
13. Fire involving LLW	5.0×10^{-1} (1.0×10^{-2})	1.2×10^{-7}	7.2×10^{-11}	4.4×10^{-4}	2.6×10^{-7}	6.6×10^{-5}	4.0×10^{-8}
14. Seismic event involving failure of HEPA system and LLW confinement	$<1.0 \times 10^{-4}$ ($<2.0 \times 10^{-6}$)	4.4×10^{-4}	2.6×10^{-7}	1.6×10^{0}	9.6×10^{-4}	3.7×10^{-1}	2.2×10^{-4}

a. 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).

b. 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.

DPC = Dual-purpose canister.
HEPA = high-efficiency particulate air.
HLW = High-level radioactive waste.
LCF = Latent cancer fatality.
LLW = low-level radioactive waste.
PWR = Pressurized-water reactor.
TAD = Transportation, aging, and disposal (canister).

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 system and low-level radioactive waste confinement) would result in an estimated 0.11 latent cancer fatality for this same population.

Radiological dose information is also provided in Table E-11 for accidents in the Initial Handling Facility that do not credit the filtration system. As indicated previously, these results are only provided to illustrate that these filtration systems may not be required to meet regulatory standards, however, since they are an integral part of the current facility design, this SEIS does credit the filters in the analysis of impacts. The estimated annual frequencies of these events are consistent with the availability of the filters.

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 container from an emplacement drift for examination or drift maintenance. No additional accident scenarios unique to monitoring or closure were identified.

E.6 Inventory Modules 1 and 2 Accident Scenarios

Inventory Modules 1 and 2 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 of materials DOE would receive would increase, as would the period of emplacement operations. 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.

E.7 Representative Sabotage Scenario

In response to the terrorist attacks of September 11, 2001, and to intelligence information that has been 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. Additional measures have been imposed 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 waste more than 200 meters (660 feet) below the surface would offer significant impediments to any 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 any security threat. NRC requires liaison with response forces to permit timely response to unauthorized entry or activities. In addition, the NRC 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.

Although it is difficult to predict if sabotage events would occur, and the nature of such events if they were to occur, 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 pads 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 storage modules on the aging pads would be separated by 5.5 meters (18 feet) (DIRS 180195-BSC 2007, all) such that an aircraft crash into the pad could not damage more than a few of the modules, and (2) the storage canisters will 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).

Table E-13. Materials at risk for aircraft crash scenario.

	Commercial SNF assemblies						
	PWR		BWR	Naval SNF		HLW	DOE SNF
Initial Handling Facility				1 canister	or	5 canisters	
Canister Receipt and Closure Facility	42[a]						9 canisters
Receipt Facility	36[b]						
Wet Handling Facility	80[c]		120[c]				
	36[d]	or	74[d]				

Source: DIRS 182084-Wisenburg 2007, all.

a. Based on 2 TAD canisters with 21 PWR assemblies each.

b. Based on 1 DPC canister with 36 PWR assemblies.

c. These assemblies would be in the Wet Handling Facility pool.

d. These assemblies would be in the loading process (under water) from a DPC into a TAD canister.

BWR = Boiling-water reactor.
DOE = U.S. Department of Energy
DPC = Dual-purpose canister.
HLW = High-level radioactive waste.
PWR = Pressurized-water reactor.
SNF = Spent nuclear fuel.
TAD = Transportation, aging, and disposal (canister).

As shown 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 the release from damaged fuel assemblies. The next largest number of fuel assemblies from Table E-13 is 42 pressurized-water reactor fuel assemblies in a Canister Receipt and Closure Facility. As the table indicates, nine canisters of DOE spent nuclear fuel could be in the 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 will remain sealed while in the Canister Receipt and Closure Facility. Further, DOE spent nuclear fuel canisters will be designed to preclude a breach if dropped during handling operations (Section E.2.1.1). The canisters will be robust steel containers not expected to breach during the aircraft crash event. There could be as many as four TAD canisters in the Canister Receipt and Closure Facility at a given time (DIRS 182084-Wisenburg 2007, all). However, at least two of these canisters are expected to be in either a transportation cask awaiting removal or a sealed waste package awaiting transfer to the emplacement drifts (Chapter 2, Section 2.1.2.1.3). These TAD canisters are expected to be protected from the aircraft impact because of the thick steel walls of the transportation cask and waste package.

For the representative scenario, DOE assumed the aircraft penetrated 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 within the canisters. The fuel aboard the aircraft is conservatively assumed to 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 to be transported to the nearest receptor and be inhaled) and (2) the large fire from the jet fuel. DOE conservatively assumed that the fire would convert all of the fuel in the two TAD canisters (a total of 42 assemblies from pressurized-water reactor spent nuclear fuel) from uranium dioxide (UO_2) to uranium trioxide (U_3O_8) and produce a powder that contained radionuclides. Because all of the fuel pellet material in the 42 pressurized-water reactor fuel 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 radionuclide inventory in the assemblies was assumed to be the representative fuel (DIRS 180185-BSC 2007, all). This fuel 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 given time during the year (DIRS 180185-BSC 2007, all). The crud source term includes 209 curies of iron-55 and 16.9 curies of cobalt-60 per assembly (DIRS 180185-BSC 2007, all). Consistent with the FEIS analysis (DIRS 155970-DOE 2002, Appendix H, p. H-38), all of the iron and cobalt would be released because the Zircaloy cladding would burn during the accident. 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 provides the source term for the aircraft crash scenario.

The analysis used the GENII program to calculate the consequences from the crash with the assumptions in Section E.4.1, except that for this case, due to the large release and potential for large doses, the analysis did assume mitigation would occur. Mitigation measures would include evacuation of affected population after 24 hours and interdiction of contaminated crops so that 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}	$1.8 x 10^{0}$
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}	$2.7 x 10^{2}$
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}	$1.9 \times 10^{--4}$
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}	$4.7 \times 10^{—2}$

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	4.0 rem	2.4×10^{-3}[(a)]
80-kilometer (50-mile) population	1.3×10^{4} person-rem	7.8[b]

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 Postclosure Repository Performance

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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 *Draft 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-S1D) (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 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 SEIS* (DIRS 182846-SNL 2007, all) (TSPA-SEIS).

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-SEIS model it used for the analysis of postclosure performance in this Repository SEIS. Section F.3 identifies and quantifies the inventory of waste constituents of concern for analysis of postclosure performance. Section F.4 describes an estimate of how the impacts could change for locations beyond the location of the reasonably maximally exposed individual (RMEI). Section F.5 provides detailed results for waterborne radioactive material impacts, and Section F.6 provides the same 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 between data sources, process models, and the TSPA-SEIS 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-SEIS 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-SEIS model that the GoldSim program 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 modeling studies. They often are simple functions or tables of numbers. This process is called abstraction. It is necessary for

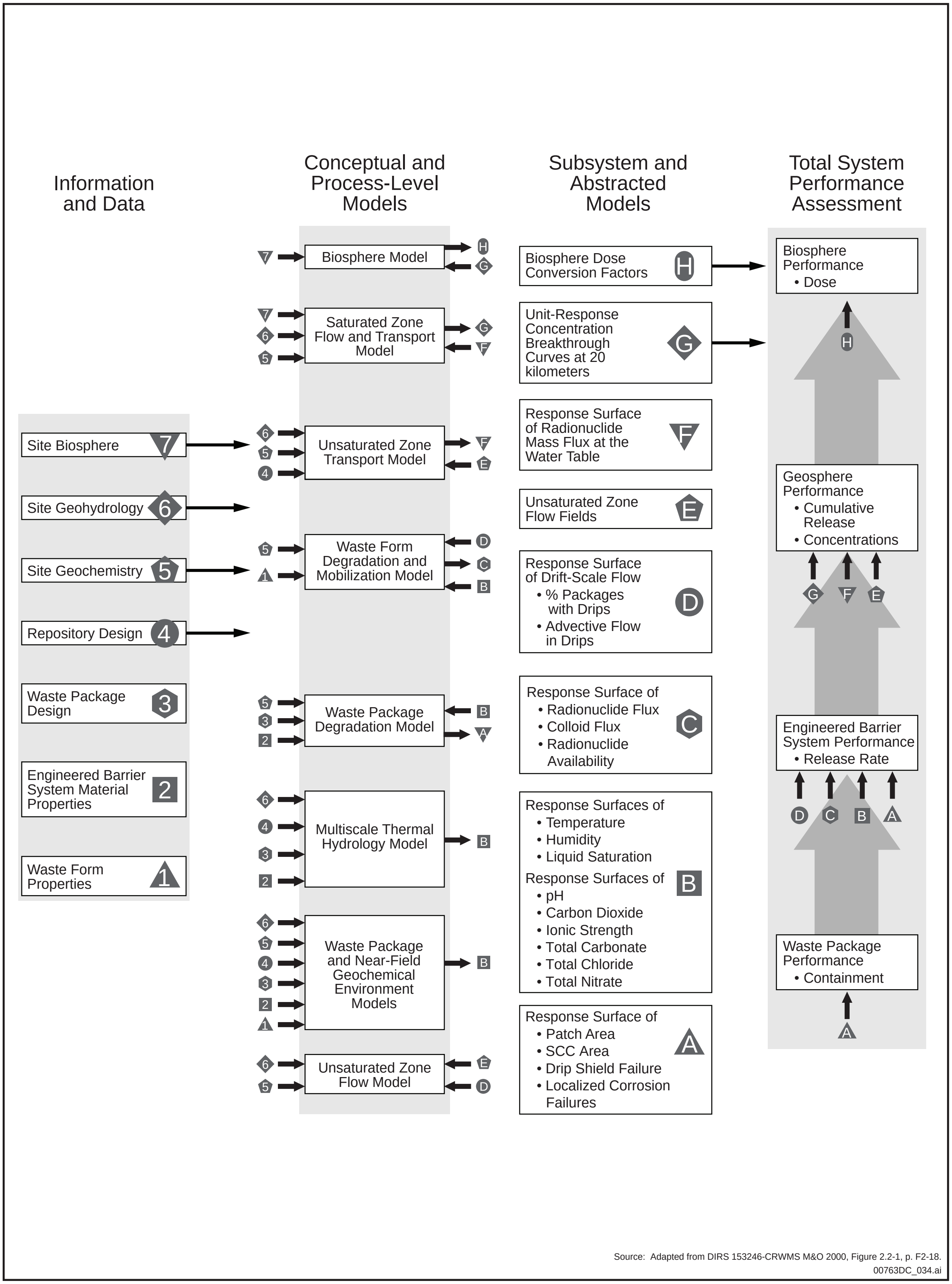

Figure F-1. Information flow in the TSPA-SEIS model.

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 can 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-SEIS model started with the model in the Yucca Mountain FEIS and includes several enhancements. Table 5-1 in Chapter 5 summarizes these enhancements.

The TSPA is a comprehensive systems analysis in which models of appropriate levels of complexity represent all important features, events, and processes to predict the behavior of the system under analysis and to compare this behavior to 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 prediction 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 would 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-SEIS 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 they can usefully be aggregated for the purposes of screening or analysis. The objective of scenario analysis for the TSPA-SEIS 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 of 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 features, events, and processes from the list:

- 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 and 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 the scenario classes listed above from the features, events, and processes retained after screening, its focus was on the 10,000-year compliance period. The proposed Environmental Protection Agency and Nuclear Regulatory Commission 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, the 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.

In the SEIS, the mechanical response of engineered barrier system components to seismic hazards was included in the TSPA-SEIS analysis of potential seismic events for the 10,000-year period after disposal and the period of geologic stability. The seismic hazards addressed included vibratory ground motion, fault displacement, and drift collapse due to ground motion. The major engineered barrier system components 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 internals and the waste form were also considered in structural response analyses. However, in the 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 the SEIS followed by a brief description of how that feature, event, and process was included in the TSPA-SEIS model.

FEP No. 1.2.03.02.0A: Seismic ground motion damages EBS components

Seismic activity that causes repeated vibration of the EBS 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 EBS 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 were 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-SEIS model for the Seismic Scenario Class.

FEP No. 1.2.03.02.0D: Seismic-induced drift collapse alters in-drift thermohydrology

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 impact flow pathways and condensation within the EBS, mechanisms for water contact with EBS components, and thermal properties within the EBS.

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-SEIS to account for the effect of drift collapse on thermal-hydrologic conditions in the drift for the Seismic Scenario Class.

FEP No. 1.2.03.02.0C Seismic-induced drift collapse damages EBS components

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 EBS 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 damage areas were used to develop drip shield and waste package damage abstractions that were implemented in the TSPA-SEIS model.

FEP No. 1.2.02.03.0A: Fault displacement damages EBS components

Movement of a fault that intersects drifts within the repository may cause the EBS components to experience related movement or displacement. Repository performance may 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 this analysis, estimates of very low probability fault displacement were compared with the dimensions of the engineered barrier 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-SEIS. The output of these abstractions is the number of drip shields and waste packages failed 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

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. The analysis assumed that all future climates would be similar to or wetter than current conditions. The climate model provided a forecast 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 U.S. Nuclear Regulatory Commission (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 used in the FEIS was completely revised for this 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 174294-SNL 2007, 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 5.960 were used to describe the probability of occurrence for each of the four infiltration scenarios, and therefore, the sum of the four weighting factors is one. The same weighting factors were used in all four climate states of present-day, monsoon, glacial transition, and post 10,000 years.

Comparisons between unsaturated zone flow model simulations using the four infiltration scenarios and subsurface measured 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 175177-SNL 2007, 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-SEIS 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% 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-SEIS model performance results.

The four post-10,000-year net infiltration rates presented in Table F-1 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

Table F-1. Average net infiltration rates (millimeters per year) 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 175177-SNL 2007, all.

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 175177-SNL 2007, all). Then all infiltration rates for that map are scaled such that the four target values for the average infiltration through the repository footprint are 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.

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. Water flowing in the fractures moves much more rapidly than water moving through the rock matrix. 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.

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 175177-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 175177-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-SEIS model used while the system model was running. The TSPA-SEIS model simply switched to the flow field for the sampled infiltration rate and climate state.

After the repository cooled, water returned to the repository walls. However, because of a capillary barrier effect at the drift wall only a small fraction of this returned water dripped into the emplacement drifts. The remaining water was 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 177395-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. 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. 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 177395-SNL 2007, all). All of these parameters were input as uncertainty distributions and sampled in the probabilistic TSPA-SEIS 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 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 rock fall 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 rock fall 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-SEIS 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 177405-SNL 2007, 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 177405-SNL 2007, 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 to packages near the center. DOE developed a multiscale modeling and abstraction method to couple drift-scale processes with mountain-scale processes (DIRS 177405-SNL 2007, 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 to 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 pre-emplacement 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 affected 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-SEIS 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, 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 predicting the amount of seepage into drifts is an important aspect of assessing the performance of the repository. The TSPA-SEIS seepage abstraction model is based on a synthesis of detailed modeling studies (DIRS 177395-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 178868-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 177405-SNL 2007, 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 (roof) 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 TAD canister. 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 induced corrosion) occurred the consequences to drip shield performance would be insignificant. 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 induced 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 rock fall under nominal conditions due to low consequence. However, it included mechanical failure of the drip shield and waste package by rock fall 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 the SEIS, drip shield failure by general corrosion occurred between approximately 260,000 years and 310,000 years, with the failure time different for each epistemic realization. In addition, because there was no spatial variability in drip shield corrosion rates, all drip shields in the repository failed 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 100,000 years to 1 million years, with the breaches caused by stress corrosion cracking in the weld of the outer closure lid. General corrosion failures would start at around 400,000 years and about 10 percent of the waste packages would experience a general corrosion breach within 1 million years. Diffusion was 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 was 7.7×10^{-6} square meter (DIRS 177407-SNL 2007, all). On average approximately 60 percent of the commercial spent nuclear fuel waste packages and 60 percent of the codisposal waste packages experienced a first breach by stress corrosion cracking by 1 million years. The average number of cracks per breached waste package at 1 million years was about 4. Advection and diffusion were the transport mechanisms acting to release radionuclides from a waste package when general corrosion breaches formed. On average only about 10 percent of the commercial spent nuclear fuel and codisposal waste packages experienced a general corrosion breach within 1 million years. The average number of general corrosion breaches (patches) at 1 million years was about 4. General corrosion breaches were represented by dividing the waste package surface into sub areas called patches. The total number of possible patches on a commercial spent nuclear fuel waste package is about 1,000 and on a codisposal waste package about 1,100.

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.

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-SEIS 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 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).

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.

It was conservatively assumed in the TSPA-SEIS 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 180178-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-SEIS model 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-SEIS model used empirical degradation rate formulas DOE developed for the three different waste forms to model degradation. DOE would combine defense 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 has 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-SEIS 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. The *Particle Tracking Model and Abstraction of Transport Processes* (DIRS 181006-SNL 2007, 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-SEIS 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 the these radionuclides that diffused into the matrix or were transported in the matrix of the Calico Hills tuff would prevent their movement or significantly reduce the rate of movement from the repository to the accessible environment.

- *Colloidal transport*. Several radionuclides could move in colloidal particles in the unsaturated zone. These include plutonium-239 and -240 and americium-241 and -243. 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 these radionuclides from the repository to the accessible environment. The analysis conservatively assumed that a small fraction of the colloids would be unretarded in the unsaturated zone. The unsaturated zone transport model includes sorption of colloidal transport of radionuclides.

- *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 to 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-SEIS 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 about 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. The major purpose of the Saturated Zone Flow and Transport Model Abstraction (DIRS 177390-SNL 2007, 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 flow model (DIRS 177391-SNL 2007, all) receives inputs from the unsaturated zone flow model and produces outputs, in the form of flow fields, for the saturated zone transport model. 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. The groundwater flow rates, the rate of transport of radionuclides, and the radionuclide retardation characteristics of these different rock types are significantly different. 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 177390-SNL 2007, 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 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. 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 177390-SNL 2007, 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. 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. 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 the parent species release 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 (DIRS 177399-SNL 2007, all) 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 (40 CFR Part 197) provides the definition for the reference person (the 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 to the appropriate Oral Reference Dose.

The biosphere is the last component in the chain of TSPA-SEIS 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 to the saturated zone flow and transport model; and the other is for the Volcanic Eruption Modeling Case, where the biosphere is coupled to 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-SEIS 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 simulate the significant features, events, and processes: The first is 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; the second is 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)
- 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-SEIS 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

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.

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.

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 13 kilometers (8 miles), and would be dispersed by wind to the accessible environment. 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-SEIS 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: The Seismic Ground Motion Modeling Case includes waste packages that would fail solely due to the ground motion damage associated with the seismic event; 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-SEIS model components to evaluate the mobilization of radionuclides 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 shown in Figure F-2, 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

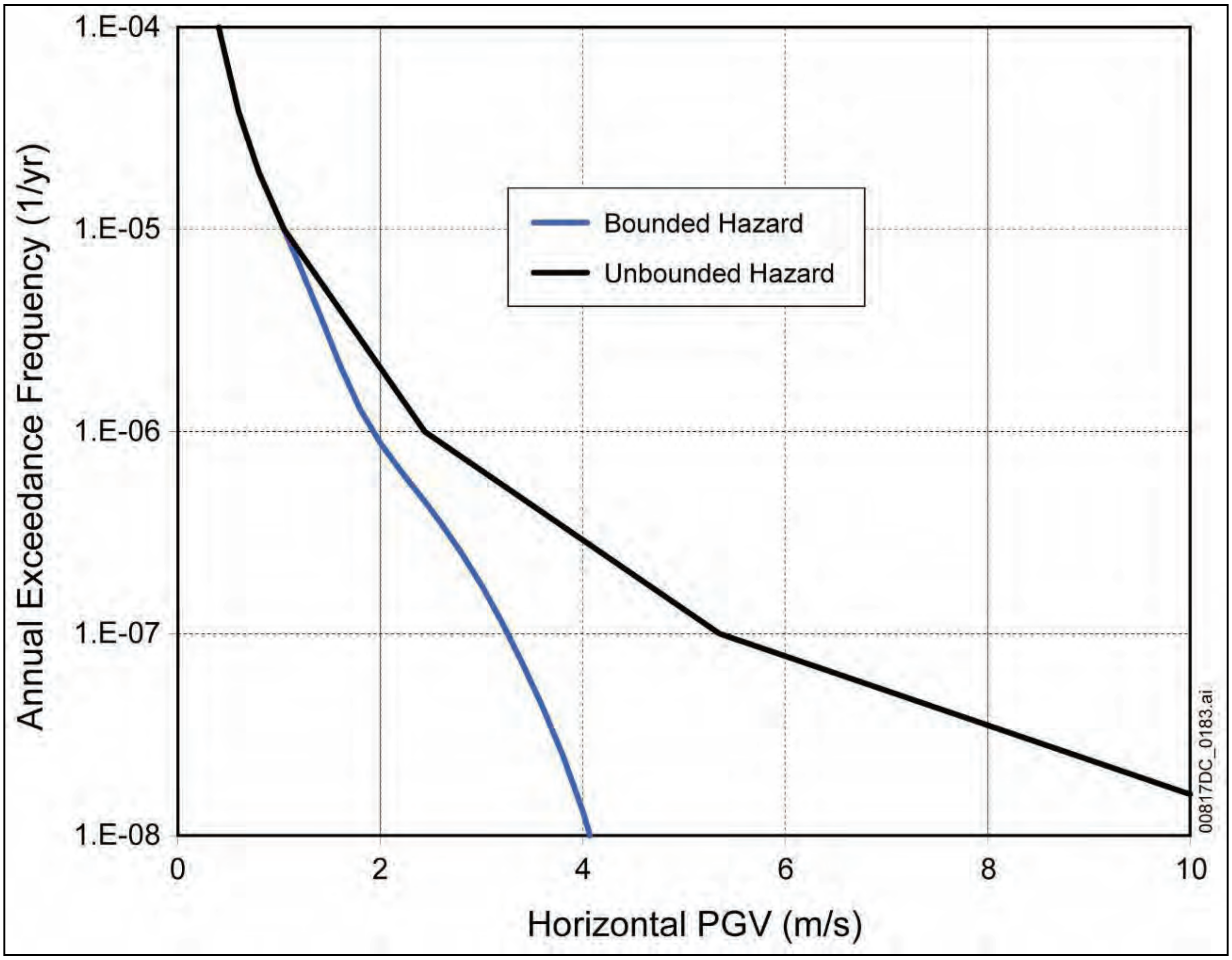

Figure F-2. Hazard curve for the seismic scenario class.

horizontal ground velocity reflects the use of that ground motion measure to set parameters for rock fall 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, rock fall, 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 transportation, aging, and disposal (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 abstractions. General corrosion thins and weakens the drip shields and waste packages over long time 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-SEIS seismic damage calculations. Last, the TSPA-SEIS 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-SEIS.

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 meters per second and mean exceedance frequencies smaller than 4.28×10^{-4} per year. Seismic events were modeled as Poisson processes that were generated randomly with the specified rate of 4.28×10^{-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 4.287×10^{-4} per year). The duration of the dose assessment ends 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 428 events (computed by multiplying the specified rate of the Poisson process, 4.28×10^{-4} per year, by the simulation time period of 1 million years), so multiple seismic events would occur in each realization of the TSPA-SEIS model. 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 transport, aging, and disposal 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 corresponding to exceedance frequencies between 1.0×10^{-4} per year and 4.28×10^{-4} per year. The extent of drift collapse, rock fall, and damage to the waste packages and drip shields was determined from rock fall 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 rock fall 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 is intact and protects the waste package from rock fall.
- 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 rock fall 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 rock fall 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 continues to act as a seepage barrier, but mechanically loads the waste package outer barrier with static and dynamically-amplified rubble loads. This accounts for future states of general corrosion thinning of the drip shield framework, 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 fails as a seepage and rock fall barrier, with subsequent rubble in direct contact with the waste package outer barrier, thus applying static and dynamically-amplified rubble loads. This accounts for future states of general corrosion thinning of the drip shield plates, 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-SEIS 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 was accelerated stress corrosion cracking in the damaged areas that exceeds 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 has 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 2×10^{-7} per year. 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, could 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 182846-SNL 2007, 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 the SEIS.

F.3 Inventory

This section discusses the inventories of waterborne radioactive materials DOE used to estimate radiological impacts, and some 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. In the Proposed Action, these radionuclides would be present in five basic waste forms—commercial spent nuclear fuel, mixed oxide fuel and plutonium ceramic (plutonium disposition waste), borosilicate glass formed from liquid wastes on DOE sites (high-level radioactive waste), DOE spent nuclear fuel, and 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-SEIS model with each waste package and its unique design would result in a model too large to run on any available computer in a practical time.

DOE developed the abstracted inventory to maintain essential characteristics of the waste forms for input to the TSPA-SEIS model. The TSPA-SEIS 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 or 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 fuel and vitrified high-level radioactive waste. For this analysis, naval spent fuel was conservatively modeled as commercial spent fuel (DIRS 182846-SNL 2007, 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 waste container) (DIRS 177424-SNL 2007, all). Table F-3 summarizes the abstracted inventory. Note that, as discussed in Chapter 5, Section 5.2.1, the TSPA simulations presented in this Repository SEIS for the first 10,000 years after closure were not based on the 32 radionuclides listed in Table F-3, but on 29 radionuclides. The three radionuclides—chlorine-36, selenium-79, and tin-126—were excluded from the assessment of postclosure repository performance for the first 10,000 years after repository closure. The exclusion of these three radionuclides from the analysis had an insignificant impact on projected doses as shown. Note also 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-SEIS 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

Table F-3. Initial radionuclide inventories (grams per package) in 2117 for each idealized waste package type in the TSPA-SEIS model.[a]

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 182846-SNL 2007, all.

a. 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.

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 (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 incorporated 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 periods up to 10,000 years after closure and the post-10,000-year period. The dose calculated for comparison to individual protection standards is 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 the farthest southern point at the boundary of the controlled area and the accessible environment (40 CFR 197.12). This location is about 18 kilometers (11 miles) downgradient from the repository. 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 (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-SEIS 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-SEIS 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

> **COLOR FIGURES**
>
> The figures illustrating results of the performance analysis presented in Chapter 5 and Appendix F can be found in color at the Office of Civilian Radioactive Waste Management website: http://www.ocwrm.doe.gov/

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 40 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. As a result, the projected doses at 10,000 years, for the 10,000 years and post-10,000-year calculations, would in general be different but sufficiently accurate to project groundwater concentrations and mean and median annual doses.

A plot of multiple dose history curves is called a horsetail plot. The plot of Nominal Scenario Class Modeling Case results in Figure F-3 is an example of a horsetail plot. Each dose-versus-time curve in Figure F-3 represents the estimates of calculated time-dependent dose for a single realization or sample of epistemic uncertainty. The TSPA-SEIS model generated the entire set of dose-versus-time curves by repeating this process a number of times in a single looping process.

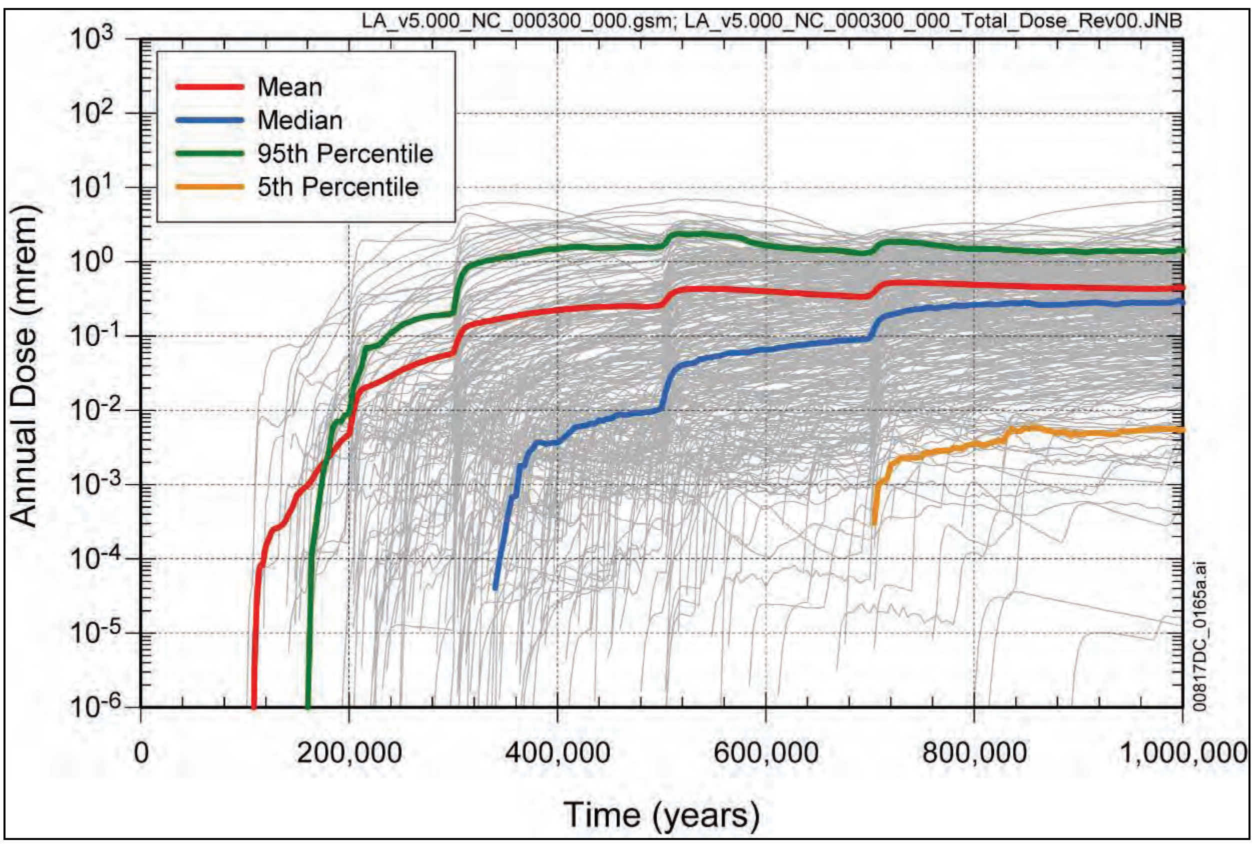

Figure F-3. Projected annual dose for the Nominal Scenario Class Modeling Case for the post-10,000-year period.

For the Repository SEIS, DOE calculated these statistical measures for 300 epistemic realizations at each time step of the projected annual individual dose histories. The plot of the mean represents the average of all 300 data points at each time step. For each point on the plot of the median dose, 50 percent of the data have a value greater than the plotted point and 50 percent have a value less than the plotted point. Similarly, for the 5th- and 95th-percentiles, the plotted data points are such that 95 percent of data are greater than the plotted point and 5 percent of the data points are greater than the plotted points, respectively, for each time step.

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 section and subsequent sections, impacts from repository performance are described using annual dose histories that illustrate the mean and median annual doses calculated for the different modeling cases. In addition, dose histories of major radionuclides that contribute to the estimate of mean annual dose are also 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-SEIS 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. This 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 because no waste packages are estimated to fail (by general corrosion, localized corrosion, or stress corrosion cracking) in this period. The first waste package failure (by nominal stress corrosion cracking) would occur at approximately 100,000 years, and the drip shields would begin to fail by general corrosion at approximately 260,000 years. As shown in Figure F-3, 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 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

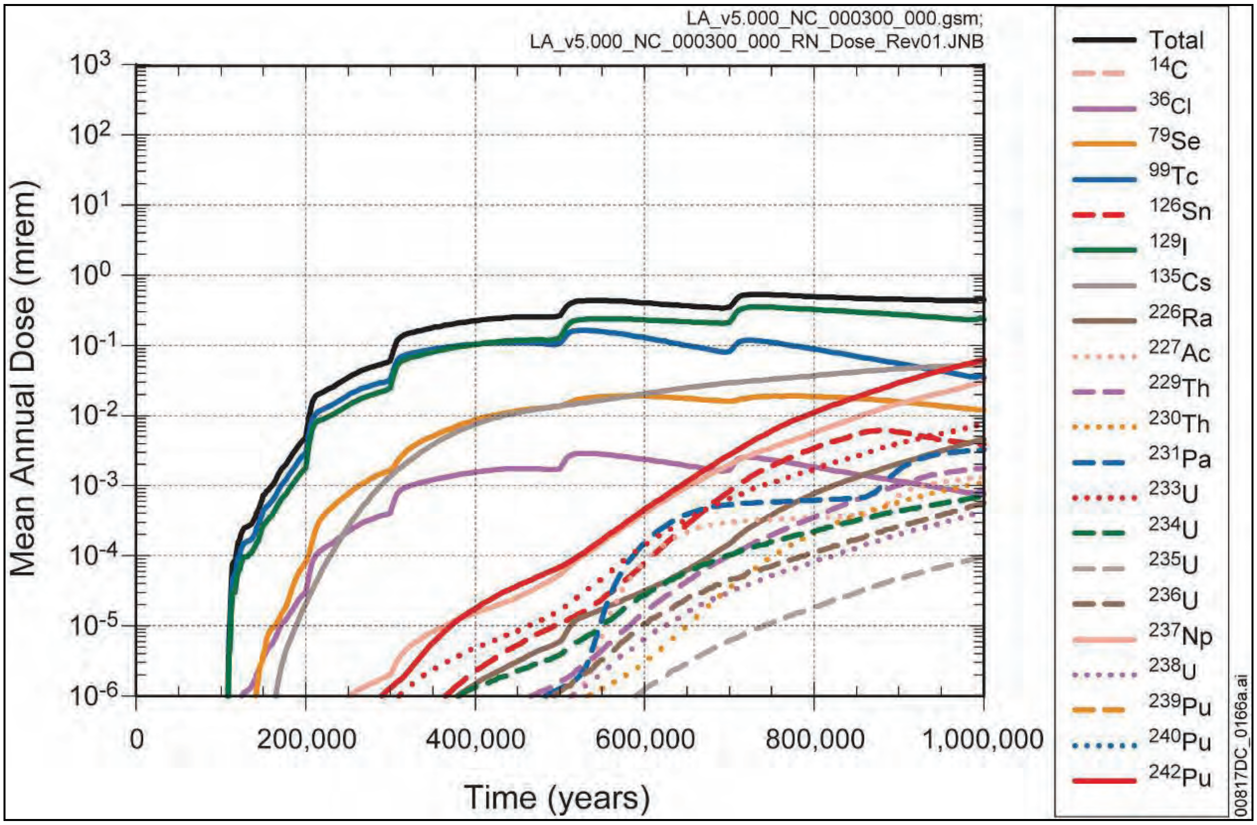

Figure F-4. Mean annual dose histories of major radionuclides for the Nominal Scenario Class Modeling Case for the post-10,000-year period.

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-SEIS 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.

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 dose 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 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-240, 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 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 about 0.004 millirem and would occur at about 9,800 years. Annual doses would increase after the climate changed at 10,000 years. The projected mean and median annual doses reach levels of about 0.2 and 0.006 millirem, respectively, before 15,000 years and gradually 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.

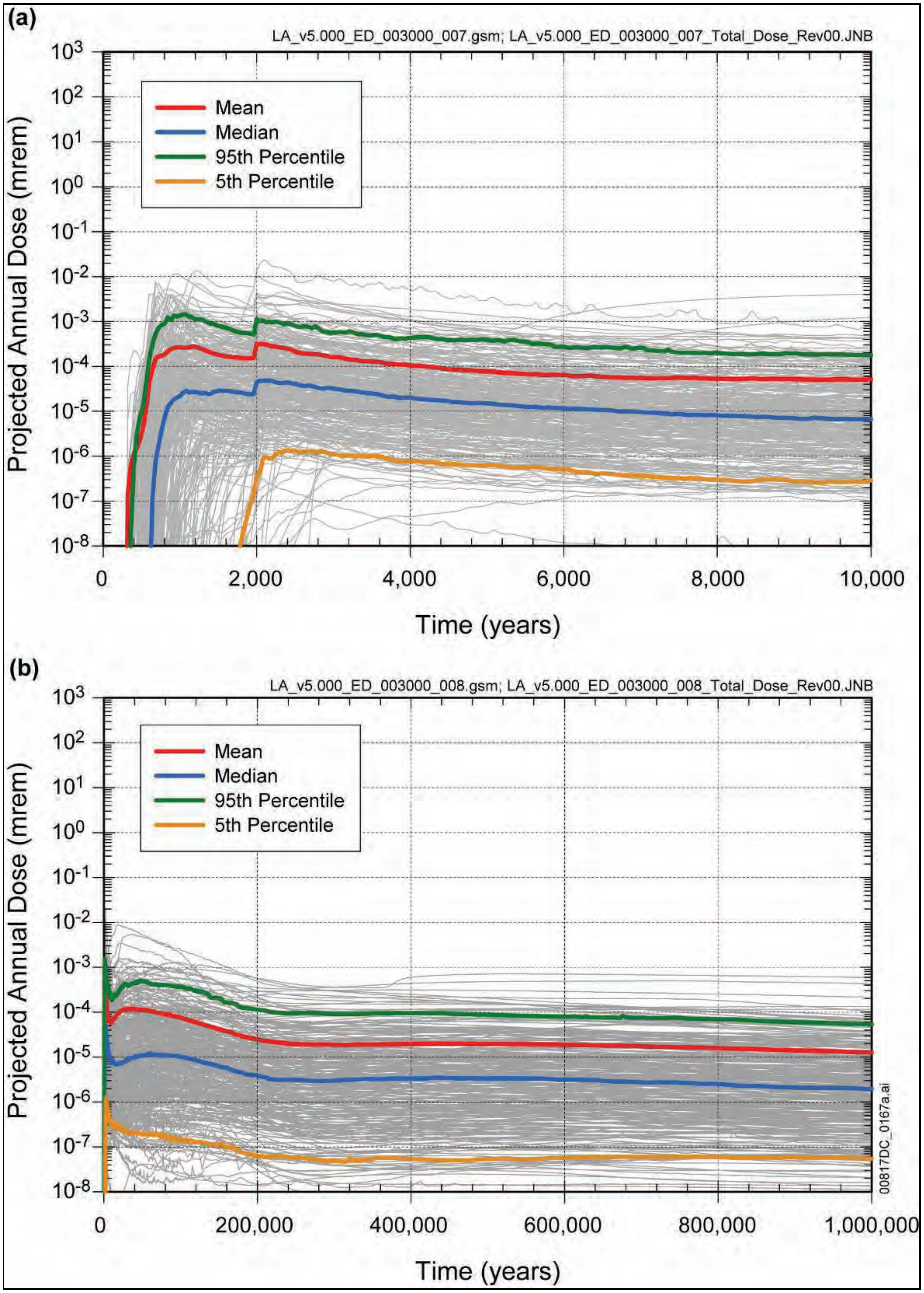

Figure F-5. Projected annual dose for the Drip Shield Early Failure Modeling Case for (a) the first 10,000 years after repository closure and (b) post-10,000-year period.

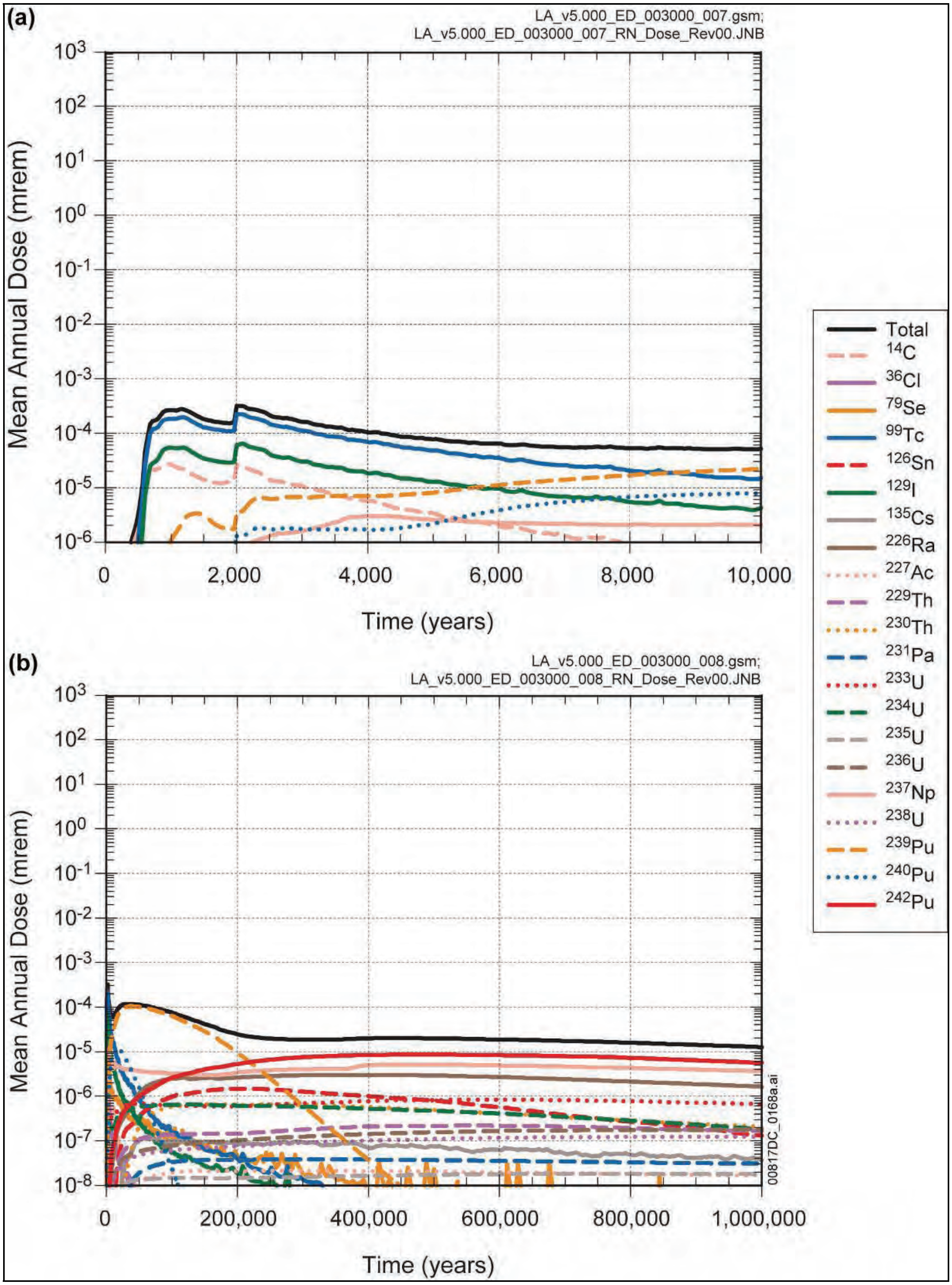

Figure F-6. Mean annual dose histories of major radionuclides for the Drip Shield Early Failure Modeling Case for (a) the first 10,000 years after repository closure and (b) post-10,000-year period.

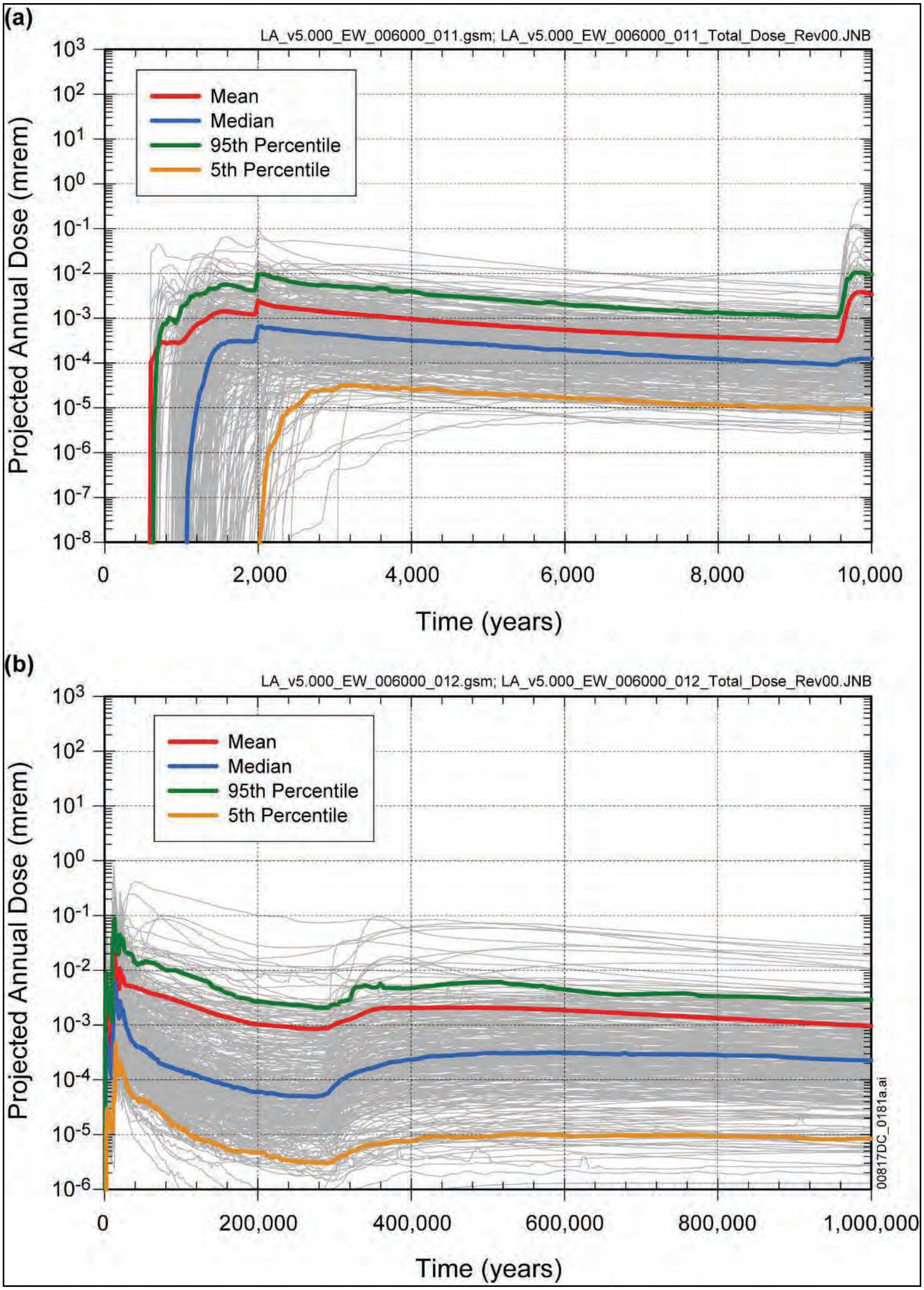

Figure F-7. Projected annual dose for the Waste Package Early Failure Modeling Case for (a) the first 10,000 years after repository closure and (b) post-10,000-year period.

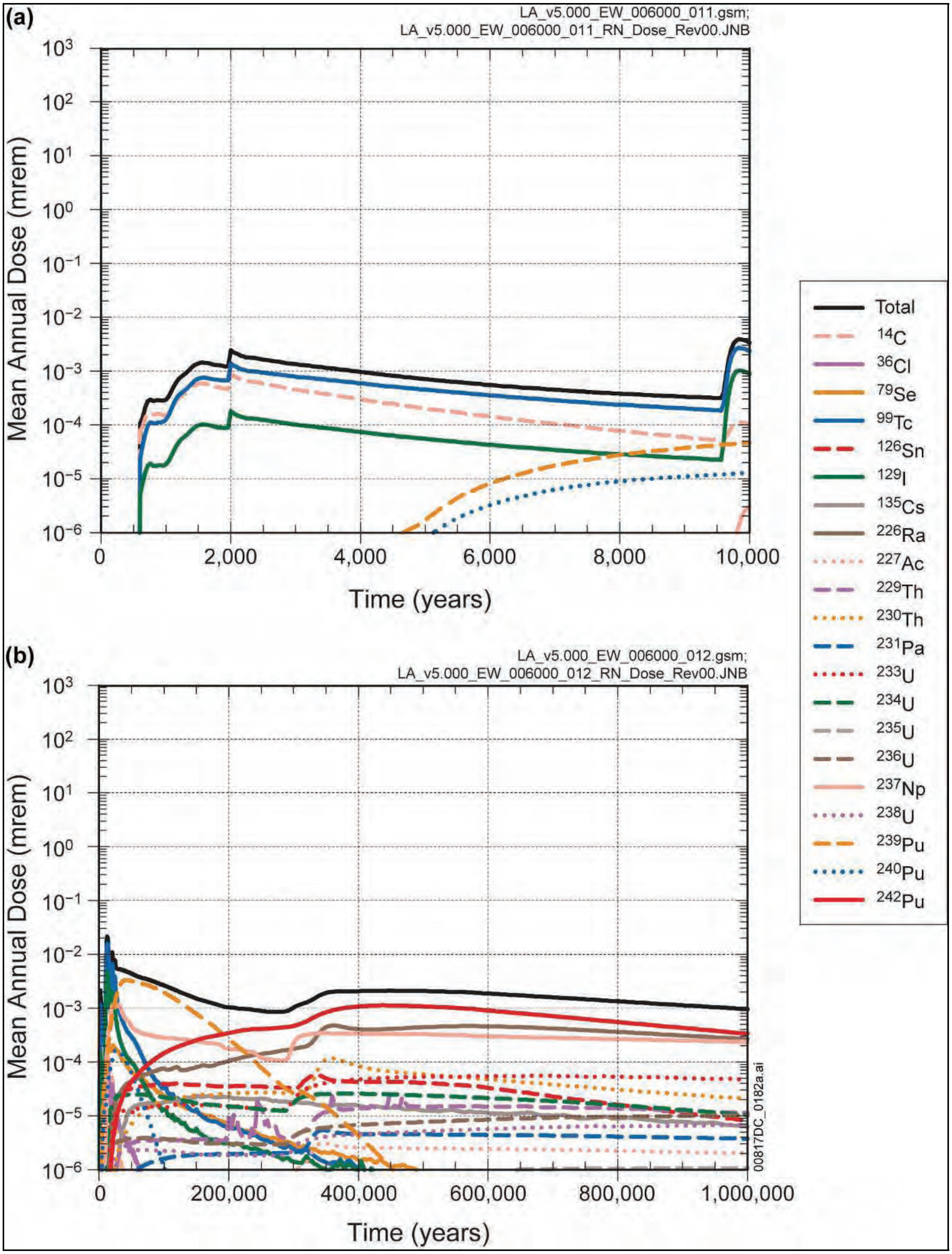

Figure F-8. Mean annual dose histories of major radionuclides for the Waste Package Early Failure Modeling Case for (a) the first 10,000 years after repository closure and (b) post-10,000-year period.

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. 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 that 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 that 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 Modeling 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, 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 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 0.06 millirem and for the post-10,000-year period is about 1.3 millirem. The median projected annual dose for the post-10,000-year period is less than 0.4 millirem.

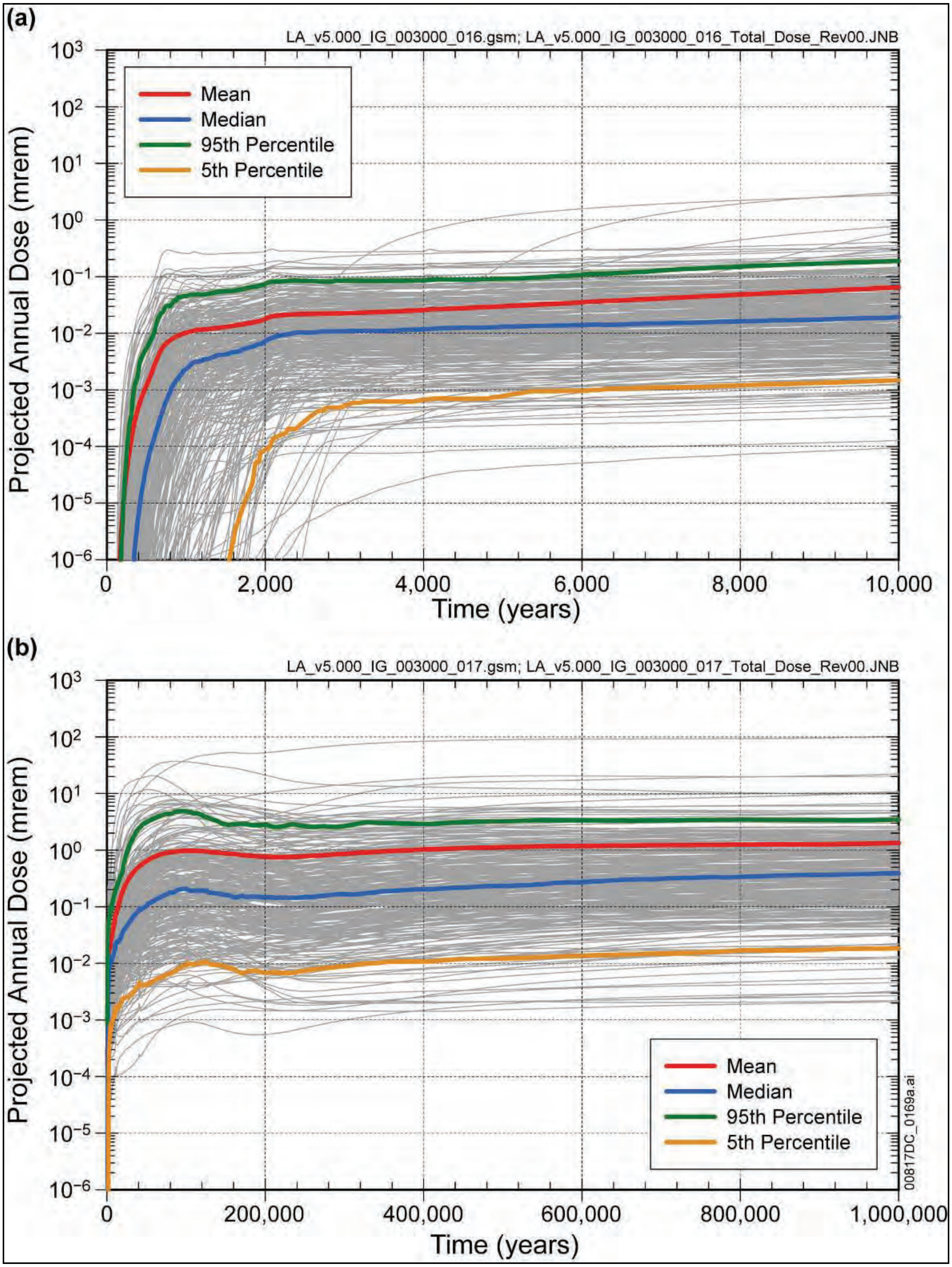

Figure F-9. Projected annual dose for the Igneous Intrusion Modeling Case for (a) the first 10,000 years after repository closure and (b) post-10,000-year period.

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 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 radium-226, plutonium-242, and neptunium-237 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 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 77 percent of intrusive events that intersected the repository would be due to 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 first 10,000 years after closure and 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. These figures 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.

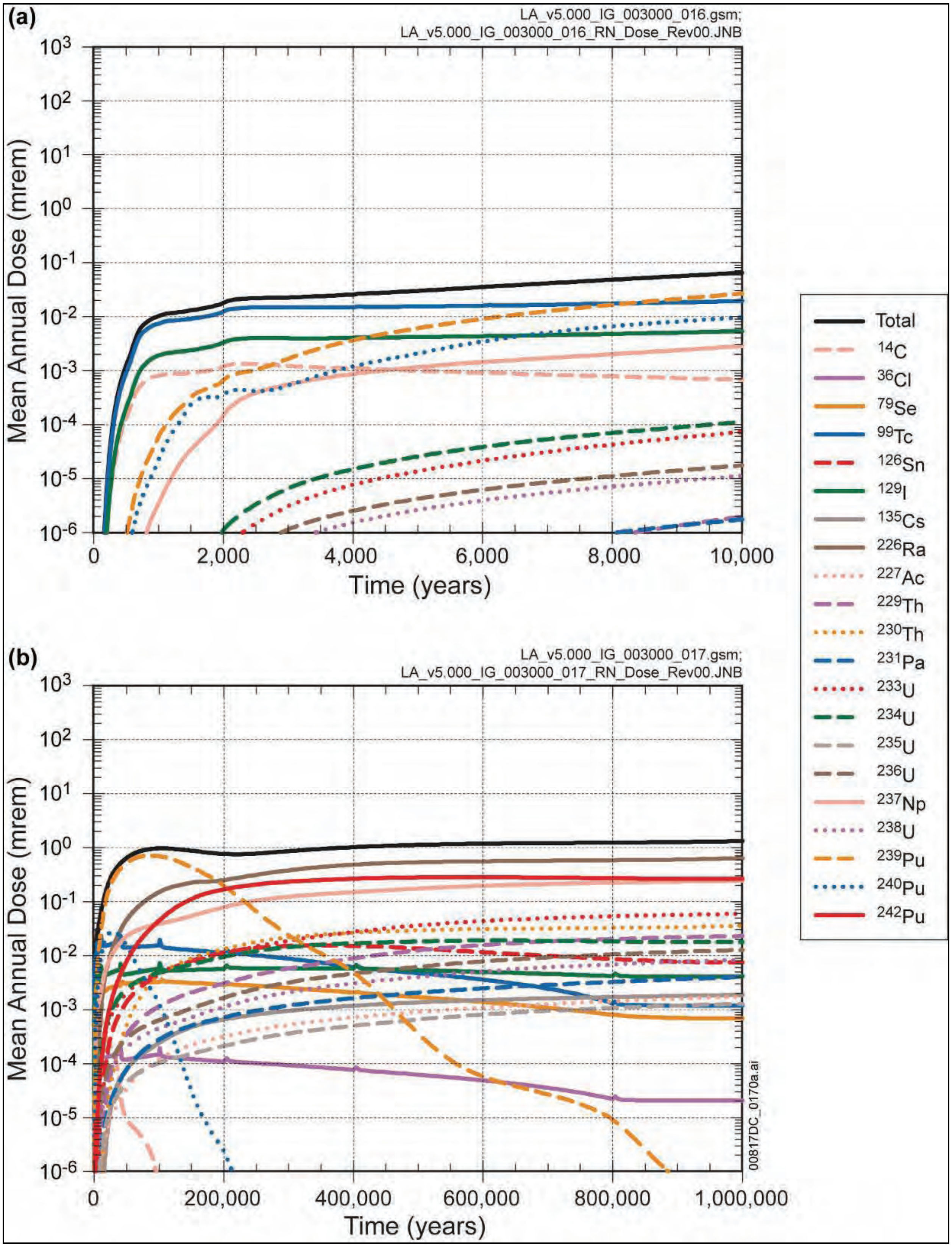

Figure F-10. Mean annual dose histories of major radionuclides for (a) the Igneous Intrusion Modeling Case for the first 10,000 years after repository closure and (b) post-10,000-year period.

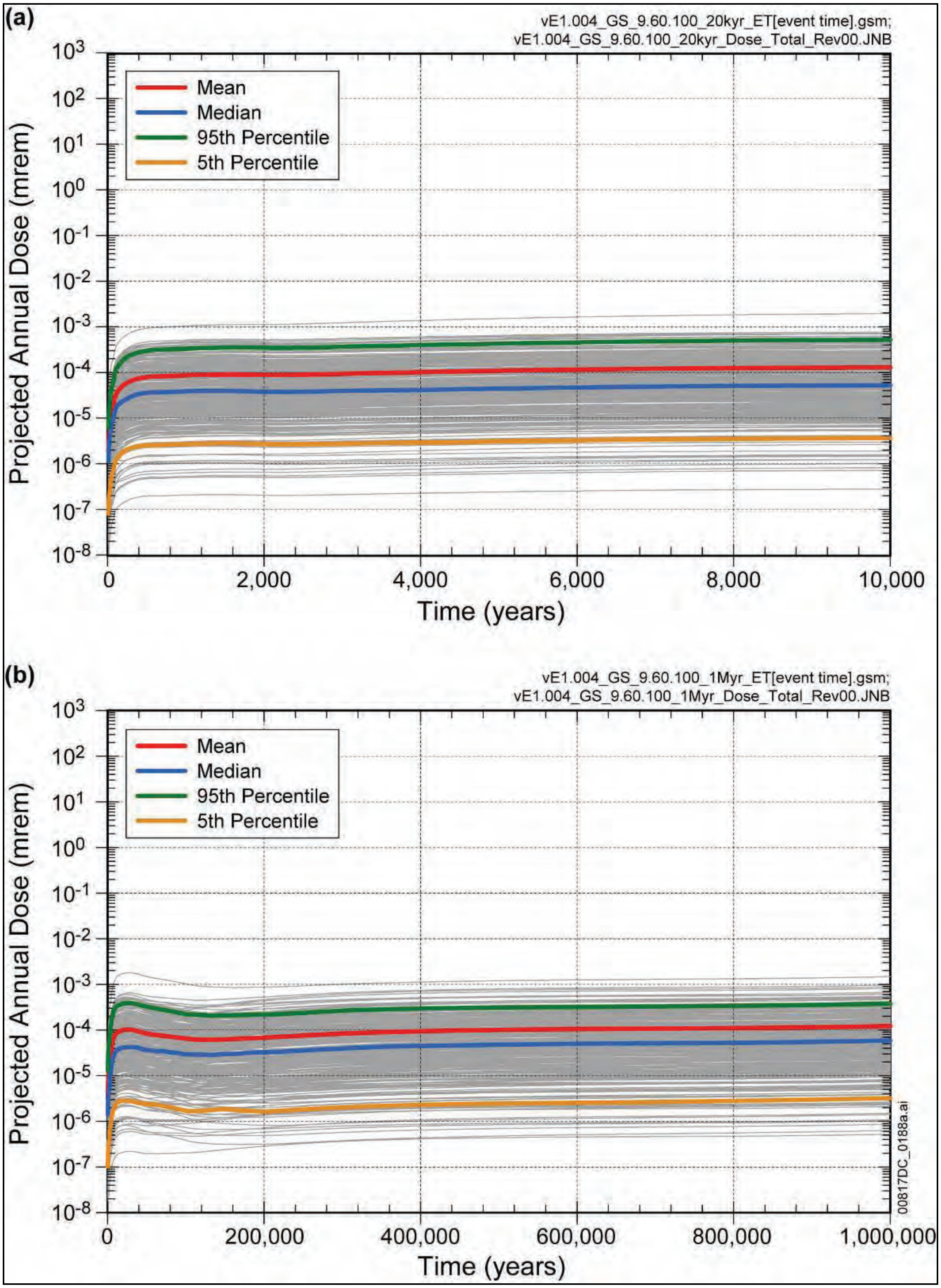

Figure F-11. Projected annual dose for the Volcanic Eruption Modeling Case for (a) the first 10,000 years after repository closure and (b) post-10,000-year period.

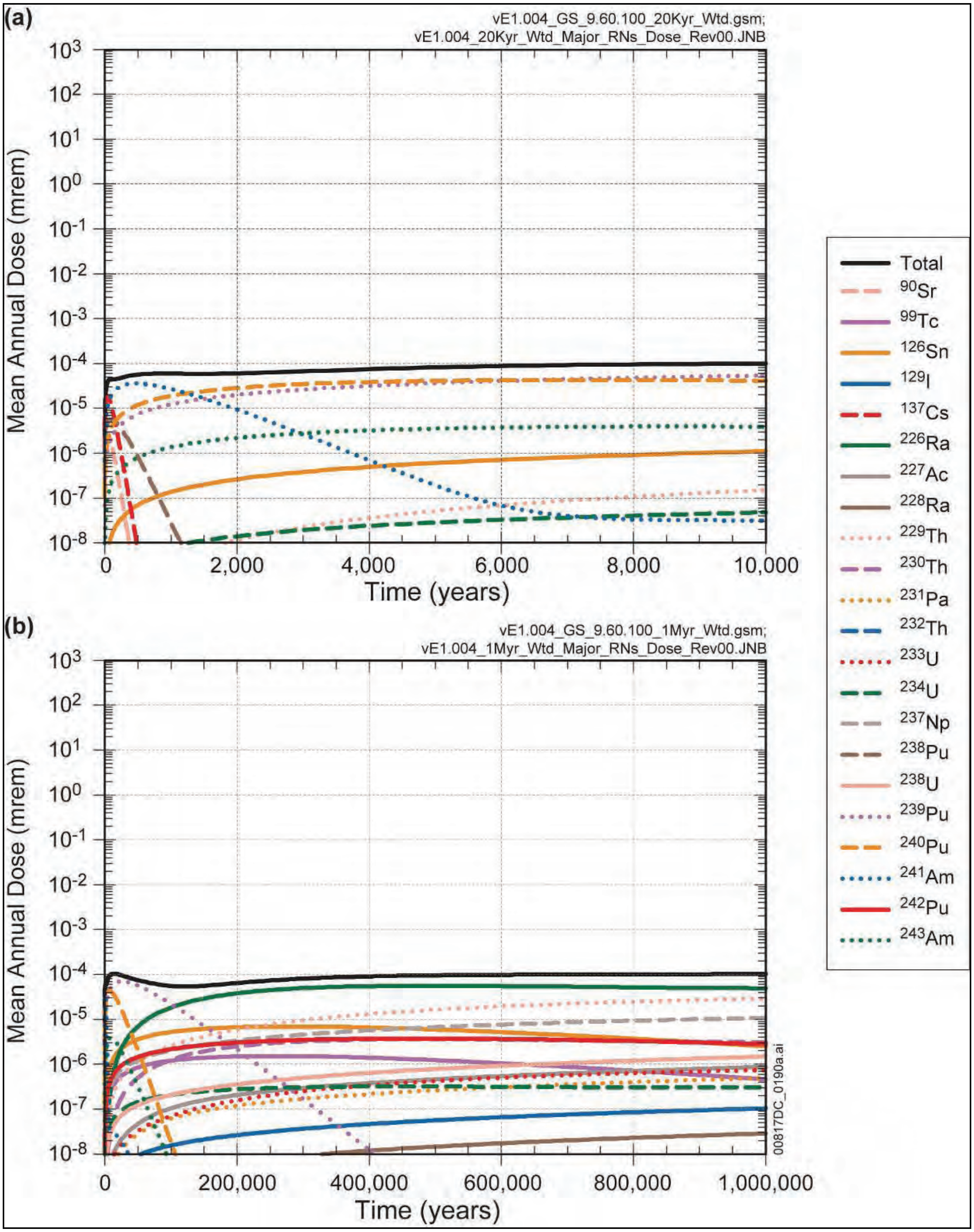

Figure F-12. Mean annual dose histories of major radionuclides for the Volcanic Eruption Modeling Case for (a) the first 10,000 years after repository closure and (b) post-10,000-year period.

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 became 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.

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, tearing or rupture, localized corrosion, 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 event’s 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 1.5 millirem. The median projected dose for the post-10,000-year period is less than 0.5 millirem. The spikes in the results correspond to the occurrence of seismic events of sufficient magnitude to cause damage to the waste packages. These spikes occur in each realization at the same time because each epistemic realization has essentially the same set of future conditions. That is, each epistemic realization has the same number of events, the same event times, and the same event magnitudes. As a result, all epistemic realizations and their spikes reinforce each other in the calculation of the mean and median annual doses and cause the spikes to become more pronounced.

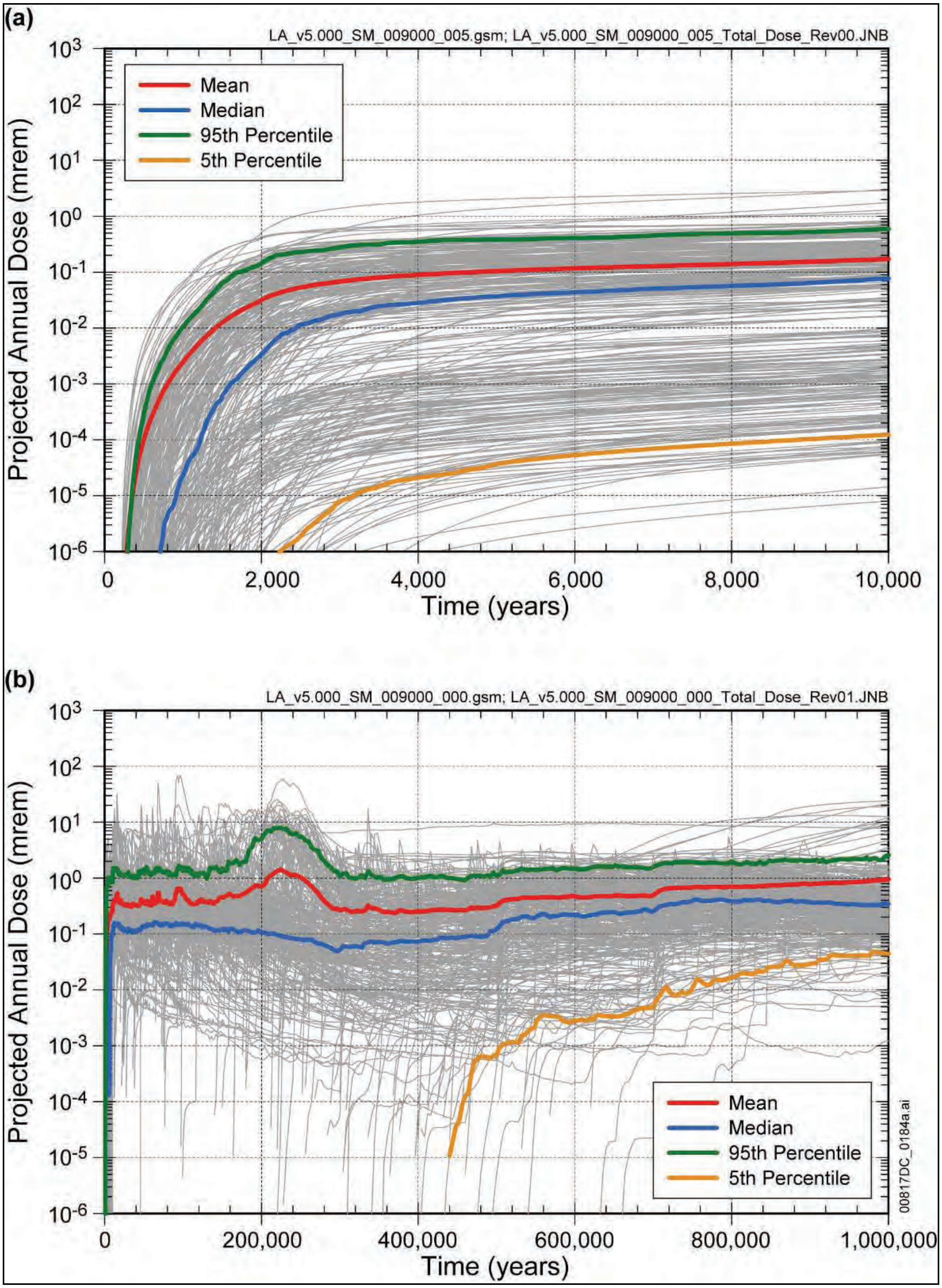

Figure F-13. Projected annual dose for the Seismic Ground Motion Modeling Case for (a) the first 10,000 years after repository closure and (b) post-10,000-year period.

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, and iodine-129 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 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. 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.

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.

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 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 is less than 0.002 millirem and for the post-10,000-year period would be approximately 0.02 millirem. The median projected dose for the post-10,000-year period is 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 technicium-99 would dominate the mean at 100,000 years and 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.

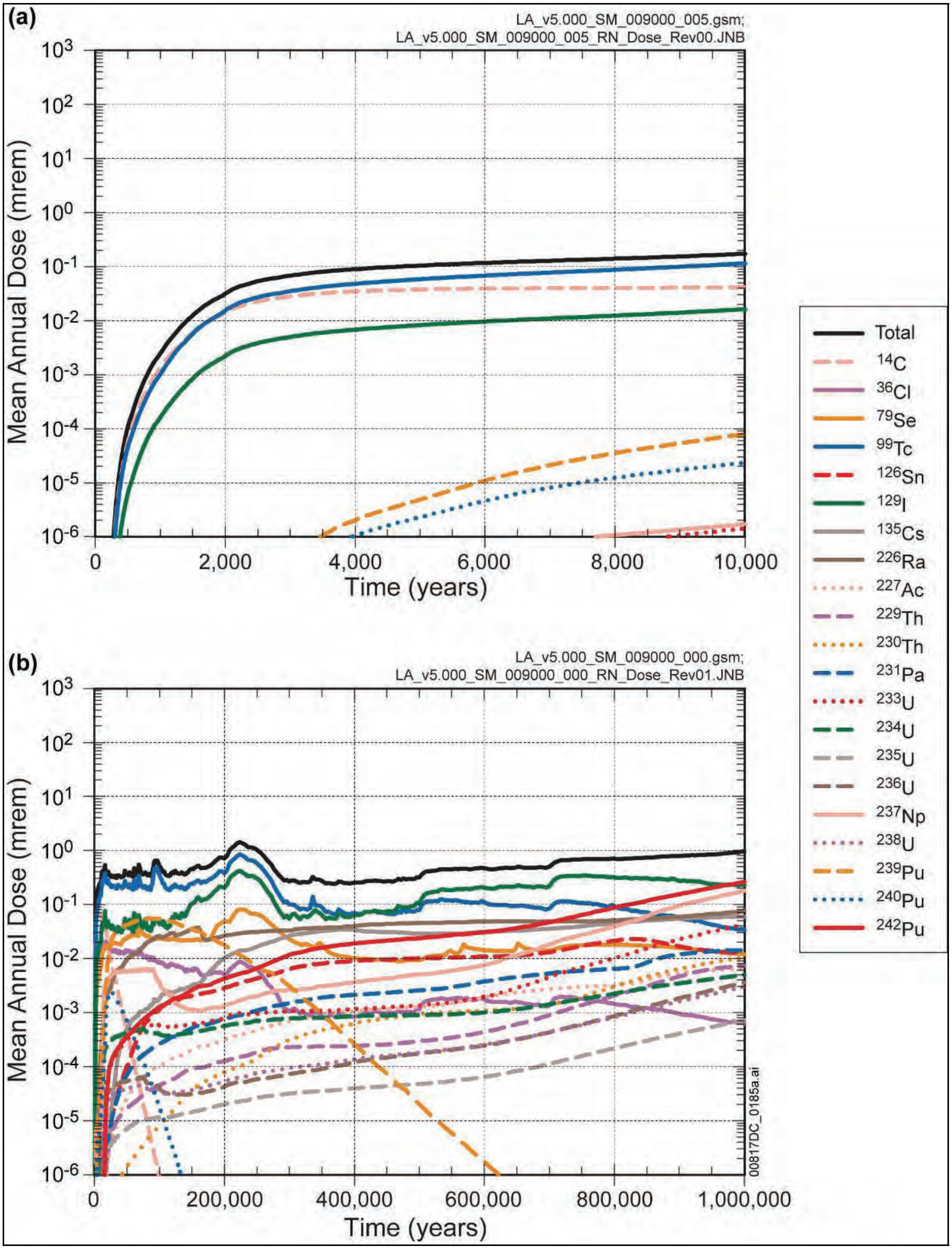

Figure F-14. Mean annual dose histories of major radionuclides for the Seismic Ground Motion Modeling Case for (a) the first 10,000 years after repository closure and (b) post-10,000-year period.

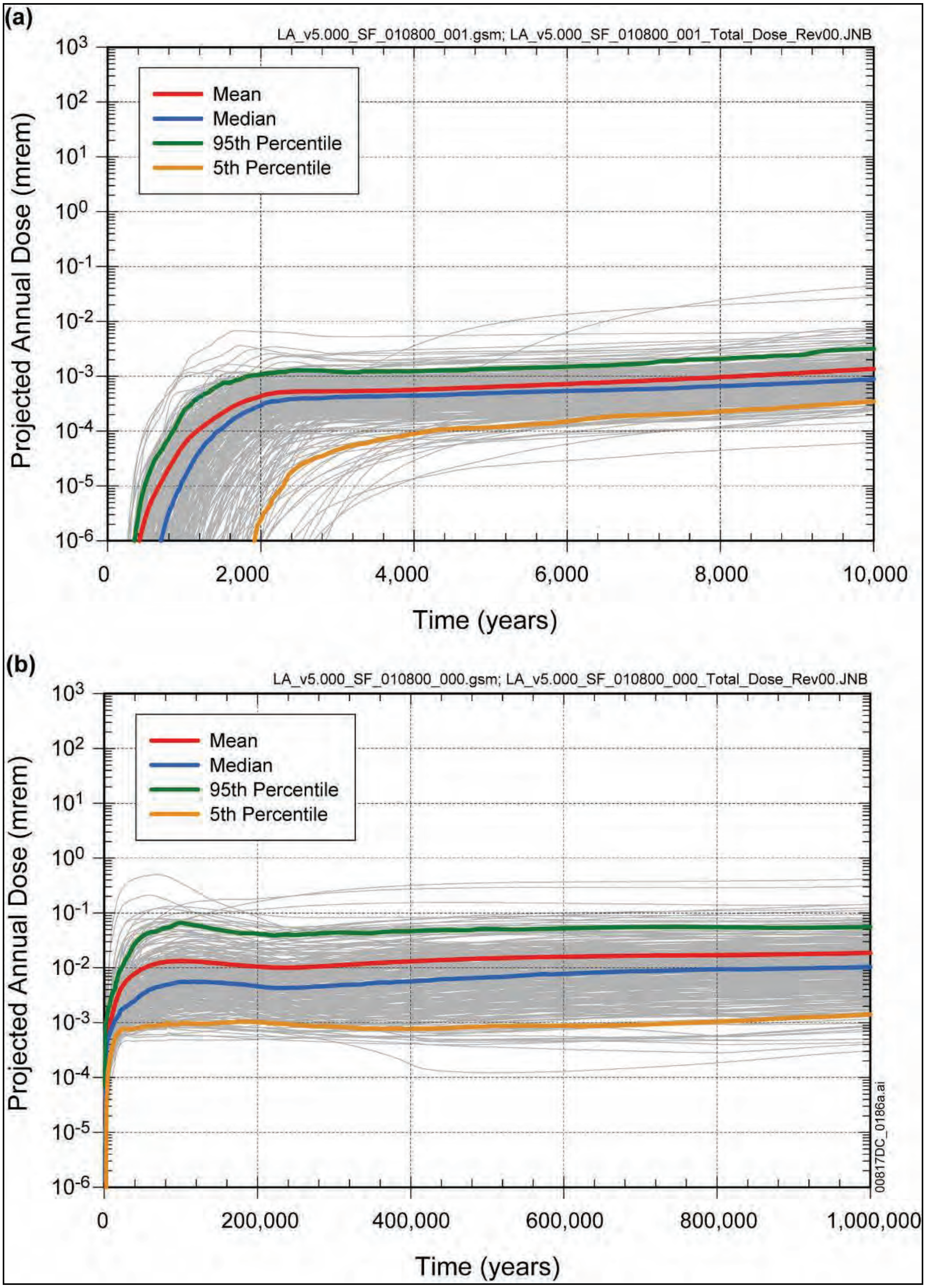

Figure F-15. Projected annual dose for the Seismic Fault Displacement Modeling Case for the first 10,000 years after repository closure and post-10,000-year period.

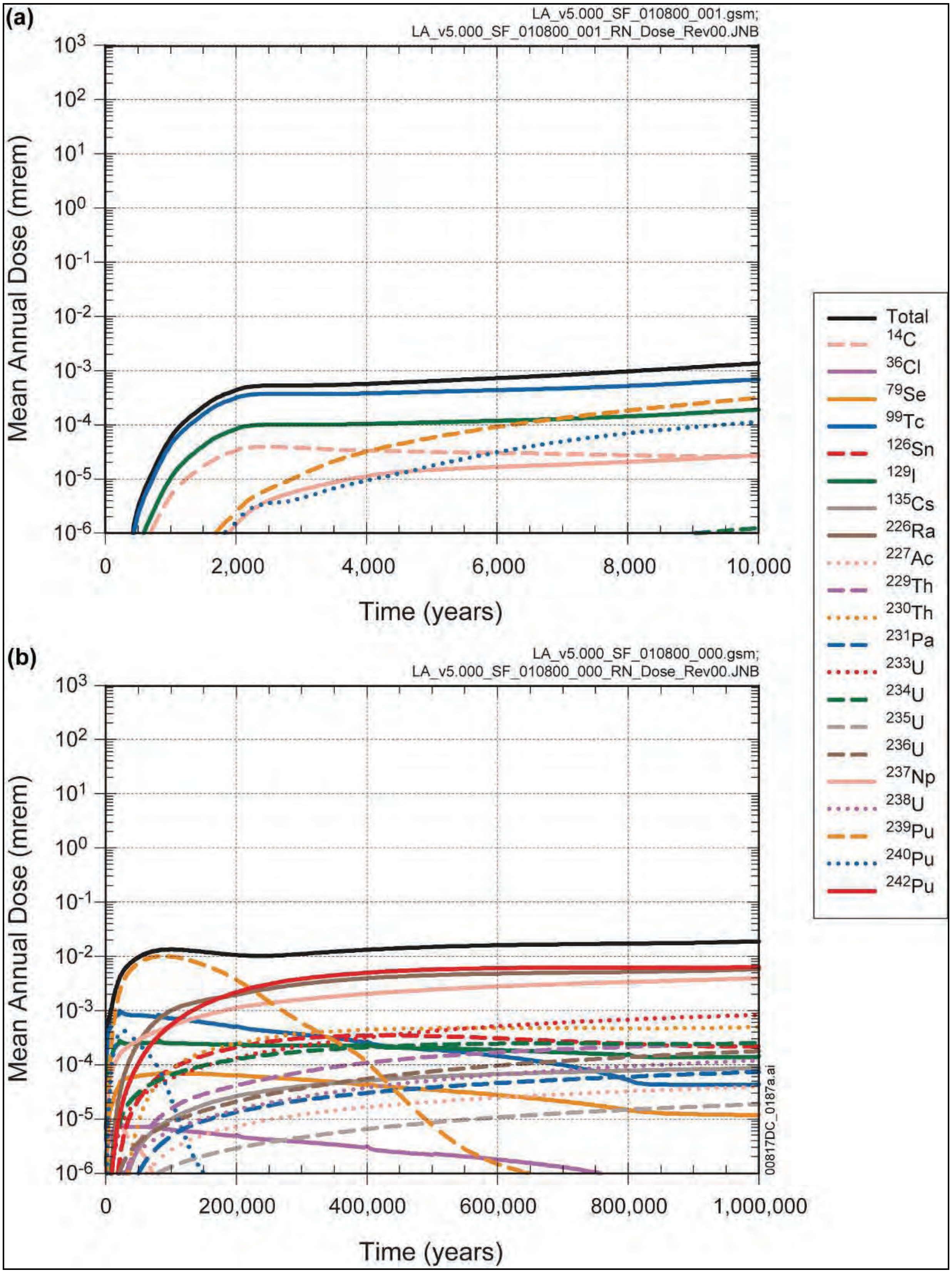

Figure F-16. Mean annual dose histories of major radionuclides for the Seismic Fault Displacement Modeling Case for the first 10,000 years after repository closure and post-10,000-year period.

Equation F-1 represents the distribution for total expected annual dose $\overline{D}_T(\tau, \boldsymbol{e_i})$ as a function of time τ:

$$\overline{D}_T(\tau, \boldsymbol{e_i}) = \overline{D}_N(\tau, \boldsymbol{e_i}) + \overline{D}_{EF}(\tau, \boldsymbol{e_i}) + \overline{D}_I(\tau, \boldsymbol{e_i}) + \overline{D}_S(\tau, \boldsymbol{e_i}) \qquad \text{(Equation F-1)}$$

where e_i denotes a realization or sampling of epistemic uncertainty *i* (Chapter 5, Section 5.3.4.2.1) and *i = 1, 2,…*. 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, 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 barrier to damage from seismic events. For the post-10,000-year, the expected annual dose for the Nominal and the Seismic Scenario Classes are combined, and are computed as:

$$\overline{D}_N(\tau, \boldsymbol{e_i}) + \overline{D}_S(\tau, \boldsymbol{e_i}) = \overline{D}_{GM}(\tau, \boldsymbol{e_i}) + \overline{D}_{FD}(\tau, \boldsymbol{e_i}) \qquad \text{(Equation F-2)}$$

where $\overline{D}_{GM}(\tau, \boldsymbol{e_i})$ is the expected annual dose from seismic ground motion events 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 182846-SNL 2007, 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.

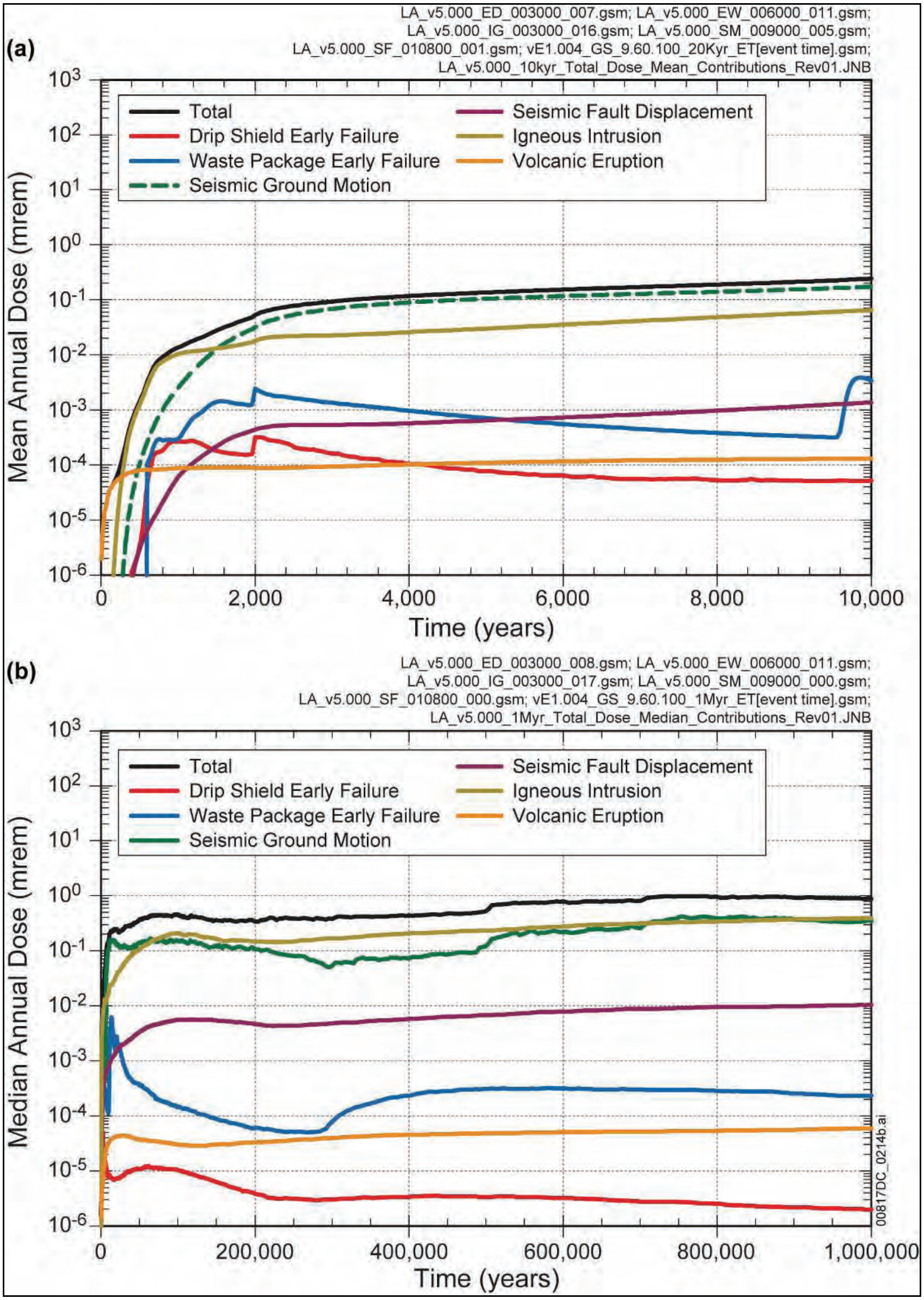

Figure F-17. Total mean annual dose and median annual doses for each modeling case for (a) the first 10,000 years after repository closure and (b) post-10,000-year period.

F.4.4 COMPARISON TO GROUNDWATER PROTECTION STANDARDS

DOE excluded unlikely natural processes and events from the performance calculations to evaluate conformance with groundwater protection, as required by the 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.

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 (0.53 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 (1.3 ounces per gallon) of total dissolved solids, and 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

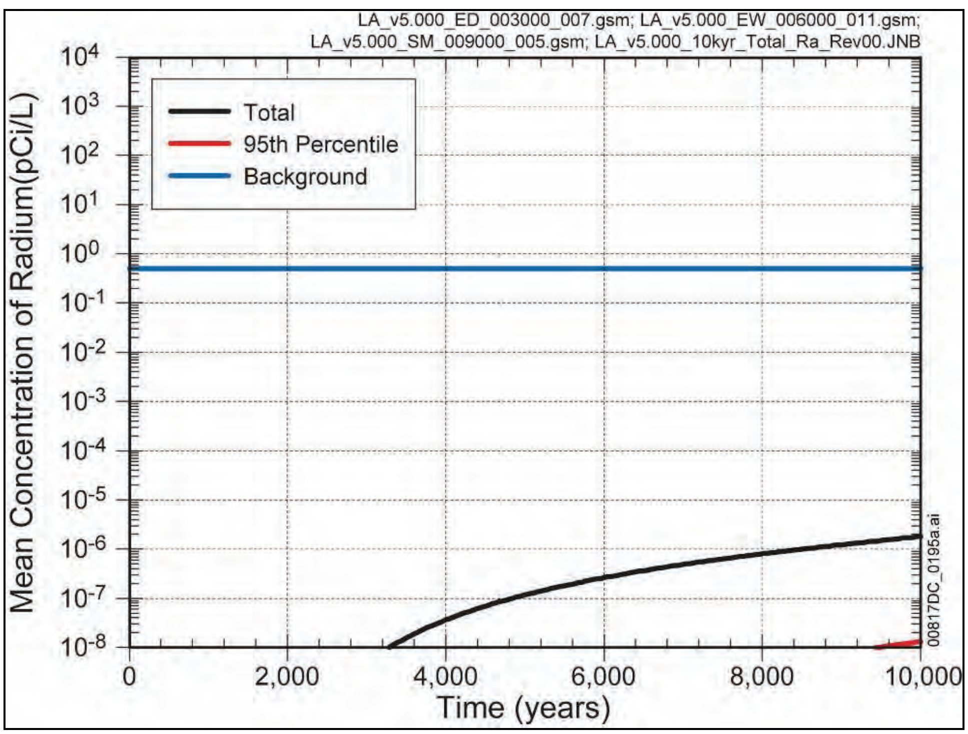

Figure F-18. Combined radium-226 and -228 activity concentrations, excluding natural background, for likely features, events, and processes using nominal, early failure, and seismic ground motion damage processes.

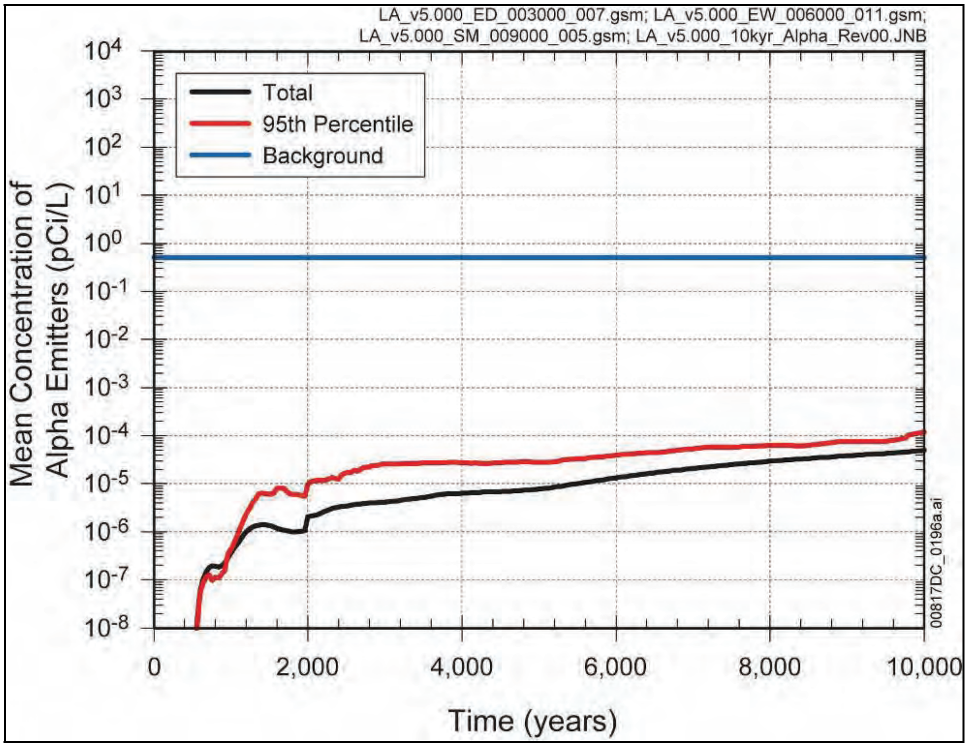

Figure F-19. Combined activity concentrations of all alpha emitters (including radium-226 but without radon and uranium isotopes), excluding natural background, for likely features, events, and processes using nominal, early failure, and seismic ground motion damage processes.

volume of groundwater for the Proposed Action inventory. The projected mean concentration for total radium in 10,000 years after closure is less than 2×10^{-6} picocurie per liter. The projected mean concentration of gross alpha activity during that period is less than 5×10^{-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 to background concentrations (about 0.5 picocurie per liter) up to 10,000 years.

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-SEIS 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 0.2 millirem. 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 0.04 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.

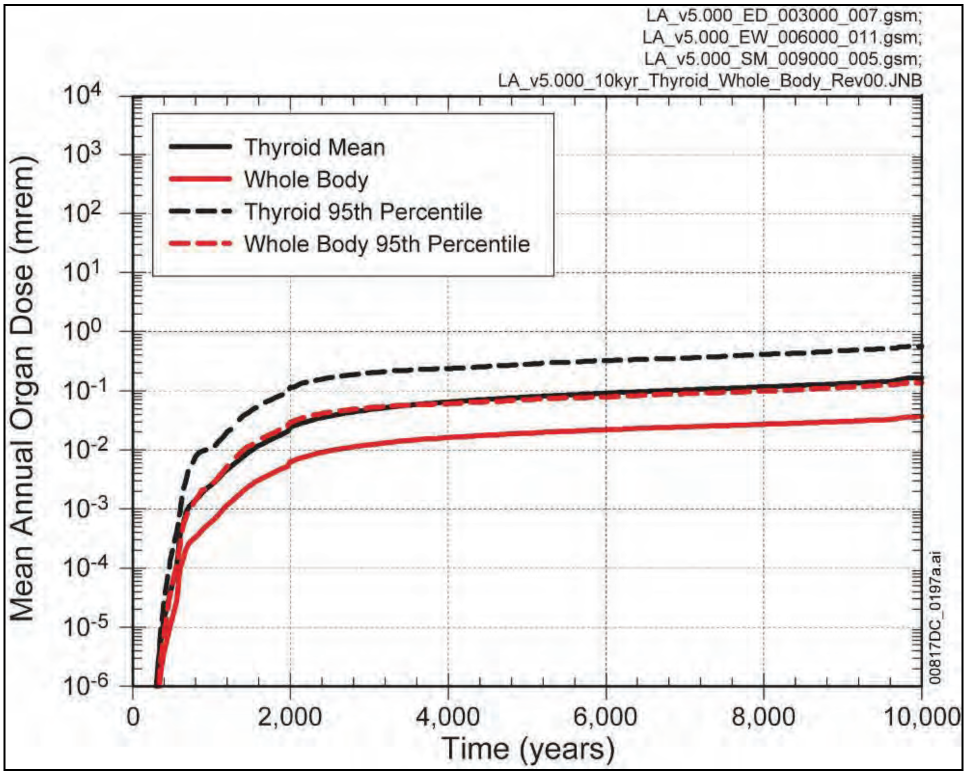

Figure F-20. Mean annual drinking water dose from combined beta and photon emitters for likely features, events, and processes using the nominal, early failure, and seismic ground motion damage processes.

Table F-4. Comparison of estimated postclosure impacts at the RMEI location to groundwater protection standards during 10,000 years following 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	1.8×10^{-6}	1.3×10^{-8}
Gross alpha activity (including radium-226 but excluding radon and uranium) (picocuries per liter)	15	4.9×10^{-5}	1.2×10^{-4}
Combined beta-and photon-emitting radionuclides (millirem per year to the whole body or any organ), based on drinking 2 liters of water per day from the representative volume	4	0.2	0.6

Source: DIRS 182846-SNL 2007, all.
Note: Conversion factors are on the inside back cover of this Repository SEIS.
RMEI = Reasonably maximally exposed individual.

F.5 Waterborne Chemically Toxic Material Impacts

DOE did not use the TSPA-SEIS 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 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 solubility.

Since the Yucca Mountain FEIS was completed, there has been additional research conducted into 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 is rapidly precipitated 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 will 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/liter for the pH 6 – 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°C on the order of 5×10^{-6} milligram/liter (DIRS 181408-Ziemniak et al. 1998, all). Another study with solutions ranging from pH 6-12, found the solubility to be 5×10^{-3} milligram/liter (DIRS 182718-Rai and Rao 2005, Figure 4). All of these values fall well below the Maximum Concentration Limit Goal of 0.1 milligram/liter set by EPA (40CFR 141.51). As water leaves the repository and is captured in the representative volume, it will have concentrations much less than the source values at the repository due to the dilution in 3,000 acre-feet per year representative volume. Thus chromium can be expected to have a concentration in the representative volume of much less than the Maximum Concentration Limit Goal. Therefore, chromium is 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 they 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 impacts are estimated to be 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 37.8°F (100°C) 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 316 stainless steel is for fresh water at 122° to 212°F (50° to 100°C) 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 above, 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 above, 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 [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

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)	Length[b] (millimeters)	Surface area (square millimeters)	Total surface area (square meters)
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 DOE SNF Long	1,406	2,126	5,304	35,425,554	49,808
2 MCO/2DHLW	149	1,831	5,279	30,366,160	4,525
5 DHLW Long/1 DOE SNF 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

Note: Conversion factors are on the inside back cover of this Repository SEIS.

a. Number of waste packages from DIRS 176937-DOE 2006, Table 2-11.

b. Waste package data from DIRS 179710-BSC 2007, all; DIRS 180192-BSC 2007, all; DIRS 179870-BSC 2007, all; DIRS 175303-BSC 2007, all; DIRS 180180-BSC 2007, all; DIRS 180187-BSC 2007, all; DIRS 182714-Morton 2007, all.

BWR = Boiling-water reactor.
DHLW = DOE high-level radioactive waste.
DSNF = DOE spent nuclear fuel.
MCO = Multicanister overpack.
PWR = Pressurized-water reactor.

Table F-6. Total exposed surface area of the Alloy-22 rails for all drip shields under the Proposed Action inventory.

Drip shield component	Number of pieces	Total waste package emplacement length[a,b] (meters)	Width[c] (millimeters)	Thickness[c] (millimeters)	Total surface area for repository[f] (square meters)
Rail	2	60,999	115	10	16,470

Note: Conversion factors are on the inside back cover of this Repository SEIS.

a. Sum of the waste package lengths plus a 0.1-meter (4-inch) spacing between packages.

b. Waste package data from DIRS 179710-BSC 2007, all; DIRS 180192-BSC 2007, all; DIRS 179870-BSC 2007, all; DIRS 175303-BSC 2007, all; DIRS 180180-BSC 2007, all; DIRS 180187-BSC 2007, all.

c. Rail dimensions from DIRS 150558-CRWMS M&O 2000, all.

d. Surface area calculated for the wetted surfaces (top and sides) of the rail.

components fabricated from Alloy-22 (from Tables F-5 to F-7) would be 576,362 square meters (6.2 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 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 exposed surface area of the stainless-steel ground support components is 2,317, 902 square meters (approximately 25 million square feet) (DIRS 182709-Duan 2007, all). Note that the figures given in the reference 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 about 2.5 million square meters (27.5 million square feet).

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	Length (millimeters)	Width (millimeters)	Number of sides	Total surface area per pallet (square meters)	Number of pallets	Total surface area repository (square meters)
Plate 1	2	1,845	552.4	1	2.038[b]		
Plate 2	2	922.5	614	2	2.266[c]		
Plate 3	2				2.219[d]		
Plate 4	4	552	462	2	2.040[e]		
Plate 5	4	552	80	2	0.353[f]		
Plate 6	4	1,266.7	603.2	2	6.113[g]		
Plate 7	4	152.4	79.9	2	0.049[h]		
Plate 8	4	152.4	552.4	1	0.337[i]		
Totals					15.415	11,177	172,293

Note: Conversion factors are on the inside back cover of this Repository SEIS.

a. Emplacement pallet details from DIRS 150558-CRWMS M&O 2000, sketches SK-0189 Rev 0 and SK-0144 Rev 1.
b. Calculated for one wetted rectangular side.
c. Calculated for both wetted rectangular sides.
d. Surface area equal to that of Plate 2 less area covered by 5.1-centimeter (2.0-inch) tube cross-sections.
e. Calculated assuming rectangular area covered by tubes is not wetted; while the inside and outside are covered by tubes, the width dimension is correct for each side.
f. Calculated assuming rectangular wetted area.
g. Calculated assuming wetted area includes exposed edge thicknesses that are added to the length and width.
h. Calculated based on triangular area.
i. Calculated assuming one wetted side only (because it is covered by the tube).

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)	Width[a] (millimeters)	Number of sides[a]	Total surface area per average waste package[b] (square meters)	Number of waste packages[c,d]	Total surface area repository (square meters)
Long pallets	4	4,147	609.6	2	20.224[e]	10,030	202,846
Short pallets	4	2,500	609.6	2	12.192[f]	1,147	13,984
Totals						11,177	216,830

Note: Conversion factors are on the inside back cover of this Repository SEIS.

a. Emplacement pallet details from DIRS 150558-CRWMS M&O 2000, sketches SK-0189 Rev 0 and SK-0144 Rev 1.
b. Calculated for area of all wetted rectangular sides.
c. Waste package data from DIRS 180187-BSC 2007, all.
d. Only waste packages of type 5 DHLW Short/1 DSNF Short use the short pallets.

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-SEIS 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-SEIS model is (DIRS 181031–SNL 2007, p. 2-1):

$$\ln(R_T) = \ln(R_0) + C_1\left(\frac{1}{T_0} - \frac{1}{T}\right) \qquad \text{(Equation F-4)}$$

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 or minus 2 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:

$$R_0 = 8.134\ \Gamma\left(1 + \frac{1}{1.476}\right) = 8.134\ \Gamma(1.677) \qquad \text{(Equation F-6)}$$

$\Gamma(1.677) = 0.905$ so that $R_0 = 7.36$ nanometers per year.

Let $T = 373.15$ kelvin (100°C); then, substituting into Equation F-4:

$$\ln(R_T) = \ln(7.36) + 4905\left(\frac{1}{333.15} - \frac{1}{373.15}\right) = 3.5756 \qquad \text{(Equation F-7)}$$

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 TSPA-SEIS 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°C to 100°C (122°F 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 (grams per year) 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)	General corrosion rate (meters per year)	Alloy release volume (cubic meters per year)	Alloy density (grams per cubic meter)	Bounding mass dissolution rate (gram per year)			
					Alloy	Molybdenum	Nickel	Vanadium
Alloy-22	576,362	3.57×10^{-8}	0.014	8,690,000	182,712	26,493	104,512	640
316NG	2,533,932	2.48×10^{-7}	0.628	7,980,000	5,01,475	125,368	601,770	0
Totals						151,861	706,282	640

Note: Conversion factors are on the inside back cover of this Repository SEIS.

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 prescribed in EPA regulation 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	0.04
Nickel	0.19
Vanadium	0.0001

In order to put these concentrations in perspective, a comparison of the intake from the maximum bounding concentrations in Table F-10 to the oral reference dose for each of these materials is presented.

Table F-11 lists the intakes by chemical under the assumption of water consumption of 2 liters (0.53 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 to Oral Reference Doses (milligrams per kilogram of body mass per day).

Material	Oral reference dose	Intake[a]
Molybdenum	0.005[b]	0.001
Nickel	0.02[c]	0.005
Vanadium	0.007[d]	0.000003

a. Assumes a daily intake of 2 liters (0.53 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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Appendix G

Transportation

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G. TRANSPORTATION

G.1 Introduction

This appendix to the *Draft 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-S1D) (Repository SEIS) summarizes the methods and data the U.S. Department of Energy (DOE or the Department) used to estimate the potential transportation impacts to workers and the public from shipments of spent nuclear fuel and high-level radioactive waste to the proposed repository. This appendix summarizes, incorporates by reference, and updates the analyses in Appendix J 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. J-1 to J-199) (Yucca Mountain FEIS).

Section G.1 discusses the methods and data used to estimate impacts at generator sites from loading activities. Section G.2 presents the representative transportation routes DOE would use to ship spent nuclear fuel and high-level radioactive waste from those sites to the proposed repository, Section G.3 lists the numbers of shipments from each site, and Section G.4 describes the radionuclide inventories the analysis used for estimation of impacts. Section G.5 presents the analysis and results for incident-free transportation, and Sections G.6 and G.7 describe transportation accident risks and the analysis of severe transportation accidents, respectively. Section G.8 disscusses sabotage events in relation to transportation. Section G.9 discusses general topics DOE examined for this analysis. Section G.10 presents the analysis and detailed results for transportation in Nevada, and Section G.11 provides those for transportation impacts due to transport of materials and personnel during construction and operation of the repository. Section G.12 contains figures of the representative transporation routes for each state through which shipments would pass, and lists the impacts of those shipments in those states.

G.2 Impacts at Generator Sites

This section describes the methods and data used to estimate the impacts from loading activities at generator sites. For rail shipments of commercial spent nuclear fuel from the generator sites, loading operations would include placement of the spent nuclear fuel into a transportation, aging, and disposal (TAD) canister, placement of the TAD canister into a rail transportation cask, and placement of the transportation cask on a railcar or heavy-haul truck. For truck shipments of commercial spent nuclear fuel, uncanistered spent nuclear fuel would be placed in a truck transportation cask and the truck cask would be placed on a truck trailer.

DOE would load its spent nuclear fuel into disposable canisters at three DOE sites and high-level radioactive waste into disposable canisters at four DOE sites. Loading operations would consist of placement of the canisters into a rail transportation cask and placement of the transportation cask on a railcar. A small amount of uncanistered spent nuclear fuel would be loaded into truck casks at the DOE sites.

G.2.1 IMPACTS OF SHIPPING CANISTERS AND CAMPAIGN KITS TO GENERATOR SITES

DOE would operate the proposed repository using a primarily canistered approach in which most commercial spent nuclear fuel would be packaged at the generator sites into TAD canisters. This would require shipment of TAD canisters to the commercial generator sites. These shipments of empty canisters would be by truck. Before the loading of a truck or rail transportation cask, equipment used in the handling and loading of the cask, known as a campaign kit, would be shipped to the generator sites. These shipments would also be by truck.

The shipments of canisters would not be radioactive material shipments, so there would be no radiation dose to the public or to workers from them. The campaign kits could become contaminated during use, but would be decontaminated before shipping. Therefore, the radiation dose and radiological risks associated with the shipping of campaign kits would be negligible. The impacts of transporting canisters and campaign kits would be from fatalities from exposure to vehicle emissions and traffic fatalities. Injuries were not estimated because they are not readily combined with radiological impacts, which were quantified in terms of latent cancer fatalities. DOE estimated these impacts based on a 6,000-kilometer (3,700-mile) round-trip shipping distance for the canisters and the campaign kits and a population density of 220 people per square kilometer (570 people per square mile). The Department used data from the 2000 Census extrapolated to 2067 to estimate the population density along the representative truck routes (see Section G.2).

Table G-1 summarizes the data DOE used to estimate the impacts of these shipments.

Table G-1. Data used to estimate impacts from shipping canisters and campaign kits.

Quantity	Value	Reference
Number of canisters shipped	6,499[a]	DIRS 181377-BSC 2007, Section 7
Number of campaign kits shipped	4,942	DIRS 181377-BSC 2007, Section 7
Vehicle emission fatality rate	1.5×10^{-11} fatalities/km per person/km^2	DIRS 157144-Jason Technologies 2001, p. 98
Traffic fatality rate	1.71×10^{-8} fatalities/km	DIRS 182082-FMCSA 2007, Table 13

Notes: Vehicle emission fatality rate and traffic fatality rate are for trucks. Conversion factors are on the inside back cover of this Repository SEIS.

a. About an additional 1,000 empty TAD canisters would be shipped directly to the repository to package commercial spent nuclear fuel that could not be shipped from the generator sites using rail casks.

km = kilometer.

G.2.2 RADIOLOGICAL IMPACTS TO WORKERS FROM LOADING

At commercial generator sites, impacts to involved workers would result from loading spent nuclear fuel into canisters, loading canisters into rail transportation casks, and, at some sites, loading spent nuclear fuel into truck casks. For DOE spent nuclear fuel and high-level radioactive waste, impacts would result from loading canisters into rail transportation casks and a small amount of uncanistered spent nuclear fuel into truck casks. Noninvolved workers would not be in proximity to the canisters or casks and would not be exposed during loading. Therefore, DOE did not estimate radiological impacts for these noninvolved workers. Table G-2 summarizes the data DOE used to estimate the radiological impacts from these activities.

A TAD canister is similar to a dry storage canister in appearance, capacity, and the operational procedures that would be in use for loading. Therefore, for the loading of spent nuclear fuel into TAD canisters at commercial generator sites, DOE based radiation doses on utility data compiled by the U.S. Nuclear Regulatory Commission (NRC) for loading 87 dry storage canisters at four commercial sites (DIRS 181757-NRC 2002, Attachment 3; DIRS 181758-Spitzberg 2004, Attachment 2; DIRS 181759-Spitzberg

Table G-2. Data used to estimate radiation doses to workers for loading.

Operation	Radiation dose	Number of canisters or casks for operation	Reference
Rail cask			
Load commercial spent nuclear fuel into canister	0.400 person-rem per canister	6,499 canisters[a]	Average of utility data in DIRS 181757-NRC 2002, Attachment 3; DIRS 181758-Spitzberg 2004, Attachment 2; DIRS 181759-Spitzberg 2005, Attachment 2; DIRS 181760-Spitzberg 2005, Attachment 2
Transfer canister from storage, load into rail cask, load rail cask onto railcar	0.663 person-rem per cask	9,495 casks[b]	Steps 12 and 13 in DIRS 104794-CRWMS M&O 1994, p. A-28
Truck cask			
Load uncanistered spent nuclear fuel into truck cask, load truck cask onto truck trailer	0.432 person-rem per cask	2,650 casks[c]	Steps 1, 2, 3a, 4a, and 5a in DIRS 104794-CRWMS M&O 1994, pp. A-9 to A-11

a. Includes only TAD canisters (DIRS 181377-BSC 2007, Section 7).
b. Includes commercial spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste (DIRS 181377-BSC 2007, Section 7).
c. DIRS 181377-BSC 2007, Section 7.
DOE = U.S. Department of Energy.
TAD = Transportation, aging, and disposal (canister).

2005, Attachment 2; DIRS 181760-Spitzberg 2005, Attachment 2). Using the utility data, DOE estimated the average radiation dose for loading spent nuclear fuel into canisters to be 0.400 person-rem per canister. For comparison, the estimated radiation dose for these same activities would be 1.992 person-rem (DIRS 104794-CRWMS M&O 1994, p. A-24).

DOE used data from *Health and Safety Impacts Analysis for the Multi-Purpose Canister System and Alternatives* (DIRS 104794-CRWMS M&O 1994, pp. A-9 and A-24) to estimate radiation doses for the loading of (1) canisters containing spent nuclear fuel into rail casks and uncanistered spent nuclear fuel into truck casks, (2) canisters containing high-level radioactive waste and canisters containing DOE spent nuclear fuel into rail casks, and (3) rail casks onto railcars and truck casks onto truck trailers. For loading uncanistered spent nuclear fuel into truck casks and loading the truck casks onto trailers, the estimated radiation dose would be 0.432 person-rem per cask (DIRS 104794-CRWMS M&O 1994, p. A-9). For loading canisters into rail casks and loading the rail casks onto railcars, the estimated radiation dose would be 0.663 person-rem per cask (DIRS 104794-CRWMS M&O 1994, p. A-24).

G.2.3 INDUSTRIAL SAFETY IMPACTS TO WORKERS FROM LOADING

DOE based the analysis of industrial safety impacts on an average loading duration of 2.3 days per rail cask for pressurized-water-reactor spent nuclear fuel and 2.5 days per rail cask for boiling-water-reactor spent nuclear fuel (DIRS 155970-DOE 2002, p. J-34). For truck casks, DOE based the analysis on an average loading duration of 1.3 days per cask for pressurized-water-reactor spent nuclear fuel and 1.4 days per cask for boiling-water-reactor spent nuclear fuel (DIRS 155970-DOE 2002, p. J-34). The Department based loading durations for DOE spent nuclear fuel and high-level radioactive waste on the loading durations for pressurized-water-reactor spent nuclear fuel. It based the industrial safety impacts on a crew size of 13 (DIRS 155970-DOE 2002, p. J-34) dedicated solely to performing cask-handling work and an 8-hour working day. Based on these data, 1,347 worker-years would be spent during loading activities for involved workers. Using the assumption that the noninvolved workforce would be 25 percent of the involved workforce, DOE determined that uninvolved workers would spend 337 worker-years during loading activities for uninvolved workers (DIRS 155970-DOE 2002, p. 6-38).

DOE based incidence and fatality rates for involved workers on Bureau of Labor Statistics data for 2005 (DIRS 179131-BLS 2006, all; DIRS 179129-BLS 2007, all). Bureau of Labor Statistics data is organized into industries. DOE used data for workers in the transportation and warehousing industries to estimate impacts because they closely represent the hazards associated with loading casks. Data from DOE sources was not used because most of the generator sites were associated with private industry rather than DOE. For noninvolved workers, the Department based the rates on the professional and business services industries.

For vehicle emission fatalities, DOE based the analysis of industrial safety impacts on a vehicle emission fatality rate of 9.4×10^{-12} fatalities per kilometer per persons per square kilometer (DIRS 157144-Jason Technologies 2001, p. 99) and on a population density of 6 persons per square kilometer (16 persons per square mile), which is representative of a rural area (DIRS 101892-NRC 1977, p. E-2). For traffic fatalities, DOE based the analysis of industrial safety impacts on a fatality rate of 1.0×10^{-8} fatalities per kilometer (DIRS 182082-FMCSA 2007, Table 2) over the period from 2001 through 2005. DOE also based the analysis on workers driving 37 kilometers (23 miles) round trip for 251 days per year. Table G-3 summarizes the data DOE used to estimate the industrial safety impacts from loading activities.

Table G-3. Data used to estimate industrial safety impacts to workers for loading.

Quantity	Value	Reference
Involved workers		
Worker-years	1,347[a]	Calculated
Total recordable cases rate	0.082 per worker-year	DIRS 179131-BLS 2006, all; for warehousing and storage industries
Lost workday cases rate	0.054 per worker-year	DIRS 179131-BLS 2006, all; for warehousing and storage industries
Fatality rate	1.76×10^{-4} per worker-year	DIRS 179129-BLS 2007, all; for transportation and warehousing industries
Noninvolved workers		
Worker-years	337	Calculated
Total recordable cases rate	0.024 per worker-year	DIRS 179131-BLS 2006, all; for professional and business services, management of companies and enterprises
Lost workday cases rate	0.012 per worker-year	DIRS 179131-BLS 2006, all; for professional and business services, management of companies and enterprises
Fatality rate	3.5×10^{-5} per worker-year	DIRS 179129-BLS 2007, all; for professional and business services
Vehicle emission fatality rate	9.4×10^{-12} fatalities/km per person/km^2	DIRS 157144-Jason Technologies 2001, p. 99
Traffic fatality rate	1.0×10^{-8} fatalities per km	DIRS 182082-FMCSA 2007, Table 2

Notes: Vehicle emission fatality rate and traffic fatality rate are for automobiles. Conversion factors are on the inside back cover of this Repository SEIS.

a. Based on loading 6,736 pressurized-water-reactor spent nuclear fuel, DOE spent nuclear fuel, and high-level radioactive waste rail casks, 1,940 pressurized-water-reactor truck casks, 2,759 boiling-water-reactor rail casks, and 710 boiling-water reactor truck casks.

km = kilometer.

G.3 Transportation Routes

At this time, before receipt of a construction authorization for the proposed repository and years before a possible first shipment, the specific rail and highway routes shipments of spent nuclear fuel and high-level radioactive waste to Yucca Mountain will use have not been identified. Consequently, the analysis of impacts presented in this supplemental environmental impact statement is based on routes that could be used and that DOE believes are representative of those that will be used. Therefore, the highway and rail routes that DOE used for analysis in this Repository SEIS are called representative routes.

DOE used the TRAGIS computer program (DIRS 181276-Johnson and Michelhaugh 2003, all) to identify the representative rail and truck routes used in the analysis. TRAGIS is a web-based geographic information system transportation routing computer code. The TRAGIS rail network is developed from a 1-to-100,000-scale rail network derived from the United States Geological Survey digital line graphs. This network currently represents more than 240,000 kilometers (150,000 miles) of rail lines in the continental United States and has over 28,000 segments (links) and over 4,000 nodes. All rail lines with the exception of industrial spurs are included. The rail network includes nodes for nuclear reactor sites, DOE sites, and military bases that have rail access. The rail network has been extensively modified and is revised on a regular schedule to reflect rail line abandonment, company mergers, short line spin-offs, and new rail construction.

To calculate rail routes, the TRAGIS computer program uses rules that are designed to simulate routing practices that have been historically used by railroad companies in moving regular freight and dedicated trains in the United States. The basic rule used to calculate rail routes causes the program to attempt to identify the shortest route from an origin to a destination. Another rule used in the program biases the lengths of route segments that have the highest density of rail traffic to make these segments appear, for purposes of calculation, to be shorter. The effect of the bias is to prioritize selection of routes that use railroad main lines, which have the highest traffic density. As a general rule routing along the high traffic lines replicates railroad operational practices. A third rule constrains the program to select routes used by an individual railroad company to lines the company owns or has permission to operate over. This rule ensures that the number of interchanges between railroads that the TRAGIS computer program calculates for a route is correct. The number of interchanges between railroads is a significant consideration when determining a realistic and representative route.

Another rule used in the TRAGIS computer program to calculate a rail route determines the sequence of different railroad companies whose rail lines would be linked to form the route. Because a delay and additional operations are involved in transferring a shipment (interchanging) from one railroad to another, in order to provide efficient service, railroads typically route shipments to minimize the number of interchanges that occur. Reducing the number of interchanges also tends to reduce the time a shipment is in transit. This practice is simulated in the TRAGIS computer program by imposing a penalty for each interchange that is identified for a route. The interchange penalties cause the TRAGIS computer program to increase the calculated length of routes when more than one railroad company's lines are linked. As a consequence, the algorithm used in the TRAGIS computer program to identify routes that have the least apparent length gives advantage to routes that also have the fewest interchanges between railroads and the fewest involved railroad companies.

Last, a rule in the TRAGIS computer program is designed to simulate the commercial behavior of railroad companies to maximize their portion of revenues from shipments. The effect of this behavior is that routing is often affected by originating railroads, who control the selection of routes on their lines to realize the as much of a shipment's revenue as possible. The result is that originating railroads transport shipments as far as possible (in the direction of the destination) on their systems before interchanging the shipments with other railroads. This behavior is simulated in the TRAGIS computer program by imposing a bias on the length of the originating railroad's lines to give the railroad an advantage when calculating a route. In evaluating the length of the route, the model treats 1 mile of travel on the originating railroad as being "less" than 1 mile on other railroads.

The TRAGIS highway network is developed from a 1-to-100,000-scale road network derived from United States Geological Survey digital line graphs and Bureau of the Census TIGER data. The network represents slightly more than 378,000 kilometers (235,000 miles) of roadways and includes all Interstate highways, most U.S. highways except those that closely parallel Interstate highways, major state highways, and other local roads that connect to various specific sites of interest. The network currently includes over 22,000 highway segments (links) and over 16,000 intersections (nodes). The network includes nuclear reactor sites, DOE sites, and commercial and military airports.

TRAGIS provides a variety of routing rules that can be used to calculate highway routes. The default rules yield highway routes that commercial motor carriers of freight would be expected to use. In addition, TRAGIS can be used to: (1) determine routes that meet the U.S. Department of Transportation regulations for shipments of highway route-controlled quantities of radioactive material; (2) identify the shortest route between an origin and destination; or (3) identify the route that could be expected to result in the least total time in transit.

The population data in TRAGIS are derived from the LandScan USA 15-arc second (approximately 360-by-460-meter) grid cell population database. This national database represents the nighttime population distribution and is developed from a combination of data sources including 2000 Bureau of the Census block group population, roads from the Bureau of the Census TIGER data, slope from the National Imagery and Mapping Agency's Digital Terrain Elevation Data, and land cover from the United States Geological Survey National Land Cover Database. The data are modeled to best approximate the actual location of the resident population. Because of the proximity of the repository to Las Vegas, the resident population in Las Vegas was modified to include casino guests and casino workers, based on data from the Nevada Agency for Nuclear Projects (DIRS 158452-Nevada Agency for Nuclear Projects 2002, Table 3.8.12).

The routes used in the analysis that are also representative of routes that could be used for shipments to the repository are illustrated in Figures G-1 and G-2. DOE determined rail routes in two steps. In the first step, representative routes were determined from the generator sites to either Caliente or Hazen, Nevada. In the second step, the rail alternative segments that comprise the rail alignment with the highest population in the Caliente or Mina rail alignment were used to determine the representative route from Caliente or Hazen to the repository. Tables G-4 and G-5 list the distances from the generator sites to Caliente and Hazen. Table G-6 lists the distances from Caliente and Hazen to the repository.

Some generator sites do not have direct rail access. For these sites, heavy-haul trucks would have to be used to move the rail cask containing spent nuclear fuel to a nearby railhead. Barges could also be used; Section G.10.10 discusses barge shipments. Table G-7 lists the distances from these generator sites to the nearby railheads.

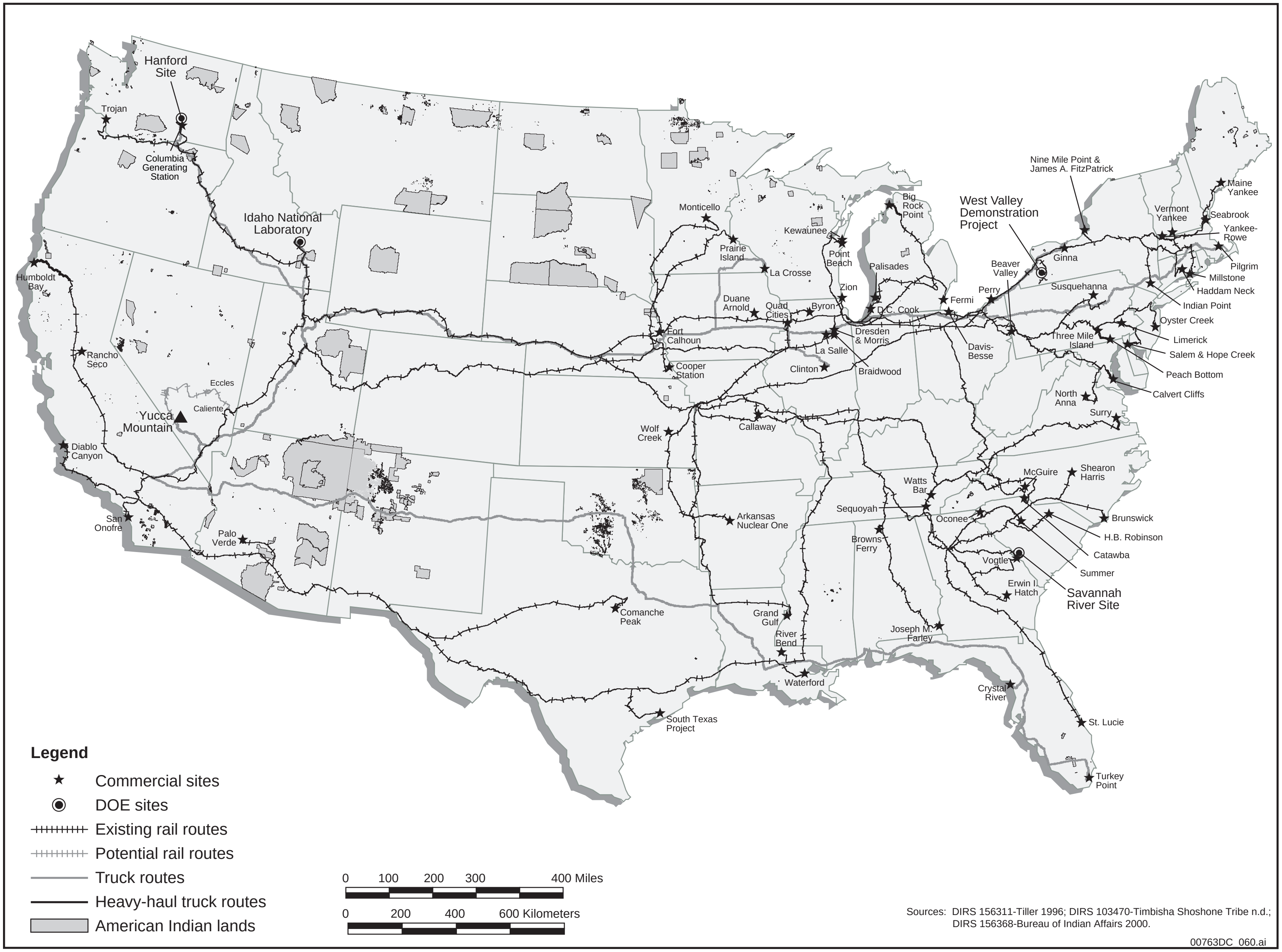

Figure G-1. Representative rail and truck transportation routes if DOE selected the Caliente rail corridor in Nevada.

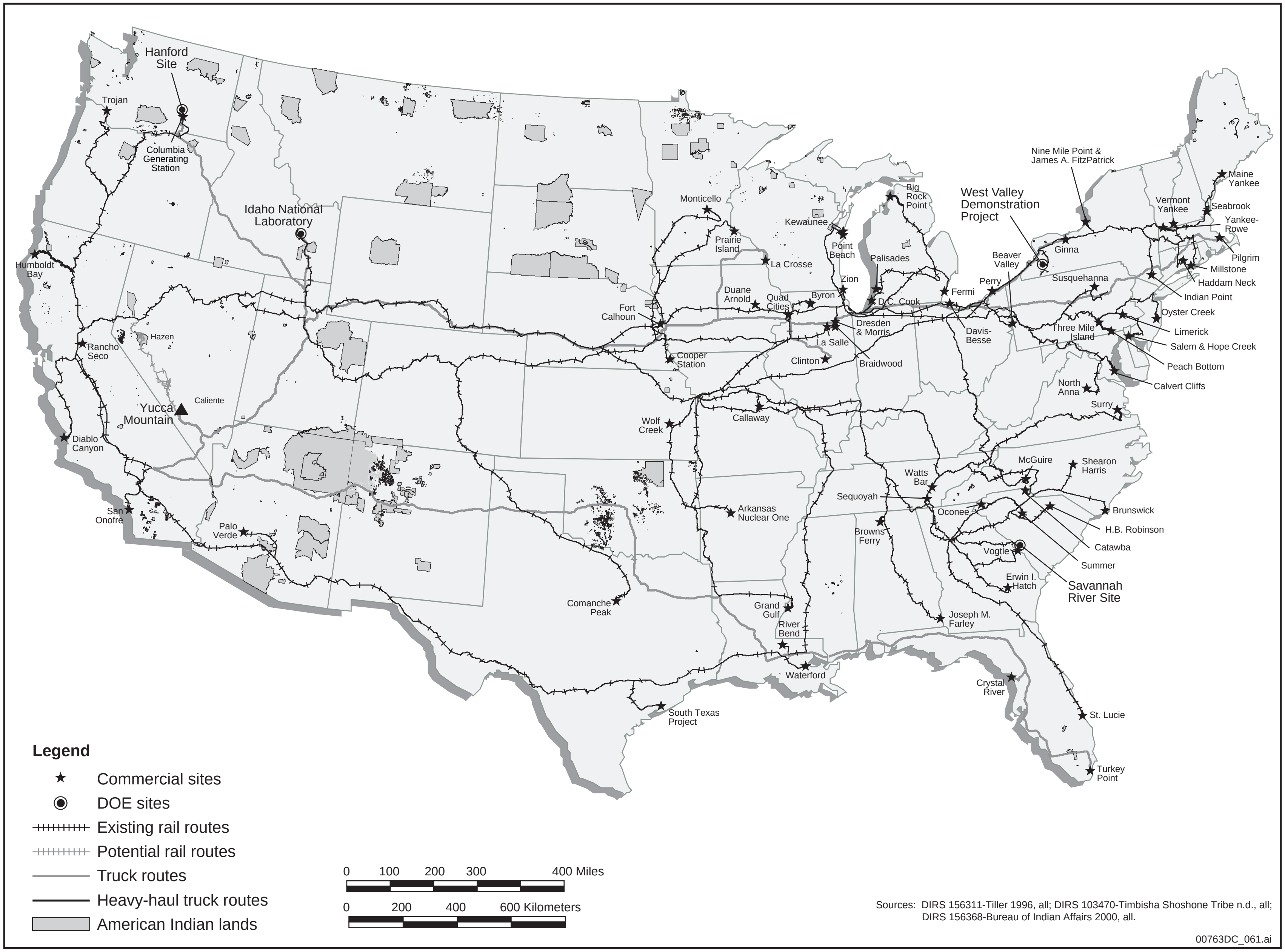

Figure G-2. Representative rail and truck transportation routes if DOE selected the Mina rail corridor.

Table G-4. Distances for representative rail routes from generator sites to Caliente, Nevada.

Origin	Origin state	Mode	Rural kilometers	Suburban kilometers	Urban kilometers
Browns Ferry	AL	Rail	2,947.0	490.2	97.9
Farley	AL	Rail	3,331.8	643.9	109.6
Arkansas	AR	Rail	2,668.0	305.9	51.0
Palo Verde	AZ	Rail	1,216.5	197.8	63.7
Diablo Canyon	CA	Rail	781.9	166.2	131.0
Humboldt Bay	CA	Rail	1,020.2	289.4	110.1
Rancho Seco	CA	Rail	853.7	213.0	82.4
San Onofre	CA	Rail	584.1	107.1	77.1
Haddam Neck	CT	Rail	3,369.2	905.6	216.2
Millstone	CT	Rail	3,417.4	942.7	218.3
St. Lucie	FL	Rail	3,642.7	940.1	166.0
Hatch	GA	Rail	3,459.9	724.0	105.4
Vogtle	GA	Rail	3,504.7	723.5	104.5
Arnold	IA	Rail	2,240.8	288.1	46.6
Idaho National Laboratory	ID	Rail	796.1	93.4	25.7
Braidwood	IL	Rail	2,657.4	402.6	96.8
Byron	IL	Rail	2,428.5	321.3	47.4
Dresden	IL	Rail	2,479.0	367.5	62.4
LaSalle	IL	Rail	2,525.7	275.8	40.4
Morris	IL	Rail	2,478.9	367.4	62.4
Quad Cities	IL	Rail	2,456.3	283.7	42.0
Zion	IL	Rail	2,467.3	387.7	86.3
Wolf Creek	KS	Rail	2,242.7	218.5	46.9
River Bend	LA	Rail	3,288.1	584.7	106.6
Waterford	LA	Rail	3,060.6	505.1	122.6
Yankee Rowe	MA	Rail	3,284.5	797.3	190.8
Calvert Cliffs	MD	Rail	3,267.0	709.0	223.4
Maine Yankee	ME	Rail	3,484.0	991.0	235.6
Big Rock Point	MI	Rail	2,913.0	666.9	154.7
Fermi	MI	Rail	2,742.3	542.6	158.5
Palisades	MI	Rail	2,543.4	434.2	119.6
Monticello	MN	Rail	2,477.4	331.4	51.6
Prairie Island	MN	Rail	2,373.0	325.0	48.0
Callaway	MO	Rail	2,346.5	243.6	52.2
Grand Gulf	MS	Rail	3,052.8	420.7	60.1
Brunswick	NC	Rail	3,529.1	877.4	142.6
Harris	NC	Rail	3,450.6	867.4	142.3

Table G-4. Distances for representative rail routes from generator sites to Caliente, Nevada (continued).

Origin	Origin state	Mode	Rural kilometers	Suburban kilometers	Urban kilometers
McGuire	NC	Rail	3,450.8	730.1	155.3
Cooper	NE	Rail	2,009.6	218.4	47.6
Fort Calhoun	NE	Rail	1,923.9	179.6	38.4
Seabrook	NH	Rail	3,420.0	930.0	221.5
Hope Creek	NJ	Rail	3,131.0	911.4	315.0
Oyster Creek	NJ	Rail	3,180.8	922.9	326.0
Salem	NJ	Rail	3,131.0	911.4	315.0
FitzPatrick	NY	Rail	3,138.8	698.0	192.5
Indian Point	NY	Rail	3,360.0	792.9	204.9
Nine Mile Point	NY	Rail	3,138.5	697.4	192.5
West Valley	NY	Rail	3,028.7	628.8	167.4
Davis-Besse	OH	Rail	2,695.9	485.2	143.6
Perry	OH	Rail	3,099.3	412.9	115.2
Trojan	OR	Rail	1,763.2	246.2	72.6
Beaver Valley	PA	Rail	3,170.2	463.7	110.6
Limerick	PA	Rail	3,430.7	681.5	195.3
Peach Bottom	PA	Rail	3,416.4	639.0	171.5
Susquehanna	PA	Rail	3,155.2	799.5	244.3
Three Mile Island	PA	Rail	3,398.9	633.0	171.9
Catawba	SC	Rail	3,339.1	784.0	113.3
Oconee	SC	Rail	3,275.2	734.1	112.1
Robinson	SC	Rail	3,334.6	839.8	147.6
Savannah River Site	SC	Rail	3,308.8	726.8	149.8
Summer	SC	Rail	3,385.4	839.9	119.8
Sequoyah	TN	Rail	3,086.3	526.1	85.3
Watts Bar	TN	Rail	3,057.4	502.6	84.7
Comanche Peak	TX	Rail	2,456.5	379.8	87.0
South Texas	TX	Rail	2,769.1	336.3	93.2
North Anna	VA	Rail	3,379.6	732.3	227.4
Surry	VA	Rail	3,552.7	812.2	111.0
Vermont Yankee	VT	Rail	3,390.0	881.3	201.3
Columbia	WA	Rail	1,540.6	176.9	40.0
Hanford Site	WA	Rail	1,575.1	177.0	40.0
Kewaunee	WI	Rail	2,619.9	490.8	125.8
Point Beach	WI	Rail	2,619.9	490.8	125.8

Notes: Rural areas have a population density less then 139 people per square kilometer. Suburban areas have a population density between 139 and 3,326 people per square kilometer. Urban areas have a population density greater than 3,326 people per square kilometer. Conversion factors are on the inside back cover of this Repository SEIS.

Table G-5. Distances for representative rail routes from generator sites to Hazen, Nevada.

Origin	Origin state	Mode	Rural kilometers	Suburban kilometers	Urban kilometers
Browns Ferry	AL	Rail	3,200.6	470.3	83.2
Farley	AL	Rail	3,585.5	624.0	94.9
Arkansas	AR	Rail	2,921.6	286.0	36.3
Palo Verde	AZ	Rail	1,250.5	459.6	172.3
Diablo Canyon	CA	Rail	512.5	233.7	103.5
Humboldt Bay	CA	Rail	359.8	140.5	32.7
Rancho Seco	CA	Rail	241.3	93.8	40.0
San Onofre	CA	Rail	774.1	306.1	161.5
Haddam Neck	CT	Rail	3,622.8	885.8	201.5
Millstone	CT	Rail	3,671.0	922.8	203.6
St. Lucie	FL	Rail	3,896.3	920.3	151.3
Hatch	GA	Rail	3,713.5	704.1	90.7
Vogtle	GA	Rail	3,758.3	703.6	89.8
Arnold	IA	Rail	2,494.4	268.3	31.9
Idaho National Laboratory	ID	Rail	1,049.1	69.6	10.3
Braidwood	IL	Rail	2,911.0	382.8	82.1
Byron	IL	Rail	2,682.1	301.4	32.7
Dresden	IL	Rail	2,732.6	347.6	47.7
LaSalle	IL	Rail	2,907.3	332.5	55.3
Morris	IL	Rail	2,732.5	347.5	47.7
Quad Cities	IL	Rail	2,837.9	340.4	56.9
Zion	IL	Rail	2,720.9	367.8	71.6
Wolf Creek	KS	Rail	2,496.3	198.6	32.2
River Bend	LA	Rail	3,541.7	564.8	91.9
Waterford	LA	Rail	3,094.7	766.9	231.2
Yankee Rowe	MA	Rail	3,538.1	777.5	176.1
Calvert Cliffs	MD	Rail	3,520.6	689.1	208.7
Maine Yankee	ME	Rail	3,737.6	971.1	220.9
Big Rock Point	MI	Rail	3,166.6	647.0	139.9
Fermi	MI	Rail	2,995.9	522.8	143.7
Palisades	MI	Rail	2,797.0	414.4	104.9
Monticello	MN	Rail	2,859.0	388.1	66.5
Prairie Island	MN	Rail	2,626.6	305.2	33.2
Callaway	MO	Rail	2,600.1	223.7	37.5
Grand Gulf	MS	Rail	3,306.5	400.8	45.4
Brunswick	NC	Rail	3,782.7	857.6	127.9
Harris	NC	Rail	3,704.2	847.6	127.6

Table G-5. Distances for representative rail routes from generator sites to Hazen, Nevada (continued).

Origin	Origin state	Mode	Rural kilometers	Suburban kilometers	Urban kilometers
McGuire	NC	Rail	3,704.4	710.2	140.6
Cooper	NE	Rail	2,263.3	198.6	32.9
Fort Calhoun	NE	Rail	2,177.5	159.8	23.7
Seabrook	NH	Rail	3,673.6	910.2	206.8
Hope Creek	NJ	Rail	3,384.6	891.5	300.3
Oyster Creek	NJ	Rail	3,434.4	903.0	311.3
Salem	NJ	Rail	,3384.6	891.5	300.3
FitzPatrick	NY	Rail	3,392.4	678.1	177.8
Indian Point	NY	Rail	3,613.6	773.0	190.2
Nine Mile Point	NY	Rail	3,392.1	677.5	177.8
West Valley	NY	Rail	3,282.3	608.9	152.7
Davis-Besse	OH	Rail	2,949.5	465.4	128.9
Perry	OH	Rail	3,352.9	393.1	100.5
Trojan	OR	Rail	1,013.2	335.4	90.9
Beaver Valley	PA	Rail	3,423.8	443.8	95.9
Limerick	PA	Rail	3,684.3	661.6	180.6
Peach Bottom	PA	Rail	3,670.0	619.2	156.8
Susquehanna	PA	Rail	3,408.9	779.6	229.6
Three Mile Island	PA	Rail	3,652.5	613.1	157.2
Catawba	SC	Rail	3,592.7	764.2	98.6
Oconee	SC	Rail	3,528.8	714.3	97.4
Robinson	SC	Rail	3,588.2	819.9	132.9
Savannah River Site	SC	Rail	3,562.4	707.0	135.1
Summer	SC	Rail	3,639.0	820.1	105.1
Sequoyah	TN	Rail	3,339.9	506.2	70.6
Watts Bar	TN	Rail	3,311.0	482.8	70.0
Comanche Peak	TX	Rail	2,731.9	340.4	65.5
South Texas	TX	Rail	2,803.2	598.0	201.9
North Anna	VA	Rail	3,633.2	712.5	212.7
Surry	VA	Rail	3,806.3	792.4	96.3
Vermont Yankee	VT	Rail	3,643.6	861.4	186.6
Columbia	WA	Rail	1,225.3	248.4	45.3
Hanford Site	WA	Rail	1,259.9	248.5	45.3
Kewaunee	WI	Rail	2,873.5	470.9	111.1
Point Beach	WI	Rail	2,873.5	470.9	111.1

Notes: Rural areas have a population density less then 139 people per square kilometer. Suburban areas have a population density between 139 and 3,326 people per square kilometer. Urban areas have a population density greater than 3,326 people per square kilometer. Conversion factors are on the inside back cover of this Repository SEIS.

Table G-6. Distances from Caliente and Hazen to the repository.

Origin	County	Mode	Rural kilometers	Suburban kilometers	Urban kilometers
Caliente					
	Lincoln	Rail	148.75	0.35	0
	Nye	Rail	358.64	0	0
	Esmeralda	Rail	31.08	0.12	0
Hazen					
	Churchill	Rail	18.61	0	0
	Lyon	Rail	89.09	0.88	0
	Mineral	Rail	154.81	0	0
	Esmeralda	Rail	132.76	0.11	0
	Nye	Rail	149.55	0	0

Notes: Rural areas have a population density less then 139 people per square kilometer. Suburban areas have a population density between 139 and 3,326 people per square kilometer. Urban areas have a population density greater than 3,326 people per square kilometer. Conversion factors are on the inside back cover of this Repository SEIS.

Table G-7. Distances for representative heavy-haul truck routes from generator sites to nearby railroads.

Origin	Origin state	Mode	Rural kilometers	Suburban kilometers	Urban kilometers
Browns Ferry	AL	Heavy haul[a]	19.4	8.4	0.4
Diablo Canyon	CA	Heavy haul[a]	22.3	7.0	2.6
Humboldt Bay	CA	Heavy haul	206.8	28.5	6.1
Haddam Neck	CT	Heavy haul[a]	10.2	9.6	1.0
St. Lucie	FL	Heavy haul[a]	13.0	7.5	0.6
Yankee Rowe	MA	Heavy haul	25.9	7.0	1.3
Calvert Cliffs	MD	Heavy haul[a]	25.4	31.5	0.3
Big Rock Point	MI	Heavy haul	60.5	12.0	0.8
Palisades	MI	Heavy haul[a]	15.9	13.9	0.1
Callaway	MO	Heavy haul	19.1	1.9	0.6
Grand Gulf	MS	Heavy haul[a]	32.6	2.2	0.0
Cooper	NE	Heavy haul[a]	18.0	1.8	0.2
Fort Calhoun	NE	Heavy haul	3.7	1.4	0.3
Hope Creek	NJ	Heavy haul[a]	29.3	6.5	0.2
Oyster Creek	NJ	Heavy haul[a]	6.0	17.4	5.1
Salem	NJ	Heavy haul[a]	29.0	6.1	0.2
Indian Point	NY	Heavy haul[a]	0.9	1.1	1.4
Peach Bottom	PA	Heavy haul	29.4	18.5	6.6
Oconee	SC	Heavy haul	8.2	3.3	0.0
Surry	VA	Heavy haul[a]	37.1	12.0	0.3
Kewaunee	WI	Heavy haul[a]	35.7	5.2	0.2
Point Beach	WI	Heavy haul[a]	30.8	5.0	0.2

Notes: Rural areas have a population density less then 139 people per square kilometer. Suburban areas have a population density between 139 and 3,326 people per square kilometer. Urban areas have a population density greater than 3,326 people per square kilometer. Conversion factors are on the inside back cover of this Repository SEIS.
a. Could also ship by barge.

Some generator sites do not have the ability to handle a rail cask at their facilities. Unless site capabilities are upgraded at these sites, truck casks would have to be used to ship the spent nuclear fuel. In addition, there would be a small number of commercial spent nuclear fuel truck shipments from the Hanford Site and the Idaho National Laboratory. For truck shipments, DOE determined the representative routes based on the U.S. Department of Transportation rules for Highway Route-Controlled Quantity shipments in 49 CFR 397.101. Figures G-1 and G-2 show the representative truck routes used in the analysis from these generator sites to the repository and Table G-8 lists the distances from these generator sites to the repository.

Table G-8. Distances for representative truck routes from generator sites to the repository.

Origin	Origin State	Mode	Rural kilometers	Suburban kilometers	Urban kilometers
Crystal River	FL	Truck	3,552.8	834.3	113.9
Turkey Point	FL	Truck	3,910.8	998.7	154.8
Idaho National Laboratory	ID	Truck	951.0	196.9	48.0
Clinton	IL	Truck	2,636.6	394.7	51.4
Pilgrim	MA	Truck	3,480.3	1086.8	120.8
Cook	MI	Truck	2,654.5	452.1	65.8
Ginna	NY	Truck	3,139.4	824.1	109.6
Hanford Site	WA	Truck	1,531.1	286.6	59.9
LaCrosse	WI	Truck	2,616.0	328.5	55.7

Notes: Rural areas have a population density less then 139 people per square kilometer. Suburban areas have a population density between 139 and 3,326 people per square kilometer. Urban areas have a population density greater than 3,326 people per square kilometer. Conversion factors are on the inside back cover of this Repository SEIS.

The population density data DOE used in this Repository SEIS from TRAGIS and for the Caliente and Mina rail alignments were for 800 meters (0.5 mile) on either side of the representative rail or truck route and were based on 2000 Census data. Because the analysis considered that the repository would operate for 50 years, DOE used Bureau of the Census population estimates for 2000 through 2030 to extrapolate population densities along the routes to 2067. DOE used population estimates for 2026 through 2030 to extrapolate population densities for 2031 through 2067. In Nevada, DOE used the *Regional Economic Model, Inc. (REMI)* computer model and data from the Nevada State Demographer to extrapolate population densities. Table G-9 lists the population escalation factors. DOE estimated 2067 population within this 1,600-meter (1 mile) band by multiplying by the appropriate state population escalation factor.

G.4 Shipments

The Yucca Mountain FEIS (DIRS 155970-DOE 2002, Tables J-5, J-6, and J-7) analyzed the shipment of 9,646 rail casks and 1,079 truck casks of spent nuclear fuel and high-level radioactive waste to the repository. Since the completion of the Yucca Mountain FEIS in 2002, DOE has updated the number of rail and truck casks to be shipped to the repository through additional data collection and analysis. In addition, the Department has developed updated estimates of shipments that incorporate the use of TAD canisters and updated cask handling assumptions at each reactor site. Table G-10 summarizes the number of rail and truck casks that would be shipped to the repository. From these estimates, there would be 9,495 rail casks and 2,650 truck casks shipped for the Proposed Action (DIRS 181377-BSC 2007,

Table G-9. Population escalation factors for 2000 to 2067.

States and counties	Population escalation factors	States and counties	Population escalation factors
Alabama	1.2277	Ohio	1.0174
Arkansas	1.4901	Oklahoma	1.3530
Arizona	4.9553	Oregon	2.2607
California	1.9439	Pennsylvania	1.0397
Colorado	1.9161	Rhode Island	1.0998
Connecticut	1.0831	South Carolina	1.6186
District of Columbia	0.7576	South Dakota	1.0604
Delaware	1.5200	Tennessee	1.7775
Florida	3.8088	Texas	2.8136
Georgia	2.1158	Utah	2.7680
Iowa	1.0099	Virginia	1.9803
Idaho	2.3948	Vermont	1.2790
Illinois	1.1383	Washington	2.5613
Indiana	1.2342	Wisconsin	1.2366
Kansas	1.1534	West Virginia	0.9511
Kentucky	1.2541	Wyoming	1.0591
Louisiana	1.1437	Nevada counties	
Massachusetts	1.1938	Churchill	2.2157
Maryland	1.7519	Clark	3.4982
Maine	1.1068	Elko	0.9005
Michigan	1.0760	Esmeralda	1.0219
Minnesota	1.6219	Eureka	0.7722
Missouri	1.3131	Humboldt	0.7332
Mississippi	1.1488	Lander	0.3521
Montana	1.2217	Lincoln	1.6673
North Carolina	2.4719	Lyon	4.8305
North Dakota	0.9445	Nye	3.9746
Nebraska	1.0965	Pershing	1.0541
New Hampshire	1.7545	Storey	2.9660
New Jersey	1.3217	Washoe	2.8725
New Mexico	1.1543	White Pine	0.6826
New York	1.0264	Mineral	0.7327

Section 7). Shipments of the 9,495 rail casks would use 2,833 trains. These estimates were based on 90-percent use of TAD canisters at the commercial sites.

G.5 Radionuclide Inventory

Appendix A of the Yucca Mountain FEIS (DIRS 155970-DOE 2002, pp. A-1 to A-71) provided the basis for the radionuclide inventory DOE used in the transportation analysis in the FEIS (DIRS 155970-DOE 2002, Chapter 6, Appendix J). Since the completion of the FEIS, DOE has updated these radionuclide inventories through additional data collection and analyses.

Table G-10. Updated cask shipment data.

Origin	Origin state	Fuel type	Mode	Casks containing uncanistered SNF	Casks containing TAD canisters	Casks containing other canisters	Total number of casks	Number of shipments
Browns Ferry	AL	BWR	Rail		245		245	82
Farley	AL	PWR	Rail		130		130	44
Arkansas	AR	PWR	Rail		107	20	127	43
Palo Verde	AZ	PWR	Rail		197	2	199	67
Diablo Canyon	CA	PWR	Rail		122		122	41
Humboldt Bay	CA	BWR	Rail			5	5	2
Rancho Seco	CA	PWR	Rail			21	21	7
San Onofre	CA	PWR	Rail		142	9	151	51
Haddam Neck	CT	PWR	Rail			40	40	14
Millstone	CT	BWR	Rail		66		66	22
Millstone	CT	PWR	Rail		110		110	37
Crystal River	FL	PWR	Truck	280			280	280
St. Lucie	FL	PWR	Rail		138		138	46
Turkey Point	FL	PWR	Truck	577			577	577
Hatch	GA	BWR	Rail		177		177	59
Vogtle	GA	PWR	Rail		115		115	39
Arnold	IA	BWR	Rail		58		58	20
Idaho National Laboratory	ID	BWR	Rail			2	2	1
Idaho National Laboratory	ID	DOE	Rail			179	179	36
Idaho National Laboratory	ID	Navy	Rail			400	400	80
Idaho National Laboratory	ID	PWR	Rail			7	7	2
Idaho National Laboratory	ID	HLW	Rail			106	106	22
Idaho National Laboratory	ID	BWR	Truck	1			1	1
Braidwood	IL	PWR	Rail		112		112	38
Byron	IL	PWR	Rail		122		122	41
Clinton	IL	BWR	Truck	327			327	327
Dresden	IL	BWR	Rail		181	14	195	65
LaSalle	IL	BWR	Rail		152		152	51
Morris	IL	BWR	Rail		67		67	23
Morris	IL	PWR	Rail		17		17	6
Quad Cities	IL	BWR	Rail		189		189	63
Zion	IL	PWR	Rail		106		106	36
Wolf Creek	KS	PWR	Rail		60		60	20
River Bend	LA	BWR	Rail		70		70	24
Waterford	LA	PWR	Rail		63		63	21
Pilgrim	MA	BWR	Truck	344			344	344

Table G-10. Updated cask shipment data (continued).

Origin	Origin state	Fuel type	Mode	Casks containing uncanistered SNF	Casks containing TAD canisters	Casks containin g other canisters	Total number of casks	Number of shipments
Yankee Rowe	MA	PWR	Rail			15	15	5
Calvert Cliffs	MD	PWR	Rail		126	12	138	46
Maine Yankee	ME	PWR	Rail			60	60	20
Big Rock Point	MI	BWR	Rail			7	7	3
Cook	MI	PWR	Truck	768			768	768
Fermi	MI	BWR	Rail		63		63	21
Palisades	MI	PWR	Rail		50	12	62	21
Monticello	MN	BWR	Rail		44		44	15
Prairie Island	MN	PWR	Rail		109		109	37
Callaway	MO	PWR	Rail		73		73	25
Grand Gulf	MS	BWR	Rail		100		100	34
Brunswick	NC	BWR	Rail		83	1	84	28
Brunswick	NC	PWR	Rail		15		15	5
Harris	NC	BWR	Rail		64		64	22
Harris	NC	PWR	Rail		64		64	22
McGuire	NC	PWR	Rail		152		152	51
Cooper	NE	BWR	Rail		49		49	17
Fort Calhoun	NE	PWR	Rail		50		50	17
Seabrook	NH	PWR	Rail		50		50	17
Hope Creek	NJ	BWR	Rail		79		79	27
Oyster Creek	NJ	BWR	Rail		79		79	27
Salem	NJ	PWR	Rail		118		118	40
FitzPatrick	NY	BWR	Rail		76		76	26
Ginna	NY	PWR	Truck	313			313	313
Indian Point	NY	PWR	Rail		133		133	45
Nine Mile Point	NY	BWR	Rail		147		147	49
West Valley	NY	HLW	Rail			56	56	12
Davis-Besse	OH	PWR	Rail		51		51	17
Perry	OH	BWR	Rail		75		75	25
Trojan	OR	PWR	Rail			33	33	11
Beaver Valley	PA	PWR	Rail		102		102	34
Limerick	PA	BWR	Rail		155		155	52
Peach Bottom	PA	BWR	Rail		206		206	69
Susquehanna	PA	BWR	Rail		162		162	54
Three Mile Island	PA	PWR	Rail		53		53	18
Catawba	SC	PWR	Rail		123		123	41
Oconee	SC	PWR	Rail		138	48	186	62
Robinson	SC	PWR	Rail		26	5	31	11
Savannah River Site	SC	DOE	Rail			45	45	9

Table G-10. Updated cask shipment data (continued).

Origin	Origin state	Fuel type	Mode	Casks containing uncanistered SNF	Casks containin g TAD canisters	Casks containin g other canisters	Total number of casks	Number of shipments
Savannah River Site	SC	HLW	Rail			698	698	140
Summer	SC	PWR	Rail		55		55	19
Sequoyah	TN	PWR	Rail		120		120	40
Watts Bar	TN	PWR	Rail		30		30	10
Comanche Peak	TX	PWR	Rail		99		99	33
South Texas	TX	PWR	Rail		95		95	32
North Anna	VA	PWR	Rail		117		117	39
Surry	VA	PWR	Rail		121		121	41
Vermont Yankee	VT	BWR	Rail		74		74	25
Columbia	WA	BWR	Rail		66	3	69	23
Hanford Site	WA	DOE	Rail			141	141	29
Hanford Site	WA	HLW	Rail			1064	1064	213
Hanford Site	WA	BWR	Truck	1			1	1
Hanford Site	WA	PWR	Truck	2			2	2
Kewaunee	WI	PWR	Rail		54		54	18
LaCrosse	WI	BWR	Truck	37			37	37
Point Beach	WI	PWR	Rail		98		98	33

Source: DIRS 181377-BSC 2007, Section 7.

BWR = Boiling-water reactor (commercial spent nuclear fuel).
DOE = U.S. Department of Energy spent nuclear fuel.
HLW = High-level radioactive waste.
PWR = Pressurized-water reactor (commercial spent nuclear fuel).
SNF= Spent nuclear fuel.

The primary sources of the new radionuclide inventory information are:

- *PWR Source Term Generation and Evaluation* (DIRS 169061-BSC 2004, all),
- *BWR Source Term Generation and Evaluation* (DIRS 164364-BSC 2003, all),
- *Source Term Estimates for DOE Spent Nuclear Fuels* (DIRS 169354-DOE 2004, all), and
- *Recommended Values for HLW Glass for Consistent Usage on the Yucca Mountain Project* (DIRS 180471-BSC 2007, all).

The radionuclide inventory DOE used in this Repository SEIS represents the radioactivity contained in about 70,000 metric tons of heavy metal (MTHM) of spent nuclear fuel and high-level radioactive waste that would be shipped to the repository. Tables G-11 through G-16 list the updated radionuclide inventories.

DOE spent nuclear fuel was organized into 34 groups based on the fuel compound, fuel enrichment, fuel cladding material, and fuel cladding condition (DIRS 171271-DOE 2004, all). The characteristics of the spent nuclear fuel, including percent enrichment, decay time, and burnup, would affect the radionuclide

Table G-11. Radionuclide inventories (curies) for DOE spent nuclear fuel groups 1 through 8.

	Uranium metal				Uranium oxide			
					Zirconium clad (intact)			Stainless-steel/Hastelloy clad (intact)
Radionuclide	Zirconium-clad LEU Group	Non-Zirconium-clad LEU Group 2	Uranium-zirconium Group 3	Uranium-molybdenum Group 4	HEU Group 5	MEU Group 6	LEU Group 7	HEU Group 8
Actinium-227	5.0×10^{-3}	5.8×10^{-4}	3.0×10^{-3}	8.4×10^{-3}	5.4×10^{-3}	2.9×10^{-5}	4.2×10^{-3}	1.0×10^{-4}
Americium-241	7.1×10^{5}	2.1×10^{4}	1.4×10^{4}	1.8×10^{2}	4.6×10^{2}	4.8×10^{3}	3.7×10^{5}	4.6×10^{-1}
Americium-242m	4.4×10^{2}	3.4×10^{1}	2.2	2.8×10^{-2}	8.6×10^{-1}	9.7	7.8×10^{2}	3.5×10^{-5}
Americium-243	3.7×10^{2}	6.4	1.3	1.6×10^{-2}	1.8	2.1×10^{1}	1.7×10^{3}	4.1×10^{-6}
Carbon-14	1.1×10^{3}	2.0×10^{3}	7.0×10^{2}	1.1×10^{1}	5.3×10^{1}	1.6	6.6×10^{2}	9.5×10^{-1}
Chlorine-36	5.2×10^{-2}	3.7×10^{1}	1.2×10^{-3}	4.8×10^{-3}	2.8×10^{-1}	2.7×10^{-2}	2.1	5.1×10^{-3}
Curium-243	1.7×10^{1}	6.6	3.1×10^{-1}	4.0×10^{-3}	7.5×10^{-1}	8.7	7.6×10^{2}	9.8×10^{-7}
Curium-244	6.5×10^{3}	8.9×10^{1}	6.5	8.3×10^{-2}	1.5×10^{2}	1.7×10^{3}	1.6×10^{5}	8.9×10^{-6}
Cobalt-60	2.7×10^{4}	4.6×10^{5}	4.0×10^{4}	6.8×10^{2}	1.6×10^{4}	1.2×10^{2}	4.7×10^{4}	2.5×10^{2}
Cesium-134	1.1×10^{2}	1.5×10^{2}	5.0	1.2×10^{-1}	1.8	1.9×10^{1}	2.6×10^{3}	1.0×10^{-2}
Cesium-135	7.6×10^{1}	1.9	5.0	4.0	7.0	4.9×10^{-1}	4.2×10^{1}	1.3×10^{-1}
Cesium-137	9.3×10^{6}	2.2×10^{5}	9.0×10^{5}	1.3×10^{5}	3.4×10^{5}	4.8×10^{4}	4.9×10^{6}	5.7×10^{3}
Europium-154	5.2×10^{4}	1.2×10^{3}	4.2×10^{3}	6.9×10^{1}	2.3×10^{2}	7.8×10^{2}	9.1×10^{4}	2.4
Europium-155	2.5×10^{3}	7.7×10^{2}	3.9×10^{2}	1.3×10^{2}	1.7×10^{2}	8.5×10^{1}	1.2×10^{4}	2.5
Iron-55	4.7×10^{1}	6.2×10^{3}	3.7×10^{1}	1.7	2.8×10^{2}	6.8	1.1×10^{3}	4.2
Hydrogen-3	2.6×10^{4}	4.2×10^{3}	1.5×10^{4}	4.9×10^{2}	6.5×10^{2}	7.6×10^{2}	8.7×10^{4}	9.4
Iodine-129	6.5	1.3×10^{-1}	4.7×10^{-1}	1.1×10^{-1}	1.7×10^{-1}	3.3×10^{-2}	2.9	3.0×10^{-3}
Krypton-85	2.1×10^{5}	7.5×10^{3}	2.4×10^{4}	3.7×10^{3}	9.6×10^{3}	1.0×10^{3}	1.3×10^{5}	1.5×10^{2}
Neptunium-237	6.4×10^{1}	1.9	3.5	3.3×10^{-1}	3.0×10^{-1}	3.8×10^{-1}	3.1×10^{1}	4.8×10^{-3}
Protactinium-231	1.2×10^{-2}	1.1×10^{-3}	5.0×10^{-3}	1.7×10^{-2}	1.0×10^{-2}	4.3×10^{-5}	6.9×10^{-3}	2.0×10^{-4}
Lead-210	2.0×10^{-3}	3.6×10^{-4}	2.7×10^{-3}	3.5×10^{-5}	3.7×10^{-7}	2.7×10^{-6}	2.2×10^{-3}	3.1×10^{-9}
Promethium-147	4.7×10^{3}	1.6×10^{4}	6.2×10^{2}	1.1×10^{2}	2.8×10^{2}	5.6×10^{1}	8.9×10^{3}	4.0
Plutonium-238	1.5×10^{5}	3.6×10^{3}	4.0×10^{3}	6.5×10^{1}	2.9×10^{2}	2.5×10^{3}	2.1×10^{5}	1.2
Plutonium-239	2.2×10^{5}	7.1×10^{3}	1.2×10^{4}	1.8×10^{3}	2.0×10^{2}	3.9×10^{2}	4.0×10^{4}	2.8
Plutonium-240	1.7×10^{5}	3.5×10^{3}	5.2×10^{3}	7.1×10^{1}	7.3×10^{1}	5.1×10^{2}	4.4×10^{4}	3.6×10^{-1}
Plutonium-241	4.5×10^{6}	1.4×10^{5}	9.1×10^{4}	1.1×10^{3}	3.5×10^{3}	3.2×10^{4}	3.2×10^{6}	2.7
Plutonium-242	1.1×10^{2}	1.9	1.3	1.6×10^{-2}	1.9×10^{-1}	2.2	1.7×10^{2}	8.2×10^{-6}

Table G-11. Radionuclide inventories (curies) for DOE spent nuclear fuel groups 1 through 8 (continued).

	Uranium metal				Uranium oxide			
					Zirconium clad (intact)			Stainless-steel/Hastelloy clad (intact)
Radionuclide	Zirconium-clad LEU Group 1	Non-Zirconium-clad LEU Group 2	Uranium-zirconium Group 3	Uranium-molybdenum Group 4	HEU Group 5	MEU Group 6	LEU Group 7	HEU Group 8
Radium-226	5.6×10^{-3}	9.7×10^{-4}	7.4×10^{-3}	9.4×10^{-5}	1.0×10^{-6}	7.3×10^{-6}	6.0×10^{-3}	8.2×10^{-9}
Radium-228	4.9×10^{-4}	2.4×10^{-5}	7.4×10^{-4}	1.1×10^{-5}	1.9×10^{-6}	1.8×10^{-7}	5.7×10^{-4}	3.4×10^{-8}
Ruthenium-106	4.4×10^{-3}	1.1×10^{3}	2.1×10^{-4}	2.9×10^{-5}	2.1×10^{-3}	2.6×10^{-1}	5.1×10^{2}	6.3×10^{-7}
Selenium-79	8.4×10^{1}	3.1	7.8	1.5	3.1	4.2×10^{-1}	3.9×10^{1}	5.5×10^{-2}
Tin-126	6.6	2.5	7.5	3.4	2.7	8.5×10^{-1}	7.2×10^{1}	4.8×10^{-2}
Strontium-90	6.7×10^{6}	1.6×10^{5}	7.9×10^{5}	1.1×10^{5}	3.2×10^{5}	3.2×10^{4}	3.4×10^{6}	5.4×10^{3}
Technetium-99	2.8×10^{3}	5.9×10^{1}	2.8×10^{2}	4.2×10^{1}	1.1×10^{2}	1.3×10^{1}	1.2×10^{3}	1.9
Thorium-229	1.8×10^{-3}	1.8×10^{-4}	2.7×10^{-3}	3.8×10^{-5}	3.7×10^{-6}	4.0×10^{-6}	2.3×10^{-3}	6.4×10^{-8}
Thorium-230	5.6×10^{-1}	8.8×10^{-2}	6.7×10^{-1}	8.6×10^{-3}	9.6×10^{-5}	6.9×10^{-4}	5.5×10^{-1}	7.3×10^{-7}
Thorium-232	4.9×10^{-4}	2.4×10^{-5}	7.5×10^{-4}	1.1×10^{-5}	1.9×10^{-6}	1.8×10^{-7}	5.8×10^{-4}	3.5×10^{-8}
Thallium-208	3.0×10^{-2}	2.0×10^{-2}	2.9×10^{-2}	8.7×10^{-4}	5.5×10^{-3}	6.0×10^{-3}	5.1×10^{-1}	8.8×10^{-5}
Uranium-232	8.2×10^{-2}	5.4×10^{-2}	7.8×10^{-2}	2.3×10^{-3}	1.5×10^{-2}	1.6×10^{-2}	1.4	2.4×10^{-4}
Uranium-233	3.9×10^{-1}	3.9×10^{-2}	5.7×10^{-1}	8.0×10^{-3}	8.0×10^{-4}	8.5×10^{-4}	5.0×10^{-1}	1.3×10^{-5}
Uranium-234	1.4×10^{3}	1.9×10^{2}	1.5×10^{3}	1.9×10^{1}	2.6×10^{-1}	1.7	1.2×10^{3}	1.6×10^{-3}
Uranium-235	4.8×10^{1}	8.2×10^{-2}	6.0×10^{-3}	2.0	9.9×10^{-1}	2.0×10^{-1}	2.3	3.9×10^{-1}
Uranium-236	9.7×10^{1}	2.8	1.7×10^{1}	1.3	3.7	2.6×10^{-1}	3.3×10^{1}	6.7×10^{-2}
Uranium-238	7.0×10^{2}	2.1	3.3×10^{-1}	1.0	2.1×10^{-2}	6.0×10^{-1}	3.0×10^{1}	4.7×10^{-3}

Source: Compiled from data contained in DIRS 169354-DOE 2004, Volume II, Appendix C.
HEU = Highly enriched uranium.
LEU = Low-enriched uranium.
MEU = Medium-enriched uranium.

Table G-12. Radionuclide inventories (curies) for DOE spent nuclear fuel groups 9 through 16.

	Uranium oxide							Uranium-aluminum
	Stainless-steel clad (Intact)		Non-aluminum clad Non-intact or declad			Aluminum clad		
Radionuclide	HEU Group 9	LEU Group 10	HEU Group 11	MEU Group 12	LEU Group 13	HEU Group 14	MEU and LEU Group 15	HEU Group 16
Actinium-227	1.4×10^{-4}	9.5×10^{-4}	5.6×10^{-3}	8.5×10^{-4}	4.2×10^{-3}	8.8×10^{-4}	1.3×10^{-5}	1.0×10^{-3}
Americium-241	1.1	1.8×10^{4}	1.9×10^{4}	1.5×10^{3}	4.7×10^{4}	4.9×10^{3}	4.8×10^{1}	5.2×10^{3}
Americium-242m	1.1×10^{-4}	8.8	3.8×10^{1}	3.0	1.1×10^{2}	9.9×10^{-1}	1.6×10^{-2}	1.6
Americium-243	1.2×10^{-5}	4.5	3.7×10^{1}	6.5	2.3×10^{2}	1.5×10^{1}	5.4×10^{-2}	1.8×10^{1}
Carbon-14	2.7	1.9×10^{3}	2.8×10^{2}	1.5×10^{1}	8.5×10^{1}	1.6×10^{-2}	2.1×10^{-4}	3.0×10^{-1}
Chlorine-36	1.5×10^{-2}	3.6×10^{1}	5.2	8.4×10^{-2}	6.5×10^{-1}	1.7×10^{-25}	4.7×10^{-28}	2.7×10^{-4}
Curium-243	4.2×10^{-6}	1.4	2.0	2.7	1.1×10^{2}	2.5	7.9×10^{-3}	3.7
Curium-244	4.9×10^{-5}	6.3×10^{1}	3.9×10^{2}	5.3×10^{2}	2.6×10^{4}	2.1×10^{3}	1.7	3.3×10^{3}
Cobalt-60	1.1×10^{4}	4.4×10^{5}	1.0×10^{5}	1.6×10^{4}	8.1×10^{4}	5.1×10^{1}	1.1	3.6×10^{2}
Cesium-134	1.7×10^{2}	5.2	6.8×10^{2}	7.1	4.4×10^{2}	7.4×10^{4}	1.3×10^{4}	1.3×10^{6}
Cesium-135	3.6×10^{-1}	1.1	1.8	2.0	1.4×10^{1}	5.5	1.2×10^{-1}	9.7
Cesium-137	2.4×10^{4}	1.6×10^{5}	1.0×10^{5}	1.3×10^{5}	1.2×10^{6}	3.2×10^{6}	9.6×10^{4}	6.9×10^{6}
Europium-154	3.2×10^{1}	8.1×10^{2}	3.0×10^{3}	3.3×10^{2}	1.7×10^{4}	5.9×10^{4}	2.5×10^{3}	2.1×10^{5}
Europium-155	1.3×10^{2}	2.4×10^{2}	6.1×10^{2}	2.0×10^{2}	3.4×10^{3}	2.0×10^{4}	1.1×10^{3}	1.1×10^{5}
Iron-55	8.5×10^{3}	4.6×10^{3}	3.5×10^{4}	1.1×10^{3}	5.4×10^{3}	4.6×10^{3}	1.9×10^{2}	3.7×10^{4}
Hydrogen-3	7.3×10^{1}	3.9×10^{3}	7.3×10^{2}	5.1×10^{2}	1.4×10^{4}	7.5×10^{3}	3.3×10^{2}	2.3×10^{4}
Iodine-129	8.7×10^{-3}	9.7×10^{-2}	4.4×10^{-2}	5.6×10^{-2}	5.7×10^{-1}	1.1	2.7×10^{-2}	2.0
Krypton-85	1.4×10^{3}	4.4×10^{3}	4.8×10^{3}	5.2×10^{3}	4.2×10^{4}	1.8×10^{5}	8.9×10^{3}	6.0×10^{5}
Neptunium-237	1.4×10^{-2}	1.7	4.5×10^{-1}	1.9×10^{-1}	4.1	2.2×10^{1}	3.4×10^{-1}	3.4×10^{1}
Protactinium-231	3.4×10^{-4}	2.0×10^{-3}	7.3×10^{-3}	2.0×10^{-3}	9.9×10^{-3}	2.7×10^{-3}	4.6×10^{-5}	3.5×10^{-3}
Lead-210	2.4×10^{-9}	3.5×10^{-4}	5.5×10^{-5}	8.4×10^{-7}	1.2×10^{-5}	6.4×10^{-5}	1.4×10^{-6}	8.7×10^{-5}
Promethium-147	7.5×10^{3}	1.7×10^{3}	3.0×10^{4}	1.0×10^{3}	6.6×10^{3}	1.4×10^{5}	7.1×10^{4}	4.2×10^{6}
Plutonium-238	3.9	3.1×10^{3}	7.1×10^{3}	8.0×10^{2}	2.9×10^{4}	7.8×10^{4}	7.2×10^{2}	1.1×10^{5}
Plutonium-239	8.0	5.7×10^{3}	9.7×10^{2}	1.6×10^{2}	4.4×10^{3}	7.4×10^{2}	1.5×10^{1}	1.3×10^{3}
Plutonium-240	1.0	2.3×10^{3}	6.7×10^{2}	1.6×10^{2}	5.5×10^{3}	4.1×10^{2}	8.8	7.1×10^{2}
Plutonium-241	1.8×10^{1}	1.2×10^{5}	1.1×10^{5}	1.0×10^{4}	5.2×10^{5}	1.0×10^{5}	2.2×10^{3}	2.3×10^{5}
Plutonium-242	2.4×10^{-5}	1.4	5.6	6.7×10^{-1}	2.3×10^{1}	1.5	1.3×10^{-2}	2.0

Table G-12. Radionuclide inventories (curies) for DOE spent nuclear fuel groups 9 through 16 (continued).

	Uranium oxide							Uranium-aluminum
	Stainless-steel Clad (intact)		Non-aluminum clad Non-intact or declad			Aluminum clad		
Radionuclide	HEU Group 9	LEU Group 10	HEU Group 11	MEU Group 12	LEU Group 13	HEU Group 14	MEU and LEU Group 15	HEU Group 16
Radium-226	8.5×10^{-9}	9.4×10^{-4}	1.5×10^{-4}	2.3×10^{-6}	4.2×10^{-5}	2.9×10^{-4}	4.8×10^{-6}	3.6×10^{-4}
Radium-228	9.2×10^{-8}	1.9×10^{-5}	1.4×10^{-3}	5.6×10^{-7}	4.3×10^{-6}	1.9×10^{-8}	2.3×10^{-10}	1.2×10^{-6}
Ruthenium-106	3.8×10^{2}	2.1	1.6×10^{3}	3.3×10^{-2}	2.7×10^{-1}	1.6×10^{3}	5.1×10^{3}	3.6×10^{5}
Selenium-79	1.6×10^{-1}	2.7	7.9×10^{-1}	9.5×10^{-1}	8.3	1.9×10^{1}	4.7×10^{-1}	3.4×10^{1}
Tin-126	1.4×10^{-1}	2.0	6.9×10^{-1}	9.8×10^{-1}	1.2×10^{1}	1.7×10^{1}	4.2×10^{-1}	3.0×10^{1}
Strontium-90	2.3×10^{4}	1.2×10^{5}	9.6×10^{4}	1.2×10^{5}	9.3×10^{5}	3.0×10^{6}	9.2×10^{4}	6.5×10^{6}
Technetium-99	5.6	4.7×10^{1}	2.8×10^{1}	3.3×10^{1}	2.8×10^{2}	6.2×10^{2}	1.5×10^{1}	1.1×10^{3}
Thorium-229	1.0×10^{-7}	1.7×10^{-4}	4.0×10^{-3}	1.8×10^{-6}	3.4×10^{-5}	7.6×10^{-6}	1.1×10^{-7}	9.7×10^{-6}
Thorium-230	1.2×10^{-6}	8.6×10^{-2}	1.3×10^{-2}	2.2×10^{-4}	5.3×10^{-3}	5.2×10^{-2}	9.1×10^{-4}	6.8×10^{-2}
Thorium-232	9.9×10^{-8}	1.9×10^{-5}	1.4×10^{-3}	5.7×10^{-7}	4.4×10^{-6}	2.9×10^{-8}	4.2×10^{-10}	1.5×10^{-6}
Thallium-208	2.9×10^{-4}	1.3×10^{-2}	2.0×10^{-1}	3.3×10^{-3}	7.6×10^{-2}	7.0×10^{-2}	1.6×10^{-3}	1.2×10^{-1}
Uranium-232	8.0×10^{-4}	3.6×10^{-2}	5.4×10^{-1}	9.0×10^{-3}	2.1×10^{-1}	1.9×10^{-1}	4.7×10^{-3}	3.4×10^{-1}
Uranium-233	3.7×10^{-5}	3.6×10^{-2}	8.2×10^{-1}	4.5×10^{-4}	9.7×10^{-3}	4.2×10^{-3}	7.8×10^{-5}	6.7×10^{-3}
Uranium-234	4.4×10^{-3}	1.9×10^{2}	2.9×10^{1}	5.4×10^{-1}	1.7×10^{1}	2.3×10^{2}	6.6	4.3×10^{2}
Uranium-235	2.7×10^{-1}	1.8×10^{-1}	2.4	1.3×10^{-1}	4.6	7.8	6.2×10^{-2}	1.3×10^{1}
Uranium-236	1.9×10^{-1}	2.6	9.8×10^{-1}	1.1	7.5	2.4×10^{1}	5.6×10^{-1}	4.2×10^{1}
Uranium-238	1.9×10^{-1}	2.6×10^{-1}	3.6×10^{-1}	1.3×10^{-1}	2.7×10^{1}	1.3×10^{-1}	8.3×10^{-2}	3.2×10^{-1}

Source: Compiled from data contained in DIRS 169354-DOE 2004, Volume II, Appendix C.
HEU= Highly enriched uranium.
LEU = Low-enriched uranium.
MEU= Medium-enriched uranium.

Table G-13. Radionuclide inventories (curies) for DOE spent nuclear fuel groups 17 through 24.

			Thorium/uranium carbide		Plutonium/ uranium carbide	Mixed oxide		
Radionuclide	Uranium-aluminum MEU Group 17	Uranium silicide Group 18	TRISO or BISO particles in graphite Group 19	Mono-pyrolytic carbon particles Group 20	Non-graphite non-sodium bonded Group 21	Zirconium clad Group 22	Stainless-steel clad Group 23	Non-stainless steel Non-zirconium clad Group 24
Actinium-227	6.1×10^{-5}	2.7×10^{-4}	2.6	2.3×10^{-1}	2.1×10^{-8}	1.6×10^{-1}	4.2×10^{-2}	4.9×10^{-3}
Americium-241	1.9×10^{3}	8.6×10^{3}	2.3×10^{3}	1.8×10^{2}	8.9×10^{2}	5.8×10^{5}	2.5×10^{5}	3.0×10^{4}
Americium-242m	1.3	6.1	2.2	1.4×10^{-1}	1.7×10^{1}	1.2×10^{3}	2.1×10^{3}	2.8×10^{2}
Americium-243	1.1	4.4	4.0×10^{1}	2.7	9.0×10^{-1}	1.1×10^{3}	4.4×10^{2}	6.1×10^{1}
Carbon-14	3.0×10^{-2}	1.2	2.0×10^{1}	1.4	2.2×10^{-1}	8.3×10^{3}	2.6×10^{3}	3.7×10^{2}
Chlorine-36	2.5×10^{-5}	1.2×10^{-3}	9.2×10^{-1}	6.2×10^{-2}	2.9×10^{-6}	1.6×10^{2}	4.9×10^{1}	7.0
Curium-243	4.3×10^{-1}	2.0	3.0×10^{1}	1.5	4.9	7.7×10^{1}	5.8×10^{2}	7.4×10^{1}
Curium-244	3.3×10^{1}	1.3×10^{2}	9.0×10^{3}	3.8×10^{2}	2.1×10^{1}	1.2×10^{4}	7.7×10^{3}	1.2×10^{3}
Cobalt-60	3.0×10^{1}	9.1×10^{2}	2.3×10^{3}	2.7×10^{1}	8.9×10^{1}	1.9×10^{6}	3.5×10^{6}	6.4×10^{5}
Cesium-134	1.3×10^{5}	2.6×10^{5}	3.7×10^{3}	1.5×10^{1}	2.0×10^{2}	9.4×10^{1}	4.1×10^{4}	5.1×10^{3}
Cesium-135	1.3	4.8	2.1×10^{1}	1.4	4.0×10^{-1}	3.2×10^{1}	4.9×10^{1}	6.4
Cesium-137	9.1×10^{5}	2.5×10^{6}	1.5×10^{6}	7.8×10^{4}	1.6×10^{4}	1.5×10^{6}	2.3×10^{6}	3.2×10^{5}
Europium-154	2.4×10^{4}	9.2×10^{4}	3.9×10^{4}	9.3×10^{2}	3.0×10^{2}	8.6×10^{4}	1.1×10^{5}	1.8×10^{4}
Europium-155	1.1×10^{4}	3.7×10^{4}	5.9×10^{3}	6.3×10^{1}	3.8×10^{2}	5.3×10^{3}	6.7×10^{4}	9.0×10^{3}
Iron-55	1.0×10^{4}	4.7×10^{4}	1.6	5.3×10^{-3}	2.6×10^{1}	2.0×10^{4}	4.8×10^{5}	5.5×10^{4}
Hydrogen-3	3.3×10^{3}	8.8×10^{3}	6.9×10^{3}	2.3×10^{2}	6.0×10^{1}	1.7×10^{4}	1.7×10^{4}	2.7×10^{3}
Iodine-129	2.4×10^{-1}	6.6×10^{-1}	8.7×10^{-1}	5.9×10^{-2}	1.1×10^{-2}	7.8×10^{-1}	1.3	1.7×10^{-1}
Krypton-85	8.7×10^{4}	2.2×10^{5}	7.9×10^{4}	2.3×10^{3}	4.7×10^{2}	4.2×10^{4}	8.5×10^{4}	1.2×10^{4}
Neptunium-237	2.3	4.7	1.1×10^{1}	7.3×10^{-1}	2.5×10^{-2}	1.1×10^{1}	5.6	7.6×10^{-1}
Protactinium-231	3.4×10^{-4}	1.2×10^{-3}	4.1	2.8×10^{-1}	5.7×10^{-8}	2.0×10^{-1}	6.1×10^{-2}	8.7×10^{-3}
Lead-210	1.0×10^{-6}	1.2×10^{-5}	7.3×10^{-4}	8.3×10^{-5}	4.1×10^{-9}	1.6×10^{-3}	3.2×10^{-4}	1.1×10^{-5}
Promethium-147	7.5×10^{5}	1.8×10^{6}	5.2×10^{3}	1.7×10^{1}	1.1×10^{3}	1.9×10^{3}	2.2×10^{5}	2.8×10^{4}
Plutonium-238	4.8×10^{3}	8.8×10^{3}	1.5×10^{5}	9.5×10^{3}	2.2×10^{2}	1.5×10^{5}	3.8×10^{4}	3.0×10^{3}
Plutonium-239	1.3×10^{3}	6.7×10^{3}	1.2×10^{2}	7.9	1.0×10^{3}	2.2×10^{4}	1.5×10^{5}	0.0

Table G-13. Radionuclide inventories (curies) for DOE spent nuclear fuel groups 17 through 24 (continued).

			Thorium/uranium carbide		Plutonium/ uranium carbide	Mixed oxide		
Radionuclide	Uranium-aluminum MEU Group 17	Uranium silicide Group 18	TRISO or BISO particles in graphite Group 19	Mono-pyrolytic carbon particles Group 20	Non-graphite non-sodium bonded Group 21	Zirconium clad Group 22	Stainless-steel clad Group 23	Non-stainless steel non-zirconium clad Group 24
Plutonium-240	7.1×10^{2}	3.5×10^{3}	2.2×10^{2}	1.6×10^{1}	8.4×10^{2}	1.3×10^{4}	1.1×10^{5}	3.9×10^{3}
Plutonium-241	1.0×10^{5}	4.9×10^{5}	3.1×10^{4}	1.1×10^{3}	2.3×10^{4}	1.3×10^{6}	4.2×10^{6}	2.6×10^{4}
Plutonium-242	4.5×10^{-1}	2.0	3.4	2.3×10^{-1}	2.7×10^{-1}	1.3×10^{2}	4.4×10^{1}	1.8
Radium-226	9.0×10^{-6}	4.7×10^{-5}	1.2×10^{-3}	1.6×10^{-4}	1.5×10^{-8}	4.4×10^{-3}	9.2×10^{-4}	5.1×10^{-5}
Radium-228	1.2×10^{-7}	4.9×10^{-6}	7.8×10^{-1}	5.4×10^{-2}	8.1×10^{-13}	4.1×10^{-2}	1.2×10^{-2}	1.7×10^{-3}
Ruthenium-106	6.4×10^{4}	1.7×10^{5}	6.5×10^{-1}	7.9×10^{-2}	5.9×10^{1}	7.4×10^{-1}	1.2×10^{4}	1.5×10^{3}
Selenium-79	4.1	1.1×10^{1}	1.8×10^{1}	1.2	8.5×10^{-2}	1.4×10^{1}	1.3×10^{1}	1.7
Tin-126	3.7	1.0×10^{1}	1.9×10^{1}	1.3	3.7×10^{-1}	1.3×10^{1}	4.0×10^{1}	5.2
Strontium-90	8.6×10^{5}	2.3×10^{6}	1.5×10^{6}	7.4×10^{4}	5.8×10^{3}	1.4×10^{6}	1.2×10^{6}	1.7×10^{5}
Technetium-99	1.4×10^{2}	3.9×10^{2}	2.9×10^{2}	1.9×10^{1}	3.3	4.8×10^{2}	4.8×10^{2}	6.2×10^{1}
Thorium-229	5.5×10^{-7}	5.1×10^{-6}	5.8	6.2×10^{-1}	1.6×10^{-8}	1.2×10^{-1}	2.9×10^{-2}	2.7×10^{-3}
Thorium-230	3.6×10^{-3}	8.4×10^{-3}	1.2×10^{-1}	1.1×10^{-2}	3.1×10^{-6}	4.0×10^{-1}	9.6×10^{-2}	9.1×10^{-3}
Thorium-232	1.4×10^{-7}	6.4×10^{-6}	2.5	1.7×10^{-1}	1.2×10^{-12}	4.1×10^{-2}	1.3×10^{-2}	1.8×10^{-3}
Thallium-208	9.8×10^{-3}	1.7×10^{-2}	5.8×10^{2}	3.5×10^{1}	4.3×10^{-3}	6.0	2.5	3.7×10^{-1}
Uranium-232	2.9×10^{-2}	4.8×10^{-2}	1.6×10^{3}	9.4×10^{1}	1.2×10^{-2}	1.6×10^{1}	6.7	1.0
Uranium-233	5.0×10^{-4}	4.3×10^{-3}	1.8×10^{3}	1.2×10^{2}	2.5×10^{-6}	2.5×10^{1}	7.7	1.1
Uranium-234	3.7×10^{1}	4.7×10^{1}	2.4×10^{2}	1.7×10^{1}	2.2×10^{-2}	8.7×10^{2}	2.7×10^{2}	3.9×10^{1}
Uranium-235	4.4×10^{-1}	1.2	3.6	2.4×10^{-1}	1.9×10^{-4}	4.0×10^{1}	1.2×10^{1}	1.8
Uranium-236	4.7	1.2×10^{1}	7.4	5.0×10^{-1}	1.1×10^{-3}	1.6×10^{1}	5.1	7.3×10^{-1}
Uranium-238	7.9×10^{-1}	2.2	4.5×10^{-2}	3.0×10^{-3}	1.8×10^{-2}	8.0	5.0	3.9×10^{-1}

Source: Compiled from data contained in DIRS 169354-DOE 2004, Volume II, Appendix C.
HEU= Highly enriched uranium.
LEU = Low-enriched uranium.
MEU= Medium-enriched uranium.

Table G-14. Radionuclide inventories (curies) for DOE spent nuclear fuel groups 25 through 30, 32, and 34.

			Uranium/zirconium hydride					
	Thorium/uranium oxide		Stainless steel/Incoloy clad		Aluminum clad			
Radionuclide	Zirconium clad Group 25	Stainless-steel clad Group 26	HEU Group 27	MEU Group 28	MEU Group 29	Declad Group 30	Naval spent nuclear fuel Group 32[a]	Miscellaneous Group 34
Actinium-227	3.9×10^{1}	7.4	2.1×10^{-5}	6.5×10^{-5}	2.1×10^{-5}	2.7×10^{-4}	3.9×10^{-2}	5.0×10^{-3}
Americium-241	1.1×10^{2}	7.1×10^{3}	3.8×10^{2}	1.1×10^{2}	3.0×10^{1}	1.1×10^{2}	2.0×10^{4}	2.7×10^{3}
Americium-242m	7.3×10^{-1}	1.6×10^{1}	8.2×10^{-1}	7.2×10^{-2}	1.9×10^{-2}	3.3×10^{-2}	1.8×10^{2}	6.9
Americium-243	1.5×10^{-1}	1.5×10^{1}	1.1	7.7×10^{-3}	2.4×10^{-3}	4.2×10^{-3}	2.7×10^{2}	1.5×10^{1}
Carbon-14	4.4×10^{1}	1.2×10^{2}	4.4	6.7	4.4×10^{-1}	3.6	6.4×10^{3}	3.9×10^{1}
Californium-252	0.0	0.0	0.0	0.0	0.0	0.0	4.8×10^{-4}	0.0
Chlorine-36	8.5×10^{-1}	2.2	9.3×10^{-2}	1.5×10^{-1}	4.3×10^{-4}	8.0×10^{-2}	2.8×10^{2}	7.0×10^{-1}
Curium-242	0.0	0.0	0.0	0.0	0.0	0.0	5.6×10^{2}	0.0
Curium-243	1.8×10^{-1}	1.0	1.1	8.8×10^{-3}	2.4×10^{-3}	1.7×10^{-3}	3.2×10^{2}	8.1×10^{-1}
Curium-244	9.8	2.2×10^{2}	1.1×10^{2}	8.2×10^{-2}	2.6×10^{-2}	8.6×10^{-3}	2.5×10^{4}	5.4×10^{1}
Curium-245	0.0	0.0	0.0	0.0	0.0	0.0	2.9	0.0
Curium-246	0.0	0.0	0.0	0.0	0.0	0.0	5.6×10^{-1}	0.0
Curium-247	0.0	0.0	0.0	0.0	0.0	0.0	3.8×10^{-6}	0.0
Curium-248	0.0	0.0	0.0	0.0	0.0	0.0	1.0×10^{-5}	0.0
Cobalt-60	1.5×10^{3}	9.5×10^{4}	2.3×10^{4}	5.8×10^{4}	2.2×10^{2}	9.8×10^{1}	1.5×10^{6}	1.1×10^{4}
Cobalt-60 (Crud)	0.0	0.0	0.0	0.0	0.0	0.0	2.3×10^{3}	0.0
Cesium-134	3.5×10^{2}	1.1×10^{1}	9.8×10^{3}	4.0×10^{3}	7.1×10^{2}	7.0×10^{-4}	3.4×10^{7}	8.8×10^{1}
Cesium-135	1.3×10^{1}	2.6	6.9×10^{-1}	1.7	3.2×10^{-1}	9.1×10^{-1}	1.8×10^{3}	4.4
Cesium-137	8.8×10^{5}	1.4×10^{5}	8.0×10^{4}	1.4×10^{5}	2.4×10^{4}	2.8×10^{4}	1.8×10^{8}	2.1×10^{5}
Europium-154	9.1×10^{3}	3.2×10^{3}	2.7×10^{3}	7.1×10^{2}	1.0×10^{4}	1.2×10^{1}	0.0	5.1×10^{2}
Europium-155	1.3×10^{3}	3.0×10^{2}	9.8×10^{2}	1.3×10^{3}	3.1×10^{3}	1.6	0.0	2.3×10^{3}
Iron-55	1.6×10^{1}	3.8×10^{3}	1.2×10^{4}	3.4×10^{4}	6.0×10^{1}	1.4×10^{-1}	0.0	3.7×10^{2}
Hydrogen-3	1.8×10^{3}	5.5×10^{2}	2.5×10^{2}	5.2×10^{2}	8.5×10^{1}	2.5×10^{1}	5.6×10^{5}	1.1×10^{3}
Iodine-129	7.5×10^{-1}	1.3×10^{-1}	2.5×10^{-2}	3.8×10^{-2}	7.4×10^{-3}	2.1×10^{-2}	4.8×10^{1}	1.1×10^{-1}
Krypton-85	5.6×10^{4}	5.8×10^{3}	5.8×10^{3}	1.2×10^{4}	1.9×10^{3}	3.9×10^{2}	1.4×10^{7}	1.3×10^{4}

Table G-14. Radionuclide inventories (curies) for DOE spent nuclear fuel groups 25 through 30, 32, and 34 (continued).

	Thorium/uranium oxide		Uranium/zirconium hydride					
			Stainless steel/Incoloy clad		Aluminum clad			
Radionuclide	Zirconium clad Group 25	Stainless-steel clad Group 26	HEU Group 27	MEU Group 28	MEU Group 29	Declad Group 30	Naval spent nuclear fuel Group 32[a]	Miscellaneous Group 34
Niobium-93m	0.0	0.0	0.0	0.0	0.0	0.0	1.4×10^{3}	0.0
Niobium-94	0.0	0.0	0.0	0.0	0.0	0.0	7.2×10^{4}	0.0
Nickel-59	0.0	0.0	0.0	0.0	0.0	0.0	2.5×10^{4}	0.0
Nickel-63	0.0	0.0	0.0	0.0	0.0	0.0	3.1×10^{6}	0.0
Neptunium-237	5.9×10^{-2}	1.5×10^{-1}	4.2×10^{-1}	6.5×10^{-2}	1.5×10^{-2}	3.7×10^{-2}	6.4×10^{2}	3.6×10^{-1}
Protactinium-231	5.7×10^{1}	9.1	5.3×10^{-5}	2.3×10^{-4}	5.6×10^{-5}	4.4×10^{-4}	2.1×10^{-1}	1.2×10^{-2}
Lead-210	5.6×10^{-3}	1.1×10^{-3}	1.9×10^{-8}	1.2×10^{-9}	9.8×10^{-10}	2.0×10^{-8}	3.6×10^{-4}	7.7×10^{-6}
Palladium-107	0.0	0.0	0.0	0.0	0.0	0.0	2.4×10^{1}	0.0
Promethium-147	1.7×10^{3}	2.3×10^{2}	1.8×10^{4}	9.3×10^{4}	1.4×10^{4}	4.1×10^{-1}	0.0	2.2×10^{4}
Plutonium-238	2.2×10^{2}	2.9×10^{3}	1.8×10^{3}	5.3×10^{1}	1.3×10^{1}	2.1×10^{1}	4.8×10^{6}	8.6×10^{2}
Plutonium-239	1.3×10^{1}	3.8×10^{2}	4.9×10^{1}	2.9×10^{2}	5.7×10^{1}	1.6×10^{2}	4.8×10^{3}	2.1×10^{3}
Plutonium-240	7.6	2.7×10^{2}	4.0×10^{1}	1.1×10^{2}	2.3×10^{1}	6.0×10^{1}	5.6×10^{3}	1.9×10^{2}
Plutonium-241	1.1×10^{3}	7.1×10^{4}	1.1×10^{4}	4.9×10^{3}	1.0×10^{3}	3.3×10^{2}	1.6×10^{6}	1.7×10^{4}
Plutonium-242	1.9×10^{-2}	2.2	1.7×10^{-1}	1.2×10^{-2}	3.1×10^{-3}	6.6×10^{-3}	3.2×10^{1}	7.2×10^{-1}
Radium-226	6.8×10^{-3}	1.7×10^{-3}	7.8×10^{-8}	5.4×10^{-9}	3.0×10^{-9}	4.8×10^{-8}	2.2×10^{-3}	2.0×10^{-5}
Radium-228	2.2	3.5×10^{-1}	7.3×10^{-7}	1.0×10^{-5}	2.0×10^{-6}	7.2×10^{-6}	7.2×10^{-5}	3.1×10^{-4}
Rhodium-102	0.0	0.0	0.0	0.0	0.0	0.0	1.1×10^{1}	0.0
Ruthenium-106	1.8×10^{-2}	3.5×10^{-3}	1.4×10^{3}	4.0×10^{3}	6.4×10^{2}	9.7×10^{-11}	2.4×10^{6}	3.9×10^{1}
Selenium-79	1.7×10^{1}	2.9	4.5×10^{-1}	6.8×10^{-1}	1.3×10^{-1}	3.7×10^{-1}	1.4×10^{2}	1.6
Samarium-151	0.0	0.0	0.0	0.0	0.0	0.0	5.6×10^{5}	0.0
Tin-126	1.9×10^{1}	3.2	4.2×10^{-1}	6.3×10^{-1}	1.2×10^{-1}	3.5×10^{-1}	4.8×10^{2}	3.6
Strontium-90	8.9×10^{5}	1.4×10^{5}	7.5×10^{4}	1.3×10^{5}	2.3×10^{4}	2.5×10^{4}	1.8×10^{8}	1.9×10^{5}
Technetium-99	1.5×10^{2}	3.1×10^{1}	1.4×10^{1}	2.3×10^{1}	4.4	1.3×10^{1}	2.8×10^{4}	4.5×10^{1}
Thorium-229	2.2×10^{1}	4.9	5.1×10^{-6}	9.0×10^{-6}	2.7×10^{-6}	2.2×10^{-5}	3.8×10^{-3}	1.8×10^{-3}

Table G-14. Radionuclide inventories (curies) for DOE spent nuclear fuel groups 25 through 30, 32, and 34 (continued).

			Uranium/zirconium hydride					
	Thorium/uranium oxide		Stainless steel/Incoloy clad		Aluminum clad			
Radionuclide	Zirconium clad Group 25	Stainless-steel clad Group 26	HEU Group 27	MEU Group 28	MEU Group 29	Declad Group 30	Naval spent nuclear fuel Group 32[a]	Miscellaneous Group 34
Thorium-230	4.9×10^{-1}	9.0×10^{-2}	1.6×10^{-5}	1.2×10^{-6}	4.1×10^{-7}	3.7×10^{-6}	7.2×10^{-1}	1.9×10^{-3}
Thorium-232	4.5	8.0×10^{-1}	8.5×10^{-7}	1.3×10^{-5}	2.4×10^{-6}	7.2×10^{-6}	9.2×10^{-5}	2.7×10^{-2}
Thallium-208	7.2×10^{3}	1.1×10^{3}	5.0×10^{-3}	8.7×10^{-4}	1.9×10^{-4}	3.4×10^{-4}	0.0	4.5×10^{-1}
Uranium-232	2.0×10^{4}	2.9×10^{3}	1.4×10^{-2}	2.5×10^{-3}	5.3×10^{-4}	9.1×10^{-4}	2.2×10^{2}	1.2
Uranium-233	1.4×10^{4}	2.5×10^{3}	2.4×10^{-3}	6.3×10^{-3}	1.3×10^{-3}	3.5×10^{-3}	1.2	8.7×10^{1}
Uranium-234	3.9×10^{2}	7.4×10^{1}	1.2×10^{-1}	8.7×10^{-3}	2.1×10^{-3}	8.1×10^{-3}	6.0×10^{3}	4.4
Uranium-235	3.0×10^{-2}	5.3×10^{-1}	2.1×10^{-1}	5.0×10^{-1}	1.3×10^{-1}	2.6×10^{-2}	1.2×10^{2}	2.1×10^{-1}
Uranium-236	6.3×10^{-2}	2.2×10^{-1}	4.7×10^{-1}	6.6×10^{-1}	1.3×10^{-1}	3.6×10^{-1}	1.0×10^{3}	1.3
Uranium-238	1.8×10^{-3}	1.1×10^{-1}	1.6×10^{-2}	3.9×10^{-1}	9.7×10^{-2}	1.5×10^{-2}	4.8×10^{-1}	8.6×10^{-2}
Zirconium-93	0.0	0.0	0.0	0.0	0.0	0.0	4.4×10^{3}	0.0

Source: Compiled from data contained in DIRS 169354-DOE 2004, Volume II, Appendix C.
Note: There are no shipments of Group 31 and 33 spent nuclear fuel.
a. Radionuclide inventory is for 400 casks. Single cask naval spent fuel inventory is from DIRS 155857-McKenzie 2001, Table 3.
HEU= Highly enriched uranium.
LEU = Low-enriched uranium.
MEU= Medium-enriched uranium.

Table G-15. Radionuclide inventories (curies) for commercial spent nuclear fuel.

Radionuclide	BWR SNF inventory (Ci/assembly)[a]	BWR SNF total inventory[a]	PWR SNF inventory (Ci/assembly)[b]	PWR SNF total inventory[b]
Americium-241	3.73×10^{2}	4.84×10^{7}	1.28×10^{3}	1.21×10^{8}
Americium-242m	2.88	3.74×10^{5}	7.99	7.58×10^{5}
Americium-243	8.63	1.12×10^{6}	3.93×10^{1}	3.73×10^{6}
Carbon-14	1.69×10^{-1}	2.19×10^{4}	4.35×10^{-1}	4.13×10^{4}
Cadmium-113m	6.23	8.08×10^{5}	2.34×10^{1}	2.22×10^{6}
Cerium-144	1.73×10^{1}	2.24×10^{6}	6.99×10^{1}	6.63×10^{6}
Curium-242	2.38	3.09×10^{5}	6.60	6.26×10^{5}
Curium-243	5.55	7.20×10^{5}	2.48×10^{1}	2.35×10^{6}
Curium-244	9.23×10^{2}	1.20×10^{8}	5.85×10^{3}	5.55×10^{8}
Curium-245	9.07×10^{-2}	1.18×10^{4}	8.16×10^{-1}	7.74×10^{4}
Curium-246	4.26×10^{-2}	5.53×10^{3}	4.07×10^{-1}	3.86×10^{4}
Cobalt-60	1.14×10^{2}	1.48×10^{7}	2.17×10^{3}	2.06×10^{8}
Cobalt-60 (Crud)	5.66×10^{1}	7.34×10^{6}	1.69×10^{1}	1.60×10^{6}
Cesium-134	1.31×10^{3}	1.70×10^{8}	5.43×10^{3}	5.15×10^{8}
Cesium-137	2.41×10^{4}	3.13×10^{9}	7.16×10^{4}	6.79×10^{9}
Europium-154	7.79×10^{2}	1.01×10^{8}	3.01×10^{3}	2.85×10^{8}
Europium-155	1.93×10^{2}	2.51×10^{7}	6.42×10^{2}	6.09×10^{7}
Iron-55 (Crud)	9.84×10^{1}	1.28×10^{7}	2.09×10^{2}	1.98×10^{7}
Hydrogen-3	1.05×10^{2}	1.36×10^{7}	3.05×10^{2}	2.90×10^{7}
Iodine-129	9.22×10^{-3}	1.20×10^{3}	2.76×10^{-2}	2.62×10^{3}
Krypton-85	1.17×10^{3}	1.52×10^{8}	3.39×10^{3}	3.21×10^{8}
Neptunium-237	8.74×10^{-2}	1.13×10^{4}	2.94×10^{-1}	2.79×10^{4}
Promethium-147	2.11×10^{3}	2.74×10^{8}	6.06×10^{3}	5.75×10^{8}
Plutonium-238	1.02×10^{3}	1.32×10^{8}	3.98×10^{3}	3.77×10^{8}
Plutonium-239	5.41×10^{1}	7.02×10^{6}	1.75×10^{2}	1.66×10^{7}
Plutonium-240	1.27×10^{2}	1.65×10^{7}	3.63×10^{2}	3.44×10^{7}
Plutonium-241	1.57×10^{4}	2.04×10^{9}	5.64×10^{4}	5.35×10^{9}
Plutonium-242	7.08×10^{-1}	9.18×10^{4}	2.48	2.35×10^{5}
Ruthenium-106	9.05×10^{1}	1.17×10^{7}	4.04×10^{2}	3.83×10^{7}
Antimony-125	1.45×10^{2}	1.88×10^{7}	5.20×10^{2}	4.93×10^{7}
Strontium-90	1.66×10^{4}	2.15×10^{9}	4.51×10^{4}	4.28×10^{9}
Uranium-232	8.74×10^{-3}	1.13×10^{3}	3.61×10^{-2}	3.42×10^{3}
Uranium-234	2.39×10^{-1}	3.10×10^{4}	5.24×10^{-1}	4.97×10^{4}
Uranium-236	7.45×10^{-2}	9.66×10^{3}	1.77×10^{-1}	1.68×10^{4}
Uranium-238	6.24×10^{-2}	8.09×10^{3}	1.46×10^{-1}	1.38×10^{4}
Yttrium-90	1.66×10^{4}	2.15×10^{9}	4.51×10^{4}	4.28×10^{9}

Source: DIRS 169061-BSC 2004, all; DIRS 164364-BSC 2003, all.

a. Total inventory for pressurized water reactor spent nuclear fuel shipped in rail casks is based on 94,817 assemblies (calculated from rail and truck shipments and cask capacities from DIRS 181377-BSC 2007, Section 7).

b. Total inventory for boiling water reactor spent nuclear fuel shipped in rail casks is based on 129,721 assemblies (calculated from rail and truck shipments and cask capacities from DIRS 181377-BSC 2007, Section 7).

PWR = pressurized water reactor.
BWR = boiling water reactor.
SNF = spent nuclear fuel.

Table G-16. Radionuclide inventories (curies) for high-level radioactive waste.

Radionuclide	Hanford Site[a]	Idaho National Laboratory[b]	Savannah River Site[c]	West Valley[d]
Actinium-227	0.0	7.38×10^{1}	0.0	4.92×10^{1}
Americium-241	5.41×10^{3}	1.08×10^{5}	7.98×10^{5}	4.58×10^{4}
Americium-242	7.86×10^{-3}	0.0	0.0	6.56×10^{2}
Americium-242m	7.86×10^{-3}	0.0	4.55×10^{2}	6.58×10^{2}
Americium-243	6.42×10^{-3}	1.13×10^{1}	1.29×10^{3}	6.10×10^{2}
Barium-137m	4.76×10^{6}	2.80×10^{7}	2.18×10^{8}	4.80×10^{6}
Carbon-14	1.29×10^{-2}	0.0	0.0	0.0
Cadmium-113m	0.0	7.79×10^{3}	8.96×10^{-8}	0.0
Cerium-144	0.0	0.0	6.30×10^{3}	0.0
Californium-249	0.0	0.0	1.25×10^{1}	0.0
Californium-251	0.0	0.0	2.87×10^{1}	0.0
Curium-242	7.86×10^{-3}	0.0	0.0	5.43×10^{2}
Curium-243	3.99×10^{-4}	8.28	1.45×10^{3}	0.0
Curium-244	1.24×10^{-2}	1.57×10^{2}	6.51×10^{6}	6.15×10^{3}
Curium-245	1.71×10^{-6}	0.0	5.22×10^{2}	0.0
Curium-246	4.02×10^{-8}	0.0	1.52×10^{2}	0.0
Curium-247	1.43×10^{-14}	0.0	5.99×10^{-3}	0.0
Curium-248	4.32×10^{-15}	0.0	0.0	0.0
Cobalt-60	3.98×10^{2}	1.87×10^{3}	2.50×10^{6}	0.0
Cesium-134	6.75×10^{1}	6.71×10^{2}	8.40×10^{5}	0.0
Cesium-135	7.53×10^{1}	0.0	9.17×10^{2}	1.93×10^{2}
Cesium-137	4.90×10^{6}	2.80×10^{7}	2.33×10^{8}	5.08×10^{6}
Europium-152	0.0	7.74×10^{2}	0.0	0.0
Europium-154	2.08×10^{4}	5.03×10^{4}	5.88×10^{6}	0.0
Europium-155	1.41×10^{2}	1.82×10^{3}	2.35×10^{3}	0.0
Hydrogen-3	6.70×10^{3}	0.0	0.0	0.0
Iodine-129	2.61	3.61×10^{1}	2.57×10^{-1}	0.0
Niobium-93m	6.42×10^{2}	2.00×10^{3}	5.15×10^{2}	2.03×10^{2}
Niobium-94	2.48×10^{-3}	0.0	0.0	0.0
Nickel-59	0.0	1.03×10^{3}	7.56×10^{2}	1.19×10^{2}
Nickel-63	0.0	9.06×10^{4}	4.94×10^{4}	9.64×10^{3}
Neptunium-237	2.85	1.06×10^{2}	1.19×10^{2}	3.55×10^{1}
Neptunium-238	0.0	0.0	0.0	2.97
Neptunium-239	0.0	0.0	0.0	6.10×10^{2}
Protactinium-231	0.0	2.05×10^{2}	0.0	4.91×10^{1}
Palladium-107	0.0	0.0	4.52	0.0
Promethium-147	9.15×10^{3}	0.0	1.70×10^{7}	0.0
Praseodymium-144	0.0	0.0	6.30×10^{3}	0.0
Plutonium-238	5.04×10^{4}	3.42×10^{3}	2.08×10^{7}	5.19×10^{3}
Plutonium-239	8.37×10^{2}	5.20×10^{4}	1.72×10^{5}	1.56×10^{3}
Plutonium-240	7.26×10^{2}	9.25×10^{3}	1.17×10^{5}	1.11×10^{3}
Plutonium-241	2.98×10^{4}	6.10×10^{4}	1.22×10^{7}	2.67×10^{4}
Plutonium-242	1.58	7.53×10^{-1}	3.89×10^{2}	3.04×10^{-3}
Radium-226	2.60×10^{-3}	6.78×10^{-2}	0.0	0.0

Table G-16. Radionuclide inventories (curies) for high-level radioactive waste (continued).

Radionuclide	Hanford Site[a]	Idaho National Laboratory[b]	Savannah River Site[c]	West Valley[d]
Radium-228	0.0	1.58×10^{1}	0.0	2.07
Ruthenium-106	0.0	1.51	1.65×10^{4}	0.0
Antimony-125	2.72×10^{2}	1.86×10^{3}	0.0	0.0
Antimony-126	0.0	0.0	0.0	7.59
Antimony-126m	0.0	0.0	0.0	5.42×10^{1}
Selenium-79	0.0	9.19×10^{1}	2.07×10^{2}	0.0
Samarium-151	0.0	2.46×10^{6}	4.27×10^{5}	5.08×10^{4}
Tin-126	4.12×10^{1}	4.36×10^{2}	1.08×10^{2}	5.42×10^{1}
Strontium-90	6.01×10^{6}	3.06×10^{7}	2.67×10^{8}	2.89×10^{6}
Technetium-99	1.58×10^{3}	2.24×10^{4}	5.46×10^{4}	8.90×10^{2}
Thorium-229	0.0	1.51	3.07×10^{-1}	7.51×10^{-3}
Thorium-230	1.72×10^{-1}	0.0	2.76×10^{-2}	3.28×10^{-4}
Thorium-232	4.48×10^{-8}	6.02	3.30	2.54
Thallium-208	0.0	0.0	0.0	6.65×10^{-1}
Uranium-232	2.75×10^{-3}	3.01×10^{1}	1.29	0.0
Uranium-233	2.76×10^{-4}	3.84×10^{2}	9.63×10^{1}	6.10
Uranium-234	4.28×10^{1}	1.66×10^{2}	2.84×10^{2}	2.65
Uranium-235	2.73×10^{-1}	6.78	2.10	2.15×10^{-5}
Uranium-236	7.12×10^{-1}	4.52	2.64×10^{1}	4.58×10^{-4}
Uranium-237	0.0	0.0	0.0	6.40×10^{-1}
Uranium-238	1.36×10^{-2}	1.50×10^{2}	1.81×10^{2}	0.0
Yttrium-90	6.01×10^{6}	3.06×10^{7}	2.67×10^{8}	2.89×10^{6}
Zirconium-93	0.0	3.62×10^{3}	6.58×10^{2}	2.03×10^{2}

a. The Hanford Site high-level radioactive waste radionuclide inventory represents the radionuclide inventory in 5,325 canisters (DIRS 181377-BSC 2007, Section 7; DIRS 180471-BSC 2007, Table 8).

b. The Idaho National Laboratory high-level radioactive waste radionuclide inventory represents the radionuclide inventory in 550 canisters (DIRS 181377-BSC 2007, Section 7; DIRS 180471-BSC 2007, Table 19).

c. The Savannah River Site high-level radioactive waste radionuclide inventory represents the radionuclide inventory in 3,500 canisters (DIRS 181377-BSC 2007, Section 7; DIRS 180471-BSC 2007, Table 3).

d. The West Valley high-level radioactive waste radionuclide inventory represents the radionuclide inventory in 300 canisters (DIRS 181377-BSC 2007, Section 7; DIRS 180471-BSC 2007, Table 17).

inventory and thereby the radiation dose. The following descriptions are for typical spent nuclear fuel for each group listed in Tables G-11 through G-14.

Group 1: Uranium Metal, Zirconium Alloy Clad, Low-Enriched Uranium. This group contains uranium metal fuel compounds with zirconium alloy cladding. The end-of-life effective enrichment ranges from 0.5 to 1.7 percent. The cladding is in fair to poor condition. This group of fuel comprises approximately 2,103 MTHM.

Group 2: Uranium Metal, Non-Zirconium Alloy Clad, Low-Enriched Uranium. This group contains uranium metal fuel compounds with no known zirconium alloy cladding. The end-of-life effective enrichment ranges from 0.2 to 3.4 percent. The cladding is in good to poor condition. This group of fuel comprises approximately 8 MTHM.

Group 3: Uranium-Zirconium. This group contains uranium-zirconium alloy fuel compounds with zirconium alloy cladding. The end-of-life effective enrichment ranges from 0.5 to 92.9 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 0.66 MTHM.

Group 4: Uranium-Molybdenum. This group contains uranium-molybdenum alloy fuel compounds with various types of cladding. The end-of-life effective enrichment ranges from 2.4 to 25.8 percent. If present, the cladding is in good to poor condition. This group of fuel comprises approximately 3.9 MTHM.

Group 5: Uranium Oxide, Intact Zirconium Alloy Clad, Highly Enriched Uranium. This group contains uranium oxide fuel compounds with intact zirconium alloy cladding. The end-of-life effective enrichment ranges from 23.1 to 92.5 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 1 MTHM.

Group 6: Uranium Oxide, Intact Zirconium Alloy Clad, Medium-Enriched Uranium. This group contains uranium oxide fuel compounds with intact zirconium alloy cladding. The end-of-life effective enrichment ranges from 5.0 to 6.9 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 1.9 MTHM.

Group 7: Uranium Oxide, Intact Zirconium Alloy Clad, Low-Enriched Uranium. This group contains uranium oxide fuel compounds with intact zirconium alloy cladding. The end-of-life effective enrichment ranges from 0.6 to 4.9 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 89.6 MTHM.

Group 8: Uranium Oxide, Intact Stainless-Steel/Hastelloy Clad, Highly Enriched Uranium. This group contains uranium oxide fuel compounds with intact stainless-steel or Hastelloy cladding. The end-of-life effective enrichment ranges from 91.0 to 93.2 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 0.19 MTHM.

Group 9: Uranium Oxide, Intact Stainless-Steel Clad, Medium-Enriched Uranium. This group contains uranium oxide fuel compounds with intact stainless-steel cladding. The end-of-life effective enrichment ranges from 5.5 to 20.0 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 0.69 MTHM.

Group 10: Uranium Oxide, Intact Stainless Steel Clad, Low-Enriched Uranium. This group contains uranium oxide fuel compounds with stainless-steel cladding. The end-of-life effective enrichment ranges from 0.2 to 1.9 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 0.9 MTHM.

Group 11: Uranium Oxide, Non-Intact or Declad Non-Aluminum Clad, Highly Enriched Uranium. This group contains uranium oxide fuel compounds with no known aluminum cladding. The end-of-life effective enrichment ranges from 21.0 to 93.3 percent. If present, the cladding is in poor condition. This group of fuel comprises approximately 0.82 MTHM.

Group 12: Uranium Oxide, Non-Intact or Declad Non-Aluminum Clad, Medium-Enriched Uranium. This group contains uranium oxide fuel compounds with no known aluminum cladding. The end-of-life effective enrichment ranges from 5.2 to 18.6 percent. If present, the cladding is in poor condition. This group of fuel comprises approximately 0.47 MTHM.

Group 13: Uranium Oxide, Non-Intact or Declad Non-Aluminum Clad, Low-Enriched Uranium. This group contains uranium oxide fuel compounds with no known aluminum cladding. The end-of-life effective enrichment ranges from 1.1 to 3.2 percent. If present, the cladding is in poor condition. This group of fuel comprises approximately 82.5 MTHM.

Group 14: Uranium Oxide, Aluminum Clad, Highly Enriched Uranium. This group contains uranium oxide fuel compounds with aluminum cladding. The end-of-life effective enrichment ranges from 58.1 to 89.9 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 4.6 MTHM.

Group 15: Uranium Oxide, Aluminum Clad, Medium-Enriched Uranium and Low-Enriched Uranium. This group contains uranium oxide fuel compounds with aluminum cladding. The end-of-life effective enrichment ranges from 8.9 to 20.0 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 0.29 MTHM.

Group 16: Uranium-Aluminum, Highly Enriched Uranium. This group contains uranium-aluminum alloy fuel compounds with aluminum cladding. The end-of-life effective enrichment ranges from 21.9 to 93.3 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 7.5 MTHM.

Group 17: Uranium-Aluminum, Medium-Enriched Uranium. This group contains uranium-aluminum alloy fuel compounds with aluminum cladding. The end-of-life effective enrichment ranges from 9.0 to 20.0 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 2.6 MTHM.

Group 18: Uranium-Silicide. This group contains uranium-silicide fuel compounds with aluminum cladding. The end-of-life effective enrichment ranges from 5.2 to 22.0 percent. The cladding is in good to poor condition. This group of fuel comprises approximately 7.2 MTHM.

Group 19: Thorium/Uranium Carbide, TRISO or BISO-Coated Particles in Graphite. This group contains thorium/uranium carbide fuel compounds with TRISO (tristructural isotopic)- or BISO (bistructural isotopic)-coated particles. TRISO-coated particles consist of an isotropic pyrocarbon outer layer, a silicon carbide layer, an isotropic carbon layer, and a porous carbon buffer inner layer. BISO-coated particles consist of an isotropic pyrocarbon outer layer and a low density porous carbon buffer inner layer. The end-of-life effective enrichment ranges from 71.4 to 84.4 percent. The coating is in good condition. This group of fuel comprises approximately 24.7 MTHM.

Group 20: Thorium/Uranium Carbide, Mono-Pyrolytic Carbon-Coated Particles in Graphite. This group contains thorium/uranium carbide fuel compounds with mono-pyrolytic carbon-coated particles. The end-of-life effective enrichment ranges from 80.6 to 93.2 percent. The coating is in poor condition. This group of fuel comprises approximately 1.6 MTHM.

Group 21: Plutonium/Uranium Carbide, Nongraphite Clad, Not Sodium Bonded. This group contains plutonium/uranium carbide fuel compounds with stainless-steel cladding. The end-of-life effective enrichment ranges from 1.0 to 67.3 percent. The cladding is in good to poor condition. This group of fuel comprises approximately 0.08 MTHM.

Group 22: Mixed Oxide, Zirconium Alloy Clad. This group contains plutonium/uranium oxide fuel compounds with zirconium alloy cladding. The end-of-life effective enrichment ranges from 1.3 to 21.3 percent. The cladding is in good to poor condition. This group of fuel comprises approximately 1.6 MTHM.

Group 23: Mixed Oxide, Stainless-Steel Clad. This group contains plutonium/uranium and plutonium oxide fuel compounds with stainless-steel cladding. The end-of-life effective enrichment ranges from 2.1 to 87.4 percent. The cladding is in good poor condition. This group of fuel comprises approximately 10.7 MTHM.

Group 24: Mixed Oxide, Non-Stainless Steel/Non-Zirconium Alloy Clad. This group contains plutonium/uranium oxide fuel compounds with no known stainless-steel or zirconium alloy cladding. The end-of-life effective enrichment ranges from 5.0 to 54.3 percent. The cladding is in poor to nonintact condition. This group of fuel comprises approximately 0.11 MTHM.

Group 25: Thorium/Uranium Oxide, Zirconium Alloy Clad. This group contains thorium/uranium oxide fuel compounds with zirconium alloy cladding. The end-of-life effective enrichment ranges from 10.1 to 98.4 percent. The cladding is in good to poor condition. This group of fuel comprises approximately 42.6 MTHM.

Group 26: Thorium/Uranium Oxide, Stainless-Steel Clad. This group contains thorium/uranium oxide fuel compounds with stainless-steel cladding. The end-of-life effective enrichment ranges from 7.6 to 97.8 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 7.6 MTHM.

Group 27: Uranium-Zirconium Hydride, Stainless-Steel/Incoloy Clad, Highly Enriched Uranium. This group contains uranium zirconium hydride fuel compounds with stainless-steel or Incoloy cladding. The end-of-life effective enrichment ranges from 42.5 to 93.2 percent. The cladding is in good to fair condition. This group of fuel comprises approximately 0.16 MTHM.

Group 28: Uranium-Zirconium Hydride, Stainless-Steel/Incoloy Clad, Medium-Enriched Uranium. This group contains uranium zirconium hydride fuel compounds with stainless-steel or Incoloy cladding. The end-of-life effective enrichment ranges from 11.9 to 20.0 percent. The cladding is in good to poor condition. This group of fuel comprises approximately 1.4 MTHM.

Group 29: Uranium-Zirconium Hydride, Aluminum Clad, Medium-Enriched Uranium. This group contains uranium zirconium hydride fuel compounds with aluminum cladding. The end-of-life effective enrichment ranges from 16.8 to 20.0 percent. The cladding is in good condition. This group of fuel comprises approximately 0.35 MTHM.

Group 30: Uranium-Zirconium Hydride, Declad. This group contains uranium zirconium hydride fuel compounds that have been declad. The end-of-life effective enrichment is about 89.7 percent. This group of fuel comprises approximately 0.03 MTHM.

Group 31: Metallic Sodium Bonded. This group contains a wide variety of spent nuclear fuel that has the common attribute of containing metallic sodium bonding between the fuel matrix and the cladding. The end-of-life effective enrichment ranges from 0.1 to 93.2 percent. If present, the cladding is in good to poor condition. This group of fuel comprises approximately 59.9 MTHM. This spent nuclear fuel will be treated and disposed of as high-level radioactive waste.

Group 32: Naval Fuel. Naval nuclear fuel is highly robust and designed to operate in a high-temperature, high-pressure environment for many years. This fuel is highly enriched (93 to 97 percent) in uranium-235. In addition, to ensure that the design will be capable of withstanding battle shock loads, the naval fuel material is surrounded by large amounts of zirconium alloy. This group of fuel comprises approximately 65 MTHM.

Group 33: Canyon Stabilization. This spent nuclear fuel is being treated and will be disposed of as high-level radioactive waste.

Group 34: Miscellaneous. This group contains spent nuclear fuel that does not fit into other groups. The spent nuclear fuel in this group was generated from numerous reactors of different types. The end-of-life effective enrichment ranges from 14.6 to 90.0 percent. If present, the cladding is in good to poor condition. This group of fuel comprises of approximately 0.44 MTHM.

The DOE spent nuclear fuel radionuclide inventories are for the amount of spent nuclear fuel that DOE would ship in rail casks. The DOE spent nuclear fuel radionuclide inventory is based on 752 canisters from the Hanford Site, 1,603 canisters from the Idaho National Laboratory, and 400 canisters from the Savannah River Site. These inventories were compiled from data in DIRS 169354-DOE 2004, Volume II, Appendix C. For naval spent nuclear fuel, the radionuclide inventory is for 400 casks containing 400 canisters. The single cask naval spent fuel inventory was compiled from DIRS 155857-McKenzie 2001, Table 3. Tables G-11 through G-14 list the radionuclide inventories for DOE spent nuclear fuel.

For commercial spent nuclear fuel, the radionuclide inventories are for the amount of spent nuclear fuel that DOE would ship in rail and truck casks. For pressurized-water-reactor spent nuclear fuel, DOE would ship an estimated 93,671 spent nuclear fuel assemblies in rail and truck casks (DIRS 181377-BSC 2007, Section 7). For boiling-water-reactor spent nuclear fuel, the Department would ship 128,105 spent nuclear fuel assemblies in rail and truck casks (DIRS 181377-BSC 2007, Section 7). This analysis assumed that all shipping casks would be full and all trains would have a full complement of casks. This increases the number of spent nuclear fuel assemblies to 94,817 for pressurized-water-reactor spent nuclear fuel and 129,721 for boiling-water-reactor spent nuclear fuel. The representative pressurized-water-reactor assembly would have a burnup of 60,000 megawatt-days per MTHM, an enrichment of 4 percent, and a decay time of 10 years (DIRS 169061-BSC 2004, all). The representative boiling-water-reactor assembly would have a burnup of 50,000 megawatt-days per MTHM, an enrichment of 4 percent, and a decay time of 10 years (DIRS 164364-BSC 2003, all). Table G-15 lists the radionuclide inventory for commercial spent nuclear fuel.

The high-level radioactive waste radionuclide inventory is based on 5,316 canisters from Hanford Site, 528 canisters from Idaho National Laboratory, 3,490 canisters from Savannah River Site, and 277 canisters from West Valley (DIRS 181377-BSC 2007, Section 7). This analysis assumed that all shipping casks that contained high-level radioactive waste would be full and all trains would have a full complement of casks. This increases the amount of high-level radioactive waste to 5,325 canisters for

Hanford Site, 550 canisters for Idaho National Laboratory, 3,500 canisters for Savannah River Site, and 300 canisters from West Valley. Table G-16 lists the radionuclide inventory for high-level radioactive waste.

G.6 Incident-Free Transportation

The impacts from incident-free transportation can be related to either the cargo being carried or to the vehicle that carries the cargo. Incident-free impacts that are related to the cargo are known as radiological impacts. Incident-free impacts that are related to the vehicle are nonradiological in nature and are known as vehicle emission impacts.

G.6.1 RADIOLOGICAL IMPACTS

Radiation doses during normal, incident-free transportation of radioactive materials result from exposure of workers and the public to the external radiation field that surrounds the shipping containers. The radiation dose is a function of the number of people exposed, their proximity to the containers, their length of time of exposure, and the intensity of the radiation field.

In most cases, rail casks would be shipped to the repository using dedicated trains. A dedicated train would consist only of equipment and lading associated with the transportation of spent nuclear fuel and high-level radioactive waste; that is, the train would consist only of necessary motive power, buffer cars, and cask cars, together with a car for escort personnel. Such a train would not transport other rail rolling stock, other revenue, or company freight. For shipments of commercial spent nuclear fuel, there would be three casks that contained spent nuclear fuel per train. For shipments of DOE spent nuclear fuel and high-level radioactive waste, there would be five casks per train. Other numbers of casks per train could be possible for shipments of commercial spent nuclear fuel and DOE spent nuclear fuel and high-level radioactive waste. In both cases, two buffer railcars, two locomotives, and one escort railcar would be in the train. Escorts would be present in all areas (rural, suburban, and urban) for all rail shipments.

Truck casks would be shipped to the repository on overweight trucks. Escorts would be present in all areas (rural, suburban, and urban) for all truck shipments.

DOE determined radiological impacts for members of the public and workers during normal, incident-free transportation of the casks. For members of the public, the Department estimated radiation doses for:

- People within 800 meters (0.5 mile) of the transportation route. The doses to these people are referred to as off-link radiation doses.

- People in vehicles sharing the transportation route. The doses to these people are referred to as on-link radiation doses.

- People exposed at stops that occur en route to the repository. For truck transportation, these would include stops for refueling, food, and rest, and for brief inspections at regular intervals. For rail transportation, stops would occur in rail yards at the beginning of the trip, at the Staging Yard at the end of the trip, and along the route to change crews and equipment. Stops would also include the intermodal transfers of rail casks for shipments from generator sites without direct rail access.

- Workers such as truck drivers, escorts, inspectors, and workers at rail yards or at the Staging Yard at the end of the trip. Engineers and conductors would be in the train locomotives at least 46 meters (150 feet) from the closest rail cask, shielded from radiation exposure by the locomotives; therefore, there would be no radiation doses for these workers en route to the repository. Workers would also be exposed during Commercial Vehicle Safety Alliance truck inspections at the beginning and end of a shipment and during intermodal transfers of rail casks for shipments from generator sites without direct rail access.

G.6.1.1 Collective Radiation Dose Scenarios

Radiation doses received by a population of workers or members of the public are referred to as collective radiation doses. DOE estimated collective radiation doses based on unit risk factors. Unit risk factors provide an estimate of the radiation doses from transport of one shipment or container of radioactive material over a unit distance of travel in a given population density zone.

Unit risk factors can provide an estimate of the radiation dose from one container or shipment being stopped at a location such as a rail yard or the radiation dose from one container or shipment passing a train stopped at a siding. DOE used five types of unit risk factors to estimate collective incident-free radiation doses:

- Unit risk factors to estimate incident-free radiation doses that depended on the number of casks, the population density in each population zone, and the distance in each population zone.

- Unit risk factors to estimate incident-free radiation doses that depended on the number of casks and the distance in each population zone.

- Unit risk factors to estimate incident-free radiation doses that depended on the number of casks and the population density around locations such as a rail yard.

- Unit risk factors to estimate incident-free radiation doses that depended on the number of trains (that is, shipments) and the distance in each population zone.

- Unit risk factors to estimate incident-free radiation doses that depended on the number of casks.

The Yucca Mountain FEIS (DIRS 155970-DOE 2002, p. J-40) contains a more detailed explanation of how DOE used unit risk factors to estimate radiation doses. As in the FEIS (DIRS 155970-DOE 2002, Section J.1.3.2), DOE estimated the unit risk factors using the RADTRAN 5 computer program (DIRS 150898-Neuhauser and Kanipe 2000, all; DIRS 155430-Neuhauser et al. 2000, all) and the RISKIND computer program (DIRS 101483-Yuan et al. 1995, all). Both RADTRAN and RISKIND have been verified and validated for estimating incident-free radiation doses during transportation of radioactive material (DIRS 101845-Maheras and Pippen 1995, all; DIRS 177031-Osborn et al. 2005, all; DIRS 102060-Biwer et al. 1997, all).

The incident-free unit risk factors used in the analysis in this Repository SEIS are similar to those in the Yucca Mountain FEIS (DIRS 157144-Jason Technologies 2001, Tables 4-20 and 4-21) with the following changes:

- The dedicated train exposure factors are used to estimate worker and public exposures during stops at rail yards. One stop would occur at the rail yard closest to the generator site and another at the Staging Yard in Nevada. A stop time of 2 hours was used for these stops. Two-hour stops would also occur every 277 kilometers (170 miles). For shipments using regular freight trains, a 30-hour stop was used to estimate worker and public exposures.

- Escorts would be present in the escort car from the time the train was assembled at the generator site until it reached its final destination at the repository.

- For generator sites without direct rail access, four escort cars would accompany the heavy-haul truck carrying the rail cask. At the point where the rail cask was moved from the heavy-haul truck to the railcar, assembly of the dedicated train would take 30 hours. The escorts would be present for this 30-hour period.

- A train containing commercial spent nuclear fuel would contain three casks. A train containing DOE spent nuclear fuel and high-level radioactive waste would contain five casks. The escorts would be exposed only to radiation from by the cask closest to the escort car. The shielding of this car would effectively shield the escorts from the other casks in the train.

- Unit risk factors were estimated for workers at the Maintenance-of-Way Facility, workers at sidings, and noninvolved workers at the Staging Yard; the Yucca Mountain FEIS did not address these facilities and activities. These unit risk factors are discussed in Appendix K of the Rail Alignment EIS.

As in the Yucca Mountain FEIS, DOE set the external dose rates for the truck and rail casks at their regulatory maximum, 10 millirem per hour at 2 meters (6.6 feet) from the truck trailer or railcar.

G.6.1.2 Maximally Exposed Individual Dose Scenarios

Maximally exposed individuals are hypothetical workers and members of the public who would receive the highest radiation doses. The scenarios DOE used to estimate the radiation doses are similar to the scenarios in the Final Yucca Mountain EIS (DIRS 155970-DOE 2002, Section J.1.3.2.2), and were evaluated on the national level and on the Nevada level. National scenarios incorporate conditions such as speeds, distances, and exposure times that would be representative of exposures across the United States Nevada scenarios incorporate site-specific conditions for exposures in Nevada.

G.6.1.2.1 *National Scenarios*

For workers, DOE evaluated the following scenarios:

- An escort 27 meters (90 feet) from the rail cask. This person would be exposed for 2,000 hours per year. The 27-meter distance includes the length of the buffer railcar between the last rail cask car and the escort car.

- An inspector 1 meter (3.3 feet) from the rail or truck cask for 1 hour per cask. This person would be exposed for 2,000 hours per year (DIRS 155970-DOE 2002, p. J-42).

- A truck driver who would drive shipments that contained loaded casks for 1,000 hours per year and unload casks for 1,000 hours per year.

- A rail yard crew member 10 meters (33 feet) from the rail cask for 2 hours per cask. This person would be exposed for 2,000 hours per year (DIRS 155970-DOE 2002, p. J-42).

For members of the public, DOE evaluated the following scenarios:

- Typically, there is an 18-meter (60-foot) buffer zone around rail lines that is railroad property, within which people cannot build homes. Therefore, DOE estimated the radiation dose to a resident living 18 meters from a rail line. This individual was assumed to be exposed to all loaded rail casks as they passed by en route to the repository.

- A resident 200 meters (660 feet) from a rail yard. This person would be exposed for 2 hours per cask (DIRS 155970-DOE 2002, p. J-42).

- A person stuck in a traffic jam next to the cask for 1 hour. The person would be 1.2 meters (4 feet) from the cask (DIRS 155970-DOE 2002, p. J-42).

- A resident 30 meters (100 feet) from a road or highway. This individual would exposed to all loaded truck casks as they passed by en route to the repository (DIRS 155970-DOE 2002, p. J-42).

- A person at a service station. This person would be exposed for 49 minutes to each truck cask at a distance of 16 meters (52 feet) (DIRS 155970-DOE 2002, p. J-42).

G.6.1.2.2 *Nevada Scenarios*

For workers, DOE evaluated the following scenarios:

- An escort 27 meters (90 feet) from the rail cask. This person would be exposed for 2,000 hours per year. The 27-meter distance includes the length of the buffer railcar between the last rail cask car and the escort car.

- An inspector 1 meter (3.3 feet) from the rail or truck cask for 1 hour per cask. This person would be exposed for 2,000 hours per year.

- A rail yard crew member 10 meters (33 feet) from the rail cask for 2 hours per cask. This person would be exposed for 2,000 hours per year.

For workers, two scenarios that were not addressed in the Yucca Mountain FEIS have been added to the analysis for this Repository SEIS:

- In the first scenario, a worker at the Maintenance-of-Way Facility would be exposed to a loaded cask train traveling 50 kilometers (31 miles) per hour as it passed the facility en route to the repository. This worker would be 60 meters (200 feet) from the cask as it passed.

- In the second scenario, a worker at a siding would be exposed to a loaded rail cask train traveling 50 kilometers (31 miles) per hour as it passed the siding en route to the repository. This worker would be 7.62 meters (25 feet) from the rail cask as it passed.

A separate truck driver scenario was not evaluated in Nevada because the exposure of the driver was based on travel from generator sites to the repository, and there would be no drivers who drove solely in Nevada.

For members of the public, the following scenarios were evaluated:

- Typically, there is an 18-meter (60-foot) buffer zone around rail lines that is railroad property and within which people cannot build homes. Therefore, DOE estimated the radiation dose to a resident living 18 meters from a rail line. This individual was assumed to be exposed to all loaded rail casks as they passed by en route to the repository.

- In some cases, individuals could have access to locations that are closer than 18 meters (60 feet) from a rail line. For example, Nevada Agency for Nuclear Projects (DIRS 158452 Nevada Agency for Nuclear Projects 2002, p. 123) states that in the Las Vegas area, individuals could be 15, 20, 30, 35, 40, 100, and 160 meters (49, 66, 98, 115, 131, 328, and 525 feet) from the rail line. In the area of the Reno Trench, an individual could be as close as 5 meters (16 feet) from the rail line. Therefore, radiation doses were estimated for individuals at these distances from the rail line. These locations were not permanently occupied by residents. However, to provide a conservative estimate of potential impacts, they were assumed to be exposed to all loaded casks that passed through Las Vegas or Reno en route to the repository.

- In Nevada, Interstate Highway 15, the Las Vegas beltway, and U.S. Highway 95 would be used for truck shipments. There are typically buffer zones along Interstate highways and beltways so people cannot build homes much closer then about 30 meters (100 feet) from the road. However, Highway 95 passes through Indian Springs on the way to the repository. In Indian Springs, an individual could reside as close as 24 meters (80 feet) from the highway. Therefore, the radiation dose was estimated for an individual who resided at this location and who was exposed to all loaded truck casks as they passed by en route to the repository.

- A person stuck in a traffic jam next to the cask for 1 hour. The person would be 1.2 meters (4 feet) from the cask.

- A person at a service station. This person would be exposed for 49 minutes to each truck cask at a distance of 16 meters (52 feet) (DIRS 155970-DOE 2002, p. J-42).

- A resident living near the staging yard would be exposed to all loaded casks at the yard for a duration of 2 hours per cask. Table G-17 lists the distances from the staging yard for these residents, which were based on site-specific data around each yard.

G.6.2 VEHICLE EMISSION IMPACTS

The analysis estimated incident-free impacts from vehicle emissions using unit risk factors that account for fatalities associated with emissions of exhaust and fugitive dust in urban, suburban, and rural areas by

Table G-17. Distances to members of the public around staging yards.

Staging yard location	Distance (meters)	Type of location
Caliente-Indian Cove	1,600	Residence
Caliente-Upland	400	Residence
Eccles-North	1,500	Residence
Mina-Hawthorne	660	Business

Note: Conversion factors are on the inside back cover of this Repository SEIS.

transportation vehicles, including escort vehicles. Because the impacts would occur equally for trucks and railcars transporting loaded or unloaded shipping casks, the analysis used round-trip distances. Because escorts were present in all areas, escort vehicle emission impacts were also estimated based on round trips.

For trucks, the vehicle emission unit risk factor was 1.5×10^{-11} fatalities per kilometer per person per square kilometer (DIRS 157144-Jason Technologies 2001, p. 98). For escort vehicles, the vehicle emission unit risk factor was 9.4×10^{-12} fatalities per kilometer per person per square kilometer (DIRS 157144-Jason Technologies 2001, p. 99). For railcars, the vehicle emission unit risk factor was 2.6×10^{-11} fatalities per kilometer per person per square kilometer (DIRS 157144-Jason Technologies 2001, p. 99).

G.7 Transportation Accident Risks

Transportation accident risks can be related either to the cargo being carried or to the vehicle that carries the cargo. Transportation accident risks that are related to the cargo are known as radiological accident risks. Transportation accident risks that are related to the vehicle are nonradiological in nature and are known as transportation accident fatalities.

G.7.1 TRANSPORTATION RADIOLOGICAL ACCIDENT RISKS

The radiological dose risks from transporting spent nuclear fuel and high-level radioactive waste would result from: (1) accidents in which there was no breach of the containment provided by the transportation cask, but there was loss of shielding because of lead shield displacement, (2) accidents in which there was no breach of the containment and no loss of shielding, and (3) accidents that released and dispersed radioactive material from the transportation cask. In this Repository SEIS, the risk to the general public from the radiological consequences of transportation accidents is called dose risk. Dose risk is the sum of the products of the probabilities (dimensionless) and the consequences (in person-rem) of all potential transportation accidents. The probability of a single accident is usually determined by historical information on accidents of a similar type and severity. The consequences are estimated by analysis of the quantity of radionuclides likely to be released, potential exposure pathways, potentially affected population, likely weather conditions, and other information.

Potential accidents range from accidents with higher probabilities and lower consequences to accidents with lower probabilities and higher consequences. The analysis used the following information to determine the risks of accidents:

- The number of shipments;

- The distances and population densities along the transportation routes in rural, suburban, and urban areas;
- The kind and amount of radioactive material that would be transported;
- Transportation accident rates;
- Conditional probabilities of release and the fraction of cask contents that could be released in accidents;
- Conditional probabilities of amounts of lead shielding displacement that could occur during accidents, and the resulting radiation dose rates; and
- Exposure scenarios including inhalation, ingestion, groundshine, resuspension, and immersion pathways, state-specific agricultural factors, and neutral (or average) atmospheric dispersion factors.

As in the incident-free transportation analysis, DOE used the RADTRAN 5 computer program (DIRS 150898-Neuhauser and Kanipe 2000, all; DIRS 155430-Neuhauser et al. 2000, all) to estimate unit risk factors for accidents that involved loss of shielding or when the shielding was undamaged. RADTRAN5 was also used to estimate unit risk factors for accidents that involved the release of radioactive material from the cask, for each radionuclide of concern in spent nuclear fuel and high-level radioactive waste. RADTRAN has been verified and validated for estimating the accident risks from transport of radioactive material (DIRS 101845-Maheras and Pippen 1995, all; DIRS 177031-Osborn et al. 2005, all). The unit risk factors were combined with radionuclide inventories, number of shipments, accident rates, conditional probabilities of release, release fractions, distance, and population densities to determine the dose risk for populations within 80 kilometers (50 miles) of the rail alignment.

The methods and data DOE used to estimate the dose risks were the same as those in the Yucca Mountain FEIS (DIRS 155970-DOE 2002, Section J.1.4.2) with the following exceptions:

- The distances and population densities have been updated,
- The number of rail casks to be shipped has been updated,
- Track Class-specific rail accident rates were used in the analysis,
- Truck accident rates have been updated,
- The radionuclide inventories have been updated through additional data collection and analysis,
- Updated radiation dosimetry has been used to estimate unit risk factors and radiation doses, and
- Updated health risk conversion factors have been used to estimate the number of latent cancer fatalities.

TRACK CLASS

The Federal Railroad Administration's Track Safety Standards, at 49 CFR Part 213, establish track structure and track geometry requirements for nine separate classes of track (Sections 213.9 and 213.307) with designated maximum speeds for each class. Railroads indicate the class to which each track belongs. Once the designation is made, the railroads are held responsible for maintaining each track to specified tolerances for its designated class. A railroad becomes liable for civil penalties if it fails to maintain a track to proper standards, or if it operates trains at speeds in excess of the limits of the designated class.

- The lowest class is referred to as excepted track. Only freight trains are allowed to operate on this type of track, and they may run at speeds up to 10 miles per hour.
- Class 1 track is the lowest class allowing the operation of passenger trains. Freight train speeds are still limited to 10 miles per hour, and passenger trains are restricted to 15 miles per hour.
- Class 2 track limits freight trains to 25 miles per hour and passenger trains to 30 miles per hour.
- Class 3 track limits freight trains to 40 miles per hour and passenger trains to 60 miles per hour.
- Class 4 track limits freight trains to 60 miles per hour and passenger trains to 80 miles per hour. Most through lines, especially owned by the major Class 1 railroads (BNSF, CSX, Norfolk Southern, and Union Pacific), are Class 4 track.
- Class 5 track limits freight trains to 80 miles per hour and passenger trains to 90 miles per hour. The most significant portion of Class 5 track is in the western part of the Union Pacific mainlines, but the top speed on these lines is limited to 70 miles per hour.

In the United States, the regulations for Track Classes 6 through 9 are designed for passenger trains. Any freight cars moved at passenger speeds must meet the dynamic performance standards of passenger equipment. The only such track is Amtrak passenger lines in the Northeast Corridor.

- Class 6 limits freight trains and passenger trains to 110 miles per hour.
- Class 7 limits freight trains and passenger trains to 125 miles per hour. Most of Amtrak's Northeast Corridor is Class 7 track.
- Class 8 limits freight trains and passenger trains to 160 miles per hour. A few small lengths of Amtrak's Northeast Corridor is Class 8 track.
- Class 9 limits freight trains and passenger trains to 200 miles per hour. There is currently no Class 9 track in the United States.

G.7.1.1 Transportation Accident and Fatality Rates

In the Yucca Mountain FEIS, DOE used rail accident rates from the *State-Level Accident Rates of Surface Freight Transportation: A Reexamination* (DIRS 103455-Saricks and Tompkins 1999, all) to estimate radiological transportation risks. These rates were in terms of accidents per railcar kilometers and were based on 68-railcar trains. Because DOE has adopted a policy of using dedicated trains that would contain 8 to 10 cars on average for shipments of commercial spent nuclear fuel and for most DOE spent nuclear fuel and high-level radioactive waste in this Repository SEIS, a combination of rail accident rates based on both train kilometers and railcar kilometers was used to estimate accident risks (Table G-18). These rates were for Track Class 3 and include derailments and collisions (DIRS 180220-Bendixen and Facanha 2007, all). DOE updated rail fatality rates to reflect data from 2000 to 2004 (DIRS 178016-DOT 2005, all). These fatality rates were in terms of fatalities per railcar kilometer.

Table G-18. Track Class 3 rail accident rates.

Train-based accident rate (accidents per train kilometer)	Railcar-based accident rate (accidents per railcar kilometer)
7.5×10^{-7}	1.7×10^{-8}

Source: DIRS 180220-Bendixen and Facanha 2007, all.
Note: Conversion factors are on the inside back cover of this Repository SEIS.

In the Yucca Mountain FEIS, DOE used state-specific accident and fatality rate data for 1994 to 1996 (DIRS 103455-Saricks and Tompkins 1999, all) to estimate transportation impacts. For trucks, the Department obtained accident and fatality rate data it used in the FEIS from the U.S. Department of Transportation, Federal Motor Carrier Safety Administration, Motor Carrier Management Information System. Since completion of the FEIS, the Federal Motor Carrier Safety Administration has evaluated the data in the Information System. For 1994 through 1996, it found that accidents were underreported by about 39 percent and fatalities were underreported by about 36 percent (DIRS 181755-UMTRI 2003, Table 1, p. 4, and Table 2, p. 6). Therefore, in this Repository SEIS, DOE increased the state-specific truck accident and fatality rates from Saricks and Tompkins by factors of 1.64 and 1.57, respectively, to account for the underreporting.

G.7.1.2 Conditional Probabilities and Release Fractions

In this Repository SEIS, DOE spent nuclear fuel is organized into 34 groups based on the fuel compound, fuel matrix, fuel enrichment, fuel cladding material, and fuel cladding condition. Commercial spent nuclear fuel is organized into two groups, pressurized-water reactor and boiling-water reactor spent nuclear fuel. High-level radioactive waste is organized into four groups: that from Idaho National Laboratory, Hanford Site, Savannah River Site, and West Valley. These groups were assigned to a set of 10 conditional probabilities and release fractions known as release fraction groups based on the characteristics and behavior of the spent nuclear fuel or high-level radioactive waste (DIRS 157144-Jason Technologies 2001, Tables 5-24 to 5-27, 5-33, 5-35, 5-39, 5-41, 5-43, 5-45, 5-46, and 5-48). Release fractions were specified for inert gases, volatile constituents such as cesium and ruthenium, particulates, and activation products such as cobalt-60 that were deposited on the exterior surfaces of the spent nuclear fuel (also known as crud).

For loss-of-shielding accidents, the Yucca Mountain FEIS lists unit risk factors for six severity categories (DIRS 155970-DOE 2002, p. J-54, Table J-19). These unit risk factors are used in this analysis.

G.7.1.3 Atmospheric Conditions

Atmospheric conditions would affect the dispersion of radionuclides that could be released from an accident. Because it is not possible to forecast the atmospheric conditions that might exist during an accident, DOE selected neutral weather conditions (Pasquill Stability Class D) for the transportation risk assessments for the Yucca Mountain FEIS and for this Repository SEIS. Neutral weather conditions are typified by moderate wind speeds, vertical mixing in the atmosphere, and good dispersion of atmospheric contaminants. On the basis of observations from National Weather Service surface meteorological stations at 177 locations in the United States, on an annual average, neutral conditions (Pasquill Class C and D) occur 11 percent and 47 percent of the time, respectively. Stable conditions (Pasquill Class E and F) occur 12 percent and 21 percent of the time, respectively. Unstable conditions (Pasquill Class A and B) occur 1 percent and 7 percent of the time, respectively (DIRS 104800-CRWMS M&O 1999, p. 40).

G.7.1.4 Population Density Zones

DOE used three population density zones (urban, rural, and suburban) for the transportation risk assessment. Urban areas were defined as areas with a population density greater than 3,326 people per square kilometer. Rural areas were defined as areas with a population density less than 139 people per square kilometer. Suburban areas were areas with a population density between 139 and 3,326 people per square kilometer. The actual population densities were based on 2000 Census data. In Las Vegas, the population density was modified to include casino guests and casino workers, based on data from the Nevada Agency for Nuclear Projects (DIRS 158452-Nevada Agency for Nuclear Projects 2002, Table 3.8.12). The population densities and radiological impacts were escalated to 2067 using the escalation factors in Table G-6.

G.7.1.5 Exposure Pathways

DOE estimated radiological doses for an individual near the scene of the accident and for populations within 80 kilometers (50 miles) of the accident. Dose calculations considered a variety of exposure pathways, including inhalation and direct exposure (immersion or cloudshine) from the passing cloud, ingestion of contaminated food, direct exposure (groundshine) from radioactivity deposited on the ground, and inhalation of resuspended radioactive particles from the ground (resuspension).

G.7.1.6 Unit Risk Factors and Radiation Dosimetry

As discussed in Section G.7.1, DOE estimated the radiation doses from transportation accidents using unit risk factors. Unit risk factors were estimated using the RADTRAN 5 computer program (DIRS 150898-Neuhauser and Kanipe 2000, all; DIRS 155430-Neuhauser et al. 2000, all) for five pathways: (1) ingestion, (2) inhalation, (3) immersion, (4) resuspension, and (5) groundshine. For transportation accidents, unit risk factors provide estimates of:

- The radiation dose to an average person in a surrounding unit area (for example, a population density of one person per square kilometer) that could result if 1 curie of a specified radionuclide were released.

- The dose to a general population from ingestion of contaminated food from the accidental release of one curie of a specified radionuclide. The unit risk factor includes the assumption that all contaminated food is consumed.

- For transportation accidents in which a portion of a cask's radiation shield was damaged or lost (loss-of-shielding accidents), and for cases in which the cask’s shield could remain intact, unit risk factors provide estimates of the resulting radiation dose to a person in a surrounding unit area after an accident.

DOE used the inhalation and ingestion dose coefficients from DIRS 172935-ICRP (2001, all) and the groundshine and immersion dose coefficients from DIRS 175544-EPA (2002, all) to estimate the unit risk factors. These dose coefficients are based on the recommendations by International Commission on Radiological Protection Publication 60 (DIRS 101836-ICRP 1991, all) and incorporate the dose coefficients from International Commission on Radiological Protection Publication 72 (DIRS 152446-ICRP 1996, all). For each radionuclide, the dose coefficients DOE used to estimate the unit risk factors, which are listed in DIRS 176975-BMI (2006, Table 5), include radioactive progeny (DIRS 176975-BMI 2006, Table 2). This table also lists the lung absorption type and the value for the fractional absorption to blood from the small intestine (f_1) for each radionuclide.

G.8 Severe Transportation Accidents

In addition to analyzing the radiological and nonradiological risks of transporting spent nuclear fuel and high-level radioactive waste, DOE assessed the consequences of severe transportation accidents; such accidents with a frequency of about 1×10^{-7} per year are known as maximum reasonably foreseeable transportation accidents. According to DOE guidance, accidents that have a frequency of less then 1×10^{-7} rarely need to be examined (DIRS 172283-DOE 2002, p. 9).

The analysis was based on the 20 rail accident severity categories identified in Sprung et al. (DIRS 152476-Sprung et al. 2000, pp. 7-73 and 7-76). The following list describes these severity categories:

- Case 20: Case 20 is a long-duration (many hours), high-temperature fire that would engulf a cask.

- Cases 19, 18, 17, and 16: Case 19 is a high-speed [more than 190 kilometers (120 miles) per hour] impact into a hard object such as a train locomotive severe enough to cause failure of cask seals and puncture through the cask's shield wall. The impact would be followed by a very-long-duration, high-temperature engulfing fire. Cases 18, 17, and 16 are accidents that would also involve very-long-duration fires, failures of cask seals, and puncture of cask walls. However, these accidents would be progressively less severe in terms of impact speeds. The impact speeds range from 145 to 190 kilometers (90 to 120 miles) per hour for Case 18, 97 to 145 kilometers (60 to 90 miles) per hour for Case 17, and 48 to 97 kilometers (30 to 60 miles) per hour for Case 16.

- Cases 15, 12, 9, and 6: Case 15 is a high-speed [more than 190 kilometers (120 miles) per hour] impact into a hard surface such as granite severe enough to cause failure of cask seals. The impact would be followed by a long-duration, high-temperature engulfing fire. Cases 12, 9, and 6 are also accidents that would involve long-duration fires and failures of cask seals. However, these accidents would be progressively less severe in terms of impact speeds ranging from 145 to 190 kilometers (90 to 120 miles) per hour for Case 12, 97 to 145 kilometers (60 to 90 miles) per hour for Case 9, and 48 to 97 kilometers (30 to 60 miles) per hour for Case 6.

- Cases 14, 11, 8, and 5: Case 14 is a high-speed [more than 190 kilometers (120 miles) per hour] impact into a hard surface such as granite severe enough to cause failure of cask seals. The impact would be followed by a high-temperature engulfing fire that burned for hours. Cases 11, 8, and 5 are also accidents that would involve fires that would burn for hours and failures of cask seals. However, these accidents would be progressively less severe in terms of impact speeds ranging from 145 to 190 kilometers (90 to 120 miles) per hour for Case 11, 97 to 145 kilometers (60 to 90 miles) per hour for Case 8, and 48 to 97 kilometers (30 to 60 miles) per hour for Case 5.

- Cases 13, 10, 7, and 4: Case 13 is a high-speed [more than 190 kilometers (120 miles) per hour] impact into a hard surface such as granite severe enough to cause failure of cask seals. The impact would be followed by an engulfing fire lasting more than 0.5 hour to a few hours. Cases 10, 7, and 4 are accidents that would involve long-duration fires and failures of cask seals. However, these accidents are progressively less severe in terms of impact speeds ranging from 145 to 190 kilometers

(90 to 120 miles) per hour for Case 10, 97 to 145 kilometers (60 to 90 miles) per hour for Case 7, and 48 to 97 kilometers (30 to 60 miles) per hour for Case 4.

- Cases 3, 2, and 1: Case 3 is a high-speed [more than 190 kilometers (120 miles) per hour] impact into a hard surface such as granite severe enough to cause failure of cask seals with no fire. Cases 2 and 1 are also accidents that would not involve fire but would have progressively lower impact speeds, 145 to 190 kilometers (90 to 120 miles) per hour for Case 2 and 97 to 145 kilometers (60 to 90 miles) per hour for Case 1.

Each of the 20 accident cases above has an associated conditional probability of occurrence (DIRS 152746-Sprung et al. 2000, p. 7-76). These conditional probabilities were combined with the distances along the transportation routes presented in Section G.3, the shipment data presented in section G.4, and the accident rates discussed in Section G.7.1.1 to estimate the frequency of occurrence for each accident case. These frequencies are listed in Table G-19. Cases 1, 4, and 20 have frequencies greater than 1×10^{-7} per year. Based on the results presented in Table J-22 of the Yucca Mountain FEIS (DIRS 155970-DOE 2002), Case 20 is estimated to have the highest consequences of these 3 accident cases. Therefore, Case 20 is considered to be the maximum reasonably foreseeable transportation accident.

Table G-19. Annual frequencies for accident severity cases.

Accident severity case	Annual frequency (accidents per year)
1	8×10^{-7}
2	$5 \times 10^{-8} – 6 \times 10^{-8}$
3	$4 \times 10^{-10} – 5 \times 10^{-10}$
4	3×10^{-6}
5	8×10^{-8}
6	1×10^{-8}
7	7×10^{-9}
8	2×10^{-10}
9	$2 \times 10^{-11} – 3 \times 10^{-11}$
10	5×10^{-10}
11	1×10^{-11}
12	2×10^{-12}
13	4×10^{-12}
14	1×10^{-13}
15	1×10^{-14}
16	4×10^{-11}
17	$2 \times 10^{-14} – 3 \times 10^{-14}$
18	2×10^{-15}
19	1×10^{-17}
20	5×10^{-6}

Based on the analysis in the Yucca Mountain FEIS, accidents that would involve truck casks yielded lower consequences than accidents that would involve rail casks (DIRS 155970-DOE 2002, Tables J-22 and J-23). Therefore, DOE did not update severe accidents involving truck casks in this Repository SEIS.

DOE used the following assumptions to estimate the consequences of these accidents (DIRS 157144-Jason Technologies 2001, Section 5.3.3.3):

- A release height of the plume of 10 meters (33 feet) for fire- and impact-related accidents. In the case of an accident with a fire, a 10-meter release height with no plume rise from the buoyancy of the plume due to fire conditions would yield higher estimates of consequences than accounting for the buoyancy of the plume from the fire (DIRS 157144-Jason Technologies 2001, p. 176).

- A breathing rate for individuals of 10,400 cubic meters (367,000 cubic feet) per year. DOE estimated this breathing rate from data in International Commission on Radiological Protection Publication 23 (DIRS 101074-ICRP 1975, p. 346).

- The release from a severe accident would include only respirable material (DIRS 157144-Jason Technologies 2001, p. 177). The deposition velocity for respirable material would be 0.01 meter per second (0.022 mile per hour).

- A short-term exposure time to airborne contaminants of 2 hours.

- A long-term exposure time to contamination deposited on the ground of 1 year, with no interdiction or cleanup.

- In the Yucca Mountain FEIS, DOE used two sets of atmospheric conditions, neutral atmospheric conditions and moderate winds speeds, and stable atmospheric conditions and low wind speeds to determine consequences from severe accidents. Stable atmospheric conditions and low wind speeds yielded higher consequences than neutral atmospheric conditions and moderate wind speeds. Therefore, in this Repository SEIS, DOE used low wind speeds and stable atmospheric conditions [a wind speed of 0.89 meter per second (2 miles per hour) and Class F stability] to determine consequences. The atmospheric concentrations estimated from these atmospheric conditions would be exceeded only 5 percent of the time.

- The spent nuclear fuel assembly would have a burnup of 60,000 megawatt-days per MTHM, an enrichment of 4 percent, and a decay time of 10 years (DIRS 169061-BSC 2004, all). Table G-15 lists the radionuclide inventory for a single spent nuclear fuel assembly.

DOE used the RISKIND computer code (DIRS 101483-Yuan et al. 1995, all) to estimate radiation doses for the inhalation, groundshine, immersion, and resuspension pathways. RISKIND has been verified and validated for estimating radiation doses from transportation accidents involving radioactive material (DIRS 101845-Maheras and Pippen 1995, all; DIRS 102060-Biwer et al. 1997, all). In addition, DOE used the inhalation dose coefficients from DIRS 172935-ICRP (2001, all) and the groundshine and immersion dose coefficients from DIRS 175544-EPA (2002, all) to estimate radiation doses.

The analysis assumed that the severe transportation accidents could occur anywhere. Population densities in rural areas range from 0 to 139 people per square kilometer. DOE based the analysis in the rural area on a population density of 6 people per square kilometer, which is a representative population density for a rural area (DIRS 101892-NRC 1977, p. E-2). The Department estimated the population density in an urban area by identifying the 20 urban areas in the United States with the largest populations using 2000 Census data, determining the population density in successive annular rings around the center of each urban area, escalating these population densities to 2067, and averaging the population densities in each successive annular ring. Based on 2000 Census data, Las Vegas was not among the 20 largest urban areas in the United States. However, because of proximity of Las Vegas to the repository, DOE included it in the population density analysis. The resident population in Las Vegas was modified to include casino guests and casino workers. Table G-20 lists the population densities.

Table G-20. Population density in urban area.

Annular distance (kilometers)	Population density (people per square kilometer)
0 to 8.05 (0 to 5 miles)	5,012
8.05 to 16.09 (5 to 10 miles)	2,956
16.09 to 24.14 (10 to 15 miles)	2,112
24.14 to 32.19 (15 to 20 miles)	1,342
32.19 to 40.23 (20 to 25 miles)	899
40.23 to 80.47 (25 to 50 miles)	390

Note: Population densities have been escalated to 2067.

G.9 Transportation Sabotage

DOE used the following assumptions to estimate the consequences of transportation sabotage events (DIRS 157144-Jason Technologies 2001, Section 5.3.4.2):

- A breathing rate for individuals of 10,400 cubic meters (367,000 cubic feet) per year. This breathing rate was estimated from data in International Commission on Radiological Protection Publication 23 (DIRS 101074-ICRP 1975, p. 346).

- A short-term exposure time to airborne contaminants of 2 hours.

- A long-term exposure time to contamination deposited on the ground of 1 year, with no interdiction or cleanup.

- In the Yucca Mountain FEIS, DOE used neutral atmospheric conditions and moderate wind speeds to determine the consequences of sabotage events. In this Repository SEIS, DOE used moderate wind speeds and neutral atmospheric conditions [a wind speed of 4.47 meters per second (10 miles per hour) and Class D stability] to determine the consequences of sabotage.

- The release from a sabotage event would include respirable and nonrespirable material. The deposition velocity for respirable material would be 0.01 meter per second (0.022 mile per hour) and the deposition velocity for nonrespirable material would be 0.1 meter per second (0.22 mile per hour).

The DOE analysis assumed that in the sabotage event there would be an initial explosive release that involved releases of radioactive material at varying release heights. For 4 percent of the release, the analysis estimated a release height of 1 meter (3.3 feet); for 16 percent of the release, it estimated a release height of 16 meters (52 feet); for 25 percent of the release, it estimated a release height of 32 meters (105 feet); for 35 percent of the release, it estimated a release height of 48 meters (160 feet); and for 20 percent of the release, it estimated a release height of 64 meters (210 feet) (DIRS 157144-Jason Technologies 2001, p. 189).

In the Yucca Mountain FEIS, DOE used the release fraction data in Luna et al. (DIRS 104918-Luna et al. 1999, all) to evaluate the consequences of sabotage events. For truck and rail casks, a successful sabotage attempt that used the device called "high energy density device one" yielded the largest radiation doses. In this Repository SEIS, the Department used release fractions from Luna (DIRS 181279-Luna 2006, all) to estimate the impacts of such acts that involved spent nuclear fuel in truck or rail casks. The release fractions in Luna (DIRS 181279-Luna 2006, all) are based on the release fractions in Luna et al. (DIRS 104918-Luna et al. 1999, all), but they incorporate data from additional tests sponsored by *Gesellschaft für Anlagen - und Reaktorsicherheit* in Germany and conducted in France in 1994 that were not available for the 1999 report. These tests used pressurized fuel pins and provided a means to assess the effects of aerosol blowdown from pin plenum gas release. The use of these additional test data suggest that DOE overstated the consequences in the FEIS by a factor of 2.5 to 12.

For rail casks, the release fractions in Luna (DIRS 181279-Luna 2006, all) and Luna et al. (DIRS 104918-Luna et al 1999, all) were based on a rail cask that would hold 26 pressurized-water reactor spent nuclear fuel assemblies. DOE plans to operate the repository using a primarily canistered approach that calls for packaging most commercial spent nuclear fuel in TAD canisters, which would hold 21 pressurized-water reactor spent nuclear fuel assemblies. In this Repository SEIS, DOE chose to estimate the consequences of a rail sabotage event based on the radionuclide inventory in 26 pressurized-water reactor spent nuclear fuel assemblies, which overestimated consequences by about 24 percent in comparison to the inventory in 21 pressurized-water reactor spent nuclear fuel assemblies. For truck casks, the sabotage scenario involved a single truck cask that contained four pressurized-water reactor spent nuclear fuel assemblies. Table G-15 lists the radionuclide inventory for a single pressurized-water reactor spent nuclear fuel assembly.

DOE used the RISKIND computer code (DIRS 101483-Yuan et al. 1995, all) to estimate radiation doses for the inhalation, groundshine, immersion, and resuspension pathways. The analysis assumed that the transportation sabotage event could occur anywhere, either in rural or urban areas, using the same population densities as those in the severe accident analysis in Section G.8.

G.10 Transportation Topical Areas

This section discusses topics identified by the public during the scoping process for this Repository SEIS and the Rail Alignment EIS.

G.10.1 ACCIDENTS INVOLVING HAZARDOUS CHEMICALS

DOE would use dedicated trains to ship most spent nuclear fuel and high-level radioactive waste, and hazardous chemical cargos would not be on the same train as spent nuclear fuel or high-level radioactive waste. This would greatly reduce the potential for accidents involving the spent nuclear fuel or high-level radioactive waste and hazardous chemicals.

G.10.2 CRITICALITY DURING ACCIDENTS

Criticality is the term used to describe an uncontrolled nuclear chain reaction. NRC regulations in 10 CFR Part 71 require that the casks used to ship spent nuclear fuel and high-level radioactive waste be able to survive accident conditions, such as immersion in water, without undergoing a criticality. To meet this requirement, casks are typically designed such that, even if water filled the cask and the cask contained unirradiated nuclear fuel (the most reactive case from the perspective of a criticality), a criticality would not occur.

G.10.3 AIRCRAFT CRASH

An aircraft crash into a spent nuclear fuel or high-level radioactive waste cask would be extremely unlikely because the probability of a crash into such a relatively small object, whether stationary or moving, is extremely remote. Nevertheless, the Yucca Mountain FEIS analyzed the consequences of an accident in which a large commercial aircraft or a military aircraft is hypothesized to impact directly onto a cask (DIRS 155970-DOE 2002, Section J.3.3.1). The analysis showed that the penetrating force of a jet engine's center shaft would not breach the heavy shield wall of a cask. With the exception of engines, the relatively light structures of an aircraft would be much less capable of causing damage to a cask. A resulting fire would not be sustainable or able to engulf a cask long enough to breach its integrity.

The *Renewal of the Nellis Air Force Range Land Withdrawal: Legislative Environmental Impact Statement* (DIRS 103472-USAF 1999, all), and the *Final Environmental Impact Statement, Withdrawal of Public Lands for Range Safety and Training Purposes, Naval Air Station Fallon, Nevada* (DIRS 148199-USN 1998, all) discussed system malfunctions or material failures that could result in either an accidental release of ordnance or release of a practice weapon. The *Special Nevada Report* (DIRS 153277-SAIC 1991, all) stated that the probability of dropped ordnance that resulted in injury, death, or property damage ranges from about 1 in 1 billion to 1 in 1 trillion per dropped ordnance incident, with an average of about 1 in 10 billion per incident. Less than one accidentally dropped ordnance incident is estimated per year for all flight operations over the Nevada Test and Training Range and Naval Air Station Fallon. Spent nuclear fuel transportation would not affect the risk from dropped ordnance or aircraft crashes. Therefore, this Repository SEIS does not evaluate radiological consequences of an impact of accidentally dropped ordnance on a shipping cask because the probability of such an event (about 1 in 10 billion per year) is not reasonably foreseeable. Therefore, DOE believes there would be no need for associated mitigation measures and no impacts on military operations.

G.10.4 BALTIMORE TUNNEL FIRE

On July 18, 2001, a freight train carrying hazardous (nonnuclear) materials derailed and caught fire while passing through the Howard Street railroad tunnel in downtown Baltimore, Maryland. The NRC evaluated possible impacts of this fire in *Spent Nuclear Fuel Transportation Package Response to the Baltimore Tunnel Fire Scenario* (DIRS 182014-Adkins et al. 2006, all).

This study evaluated the response of three transportation casks—the HOLTEC Model No. HI-STAR 100, the TransNuclear Model No. TN-68, and the Nuclear Assurance Corporation Legal Weight Truck—to the conditions that existed during the fire. The study concluded that larger transportation packages that resembled the HI-STAR 100 and TN-68 would withstand a fire with thermal conditions similar to those that existed in the Baltimore tunnel fire event with only minor damage to peripheral components. This would be due to their sizable thermal inertia and design specifications in compliance with currently imposed regulatory requirements.

For the TN-68 and the Nuclear Assurance Corporation Legal Weight Truck casks, the maximum temperatures predicted in the regions of the lid and the vent and drain ports exceed the seals' rated service temperatures, making it possible for a small release to occur due to crud that might spall off the surfaces of the fuel rods. While a release is not expected for these conditions, any release that could occur would be very small due to the following factors: (1) the tight clearances maintained between the lid and cask body by the closure bolts, (2) the low pressure differential between the cask interior and exterior, (3) the tendency of such small clearances to plug, and (4) the tendency of crud particles to settle or plate out.

The NRC study also evaluated the radiological consequences of the package responses to the Baltimore tunnel fire. The analysis indicated that the regulatory dose rate limits specified in 10 CFR 71.51 for accident conditions would not be exceeded by releases or direct radiation from any of these packages in this fire scenario. All three packages are designed to maintain regulatory dose rate limits even with a complete loss of neutron shielding. While highly unlikely, the Nuclear Assurance Corporation Legal Weight Truck cask could experience some decrease in gamma shielding due to slump in the lead as a consequence of this fire scenario, but a conservative analysis showed that the regulatory dose rate limits would not be exceeded.

The results of this evaluation strongly indicate that neither spent nuclear fuel particles nor fission products would be released from a spent fuel shipping cask carrying intact spent nuclear fuel involved in a severe tunnel fire such as the Baltimore Tunnel Fire. None of the three cask designs analyzed for the Baltimore Tunnel fire scenario (TN-68, HI-STAR 100, and Nuclear Assurance Corporation Legal Weight Truck) experienced internal temperatures that would result in rupture of the fuel cladding. Therefore, radioactive material (spent nuclear fuel particles or fission products) would be retained in the fuel rods.

There would be no release from the HI-STAR 100 because the inner welded canister would remain leak tight. While a release is unlikely, the potential releases calculated for the TN-68 rail cask and the Legal Weight Truck cask indicated that any release of crud from either cask would be very small—less than 5 rem.

The NRC also evaluated the response of the Nuclear Assurance Corporation Legal Weight Truck cask to the conditions present during the Caldecott Tunnel fire in *Spent Fuel Transportation Package Response to the Caldecott Tunnel Fire Scenario* (DIRS 181841-Adkins et al. 2007, all). This fire took place on April 7, 1982, when a tank truck and trailer carrying 8,800 gallons of gasoline was involved in an accident in the Caldecott Tunnel on State Route 24 near Oakland, California. The trailer overturned and subsequently caught fire. This event is one of the most severe of the five major highway tunnel fires involving shipments of hazardous material that have occurred world-wide since 1949.

This study concluded that small transportation casks similar to the Nuclear Assurance Corporation Legal Weight Truck cask would probably experience degradation of some seals in this severe accident scenario. The maximum temperatures predicted in the regions of the cask lid and the vent and drain ports exceed the rated service temperature of the tetrafluoroethylene or Viton seals, making it possible for a small release to occur due to crud that could spall off the surfaces of the fuel rods. However, any release is expected to be very small due to a number of factors: (1) the metallic lid seal does not exceed its rated service temperature and therefore can be assumed to remain intact, (2) the tight clearances maintained by the lid closure bolts, (3) the low pressure differential between the cask interior and exterior, (4) the tendency for solid particles to plug small clearance gaps and narrow convoluted flow paths such as the vent and drain ports, and (5) the tendency of crud particles to settle or plate out and, therefore, not be available for release.

The NRC study also evaluated the radiological consequences of the package response to the Caldecott Tunnel fire. The results of this evaluation strongly indicate that neither spent nuclear fuel particles nor fission products would be released from a spent fuel shipping cask involved in a severe tunnel fire such as the Caldecott Tunnel fire. The Nuclear Assurance Corporation Legal Weight Truck cask design analyzed for the Caldecott Tunnel fire scenario does not reach internal temperatures that could result in rupture of the fuel cladding. Therefore, radioactive material (spent nuclear fuel particles or fission products) would be retained in the fuel rods. The potential release calculated for the Legal Weight Truck cask in this scenario indicates that any release of crud from the cask would be very small—less than 5 rem.

G.10.5 CASK RECOVERY

The recovery of rail casks loaded with spent nuclear fuel or high-level radioactive waste would use methods commonly used to recover railcars and locomotives following accidents. The capability to lift such weights exists and would be deployed as required. Railroads use emergency response contractors with the ability to lift derailed locomotives that could weigh as much as 136 metric tons (150 tons). Difficult recoveries of equipment as heavy as spent nuclear fuel casks have occurred and DOE anticipates that, if such a recovery was necessary, it would use methods and equipment similar to those used in prior difficult recoveries.

G.10.6 HUMAN ERROR AND TRANSPORTATION ACCIDENTS

The conditional probabilities and release fractions discussed in Section G.7.1.2 would be mostly a direct consequence of error on the part of transport vehicle operators, operators of other vehicles, or persons who maintained vehicles and rights-of-way. The number and severity of the accidents would be minimized through the use of trained and qualified personnel.

Others have argued that other types of human error could contribute to accident consequences: (1) undetected error in the design and certification of transportation packaging (casks) used to ship radioactive material, (2) hidden or undetected defects in the manufacture of these packages, and (3) error in the preparation of the packages for shipment. DOE has concluded that regulations and regulatory practices of the NRC and the U.S. Department of Transportation address the design, manufacture, and use of transportation packaging and are effective in the prevention of these kinds of human error by requiring:

- Independent NRC review of designs to ensure compliance with requirements (10 CFR Part 71), and

- NRC-approved and -audited quality assurance programs for design, manufacturing, and use of transportation packages.

In addition, federal provisions (10 CFR Part 21) provide additional assurance of timely and effective actions to identify and initiate corrective actions for undetected design or manufacturing defects. Further, conservatism in the approach to safety in the regulatory requirements and practices provides confidence that design or manufacturing defects that might remain undetected or operational deficiencies would not lead to a meaningful reduction in the performance of a package under normal or accident conditions of transportation.

G.10.7 COST OF CLEANUP

According to the NRC report *Reexamination of Spent Fuel Shipment Risk Estimates* (DIRS 152476-Sprung et al. 2000, pp. 7 to 76), in more than 99.99 percent of accidents radioactive material would not be released from the cask. After initial safety precautions had been taken, the cask would be recovered and removed from the accident scene. Because no radioactive material would be released, based on reported experience with two previous accidents (DIRS 156110-FEMA 2000, Appendix G, Case 4 and Case 5), the economic costs of these accidents would be minimal.

For the 0.01 percent of accidents severe enough to cause a release of radioactive material from a cask, a number of interrelated factors would affect costs of cleaning up the resulting radioactive contamination after the accident: the severity of the accident and the initial level of contamination; the weather at the time and following; the location and size of the affected land area and the use of the land; the established standard for the allowable level of residual contamination following cleanup and the decontamination method used; and the technical requirements and location for disposal of contaminated materials.

Because it would be necessary to specify each of the factors to estimate cleanup costs, an estimate for a single accident would be highly uncertain and speculative. Nevertheless, to provide a gauge of the costs that could occur DOE examined past studies of costs of cleanup following hypothetical accidents that would involve uncontrolled releases of radioactive materials.

An NRC study of the impacts of transporting radioactive materials in 1977 estimated that costs could range from about $1 million to $100 million for a transportation accident that involved a 600-curie release of a long-lived radionuclide (DIRS 101892-NRC 1977, Table 5-11). These estimates would be about 3 times higher if escalated for inflation from 1977 to the present. In 1980, Finley et al. (DIRS 155054-Finley et al. 1980, Table 6-9) estimated that costs could range from about $90 million to $2 billion for a severe spent nuclear fuel transportation accident in an urban area. Sandquist et al. (DIRS 154814-Sandquist et al. 1985, Table 3-7) estimated that costs could range from about $200,000 to $620 million. In this study, Sandquist et al. estimated that contamination would affect between 0.063 to 4.3 square kilometers (16 to 1,100 acres). A study by Chanin and Murfin (DIRS 152083-Chanin and Murfin 1996, Chapter 6) estimated the costs of cleanup following a transportation accident in which plutonium was dispersed. This study developed cost estimates for cleaning up and remediating farmland, urban areas, rangeland, and forests. The estimates ranged from $38 million to $400 million per square kilometer that would need cleanup. In addition, the study evaluated the costs of expedited cleanups in urban areas for light, moderate, and heavy contamination levels. These estimates ranged from $89 million to $400 million per square kilometer.

The National Aeronautics and Space Administration studied potential accidents for the Cassini mission, which used a plutonium powered electricity generator. The Administration estimated costs of cleaning up radioactive material contamination on land following potential launch and reentry accidents. The estimate for the cost following a launch accident ranged from $7 million to $70 million (DIRS 155551-NASA 1995, Chapter 4) with an estimated contaminated land area of about 1.4 square kilometers (350 acres). The Administration assumed cleanup costs would be $5 million per square kilometer if removal and disposal of contaminated soil were not required and $50 million per square kilometer if those activities were required. For a reentry accident that occurred over land, the study estimated that the contaminated area could range from about 1,500 to 5,700 square kilometers (370,000 to 1.4 million acres) (DIRS 155551-NASA 1995, Chapter 4) with cleanup costs possibly exceeding a total of $10 billion. In a more recent study of potential consequences of accidents that could involve the Cassini mission, the Administration estimated that costs could range from $7.5 million to $1 billion (DIRS 155550-NASA 1997, Chapter 4). The contaminated land area associated with these costs ranged from 1.5 to 20 square kilometers (370 to 4,900 acres). As in the 1995 study, these estimates were based on cleanup costs in the range of $5 million to $50 million per square kilometer.

Using only the estimates provided by these studies, the costs of cleanup following a severe transportation accident that involved spent nuclear fuel in which radioactive material was released could be in the range from $300,000 (after adjusting for inflation from 1985 to the present) to $10 billion. Among the reasons for this wide range are different assumptions about the factors that must be considered: (1) the severity of the assumed accident and resulting contamination levels, (2) accident location and use of affected land areas, (3) meteorological conditions, (4) cleanup levels and decontamination methods, and (5) disposal of contaminated materials. However, the extreme high estimates of costs are based on assumptions that all factors combine in the most disadvantageous way to create a worst case. Such worst cases are not reasonably foreseeable. Conversely, estimates as low as $300,000 might not be realistic for all of the direct and indirect costs of cleaning up following an accident severe enough to cause a release of radioactive materials.

To gauge the range of costs that it could expect for severe accidents during the transport of spent nuclear fuel to a Yucca Mountain repository, DOE considered the amount of radioactive material that could be released in the maximum reasonably foreseeable accident and compared this to the estimates of releases used in the studies discussed above. The maximum reasonably foreseeable accident would release about 30 curies (mostly cesium). This is about 50 times less than the release used by Sandquist et al. (DIRS 154814-Sandquist et al. 1985, all) (1,630 curies) and 20 times less than the release used in the estimates provided by the NRC in 1977 (600 curies). The estimated frequency for an accident this severe to occur is about 6 or 7 times in 10 million years. Based on the prior studies (in which estimated releases exceeded those estimated in this appendix for a maximum reasonably foreseeable accident) and the amount of radioactive material that could be released in a maximum reasonably foreseeable accident, DOE believes that the cost of cleaning up following such an accident could be a few million dollars. Nonetheless, as stated above, the Department also believes that estimates of such costs contain great uncertainty and are speculative; they could be less or 10 times greater, depending on the contributing factors.

For perspective, the current insured limit of responsibility for an accident that involves releases of radioactive materials to the environment is $10.26 billion (Appendix H).

OPPOSING VIEW: COSTS OF CLEANUP

The State of Nevada has provided analyses that assert that the costs of cleanup could be much higher than the estimates discussed in this Repository SEIS, up to $189.7 billion for accidents that involved rail casks (DIRS 181756-Lamb et al. 2001, p. 48) and up to $299.4 billion for sabotage that involved a rail cask (DIRS 181892-Lamb et al. 2002, p. 15).

DOE believes that these extremely high estimates of costs are based on assumptions that all factors combine in the most disadvantageous way to create a worst case. Such worst cases are not reasonably foreseeable.

G.10.8 UNIQUE LOCAL CONDITIONS

Scoping comments on this Repository SEIS stated the unique local conditions in Nevada require special consideration in the transportation accident analysis. In this SEIS, DOE analyzed a range of accidents that reflect the range of reasonably foreseeable real-life conditions. Real-life conditions that would involve various types of collisions, various natural disasters, specific locations (such as mountain passes), or various infrastructure accidents (such as track failure) in effect constitute a combination of cask failure mechanisms, impact velocities, and temperature ranges, which the EIS does evaluate. Because it is impossible to predict what real-life conditions might be involved in accidents that could occur, and to ensure that the analysis accounts for all reasonably foreseeable real-life conditions, DOE has described the maximum reasonably foreseeable accident in terms of cask failure mechanisms and accident forces. Accident scenarios are modeled in this fashion to accommodate the almost infinite number of variables that any given accident could involve.

G.10.9 COMPREHENSIVE RISK ASSESSMENT

The State of Nevada recommended that DOE should use comprehensive risk assessment as a substitute for probabilistic risk assessment in the transportation analysis. According to the state, comprehensive risk assessment calculates probabilities only if there are existing data, theories, and models to support use of rigorous quantitative methods, and uses sensitivity analysis to illustrate impacts of differing assumptions and variations in the quality of data.

Probabilistic risk assessment has been and continues to be the standard tool used for transportation risk assessments since the NRC published the *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes* in 1977 (DIRS 101892-NRC 1977, all). DOE used probabilistic risk assessment to estimate transportation impacts in this Repository SEIS because there are adequate data, methods, and computer programs that make it a valid, state-of-the-art tool to estimate transportation impacts. In addition, DOE has performed sensitivity analyses related to transportation impacts; these analyses are discussed in Appendix A.

G.10.10 BARGE SHIPMENTS

DOE evaluated the impacts of barge shipments of spent nuclear fuel in the Yucca Mountain FEIS (DIRS 155970-DOE 2002, Section J.2.4) for those generator sites without direct rail access but with barge access. The impacts of the use of barges to ship spent nuclear fuel from the generator sites with direct rail access were similar to the those of using heavy-haul trucks to ship from the generator sites without direct rail access for the mostly rail scenario (DIRS 155970-DOE 2002, Tables J-29, J-30, and J-32). The estimated exposed population along the barge routes analyzed in the FEIS would be 502,132 people (DIRS 157144-Jason Technologies 2001, Table 3-10).

For this Repository SEIS, DOE used the TRAGIS computer program to reevaluate the representative routes that could be used for barge shipments of spent nuclear fuel (DIRS 181276-Johnson and Michelhaugh 2003, all). Table G-21 lists the sites, the locations of the intermodal transfer between the barge and the railroad, the lengths of the barge route, and the exposed populations along the barge route. In some cases, DOE evaluated multiple locations for the intermodal transfer.

For the 15 generator sites without direct rail access but with barge access listed in Table G-21, the estimated exposed population along the barge routes would range from 199,193 to 418,945 people. This exposed population would be less than or similar to the exposed population estimated in the Yucca Mountain FEIS. The locations of the intermodal transfer between the barge and the railroad were similar to the locations analyzed in the FEIS (DIRS 155970-DOE 2002, Table J-27) and the distances were similar to the distances estimated in the FEIS (DIRS 155970-DOE 2002, Table J-26). Because the exposed populations, distances, and intermodal transfer locations were similar to the exposed populations,

Table G-21. Data used in reevaluation of barge shipments.

Site	Distance (kilometers)	Exposed population	Barge port assumed for barge-to-rail intermodal transfer
Browns Ferry	6.9	1	Port of Decatur
Browns Ferry	65.2	1,458	Port of Sheffield
Diablo Canyon	249.7	1,514	Port Hueneme
Haddam Neck	75.1	3,557	Port of New Haven
Haddam Neck	55.8	3,593	Port of New London
St. Lucie	141.2	155,517	Port Everglades
St. Lucie	175.0	204,530	Port of Miami
St. Lucie	20.7	355	Port of Fort Pierce
Calvert Cliffs	110.8	2,213	Port of Baltimore
Calvert Cliffs	189.1	63	Port of Norfolk
Palisades	102.4	16	Port of Muskegon
Grand Gulf	51.6	32	Port of Vicksburg
Cooper	117.1	2,780	Port of Omaha
Hope Creek	30.3	85	Port of Wilmington
Hope Creek	69.5	1,159	Port of Philadelphia
Hope Creek	131.6	6,052	Port of Baltimore
Oyster Creek	131.3	43,595	Port of Newark
Salem	31.6	85	Port of Wilmington
Salem	70.8	1,159	Port of Philadelphia
Salem	132.9	6,052	Port of Baltimore
Indian Point	89.6	59,215	Port of Newark
Surry	59.8	43	Port of Norfolk
Kewaunee	149.0	43,977	Port of Milwaukee
Point Beach	142.5	43,875	Port of Milwaukee
Total	1,349.1 – 1,861.9	199,193 – 418,945	

Note: Conversion factors are on the inside back cover of this Repository SEIS.

distances, and intermodal transfer locations analyzed in the FEIS, the impacts of using barge shipments would be similar to the impacts of using barge shipments in the FEIS, and DOE did not evaluate barge shipments further in this Repository SEIS.

G.10.11 USE OF NUREG/CR-6672 TO ESTIMATE ACCIDENT RELEASES

The evaluations of the radiological impacts of transportation accidents in the Yucca Mountain FEIS (DIRS 155970-DOE 2002, Chapter 6) are based on data in NUREG/CR-6672 (*Reexamination of Spent Nuclear Fuel Shipment Risk Estimates*, DIRS 152476-Sprung et al. 2000, all) on conditional probabilities for the occurrence of severe accidents and on corresponding fractions of cask contents that could be released in such accidents.

In September 1977, the NRC issued a generic EIS, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes* (NUREG-0170; DIRS 101892-NRC 1977, all). This EIS addressed environmental impacts associated with the transport of all types of radioactive material by all transport modes (road, rail, air, and water). It provided the basis under NEPA for the NRC to issue general licenses for transportation of radioactive material under 10 CFR Part 71. Based in part on the findings of the EIS, the NRC concluded that "present regulations are adequate to protect the public against unreasonable risk from the transport of radioactive materials" (46 FR 21629, April 13, 1981) and stated that "regulatory policy concerning transportation of radioactive materials be subject to close and continuing review."

In 1996, the NRC decided to reexamine the risks associated with the shipment of spent power reactor fuel by truck and rail to determine if the estimates of environmental impacts in NUREG-0170 (DIRS 101892-NRC 1977, all) remained valid. According to the Commission, the reexamination was initiated because (1) many spent fuel shipments are expected to be made during the next few decades, (2) these shipments will be made to facilities along routes and in casks not specifically examined by NUREG-0170, and (3) the risks associated with these shipments can be estimated using new data and improved methods of analysis. In 2000, the NRC published the results of the reexamination in a report prepared by the Sandia National Laboratories, *Reexamination of Spent Nuclear Fuel Shipment Risk Estimates* (NUREG/CR-6672; DIRS 152476-Sprung et al. 2000, all).

Some have been critical of NUREG/CR-6672 (DIRS 152476-Sprung et al. 2000, all) [for example, see DIRS 181884-Lamb and Resnikoff (2000, all) and Appendix A in DIRS 181756-Lamb et al. (2001, Appendix A]. However, the NRC has stated that that many of the purported methodological flaws appear to be related to differing views on assumptions and that critical comments do not appear to recognize that many of the assumptions overstated risks (DIRS 181603-Shankman 2001, all).

Supporting the NRC assessment, in its review of NUREG/CR-6672 (DIRS 152476-Sprung et al. 2000, all) (see *Going the Distance? The Safe Transport of Spent Nuclear and High-Level Radioactive Waste in the United States*; DIRS 182032-National Research Council 2006, all), the National Academy of Sciences Committee on Transportation of Radioactive Waste noted that the conservative assumptions were reasonable for producing bounding estimates of accident consequences. Conversely, the Committee indicated less confidence about the analysis of overall transport risks in the report. The Committee noted that the truck and rail routes used in the analyses were based on realistic, not bounding characteristics. The Committee considered “many other uncertainties” and ultimately concluded that the overall results of the “Sandia analyses are likely to be neither realistic nor bounding and ‘probably’ overestimate transport risks.”

Based on the review by the National Academy of Sciences and comments made by NRC, DOE has concluded that NUREG/CR-6672 (DIRS 152476-Sprung et al. 2000, all) represents best available information for use in estimating the consequences of transportation accidents that involve spent nuclear fuel and high-level radioactive waste and has used it in this Repository SEIS.

G.11 State-Specific Impacts and Route Maps

This section contains tables (G-22 through G-66) and maps (Figures G-3 through G-47) that illustrate the estimated impacts to 44 states and the District of Columbia (Alaska and Hawaii are not included; estimated impacts in Delaware, Montana, North Dakota, and Rhode Island would be zero). As discussed above, DOE used state- and route-specific data to estimate transportation impacts. At this time, about 10 years before shipments could begin, DOE has not determined the specific routes it would use to ship spent nuclear fuel and high-level radioactive waste to the proposed repository. Therefore, the transportation routes discussed in this section might not be the exact routes used for shipments to Yucca Mountain. Nevertheless, because the analysis is based primarily on the existing Interstate Highway System and the existing national rail network, the analysis presents a representative estimate of what the actual transportation impacts would probably be.

Table G-22. Estimated transportation impacts for the State of Alabama.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	1,514	3.9	62	0.0024	0.037	0.0030	0.011	6.3×10^{-6}	0.0087	0.052
Truck	857	4.7	7.5	0.0028	0.0045	0.0018	9.0×10^{-4}	5.4×10^{-7}	0.0052	0.014
Total	2,371	8.7	70	0.0052	0.042	0.0047	0.011	6.9×10^{-6}	0.014	0.066
Mina										
Rail	1,514	3.9	62	0.0024	0.037	0.0030	0.011	6.3×10^{-6}	0.0087	0.052
Truck	857	4.7	7.5	0.0028	0.0045	0.0018	9.0×10^{-4}	5.4×10^{-7}	0.0052	0.014
Total	2,371	8.7	70	0.0052	0.042	0.0047	0.011	6.9×10^{-6}	0.014	0.066

a. Totals might differ from sums of values due to rounding.

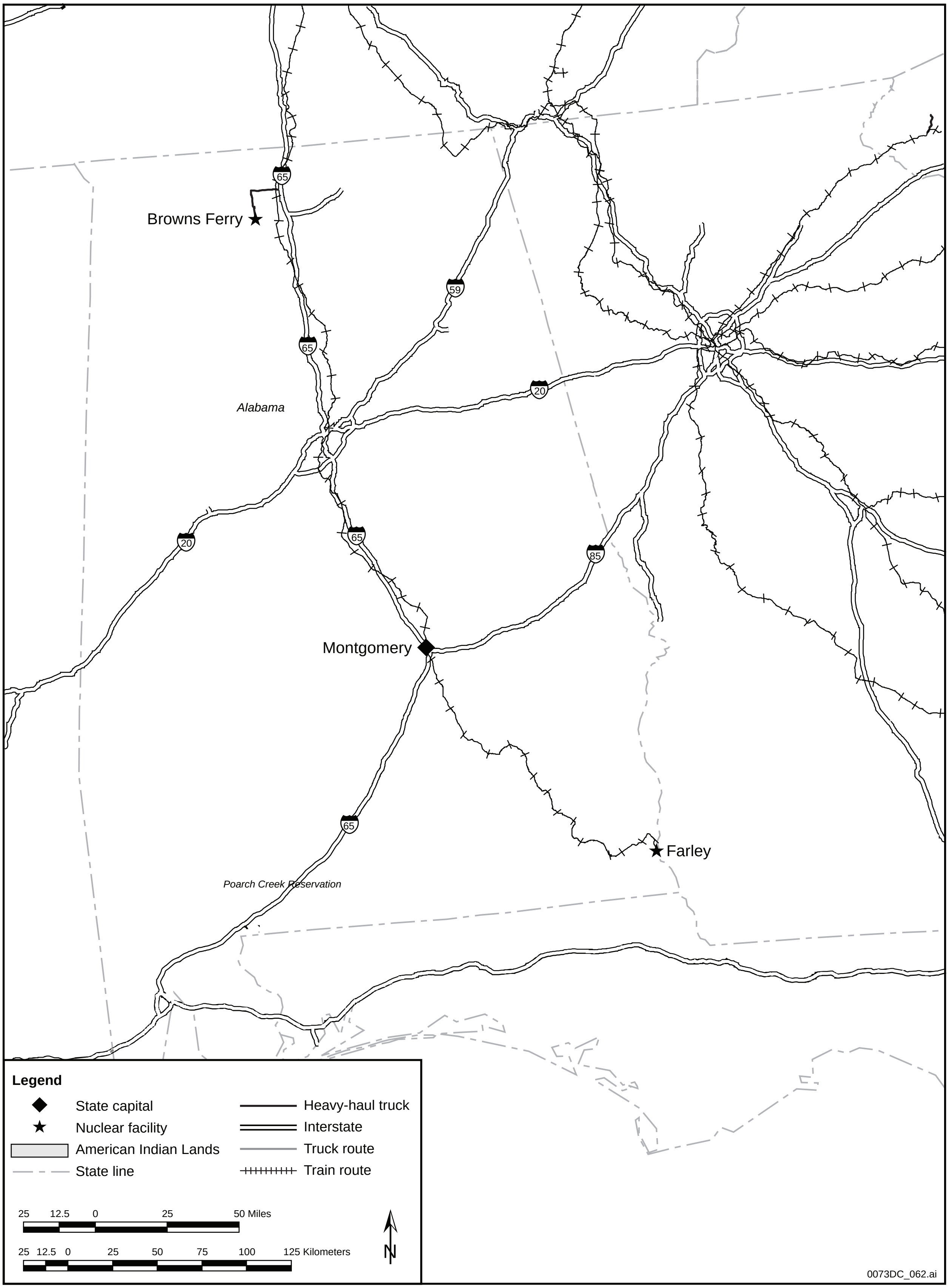

Figure G-3. Representative transportation routes for the State of Alabama.

Table G-23. Estimated transportation impacts for the State of Arizona.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	456	18	35	0.011	0.021	0.025	0.092	5.5×10^{-5}	0.016	0.073
Truck	2,650	15	38	0.0090	0.023	0.0055	0.0013	7.9×10^{-7}	0.029	0.066
Total	3,106	33	74	0.020	0.044	0.030	0.093	5.6×10^{-5}	0.045	0.14
Mina										
Rail	357	15	30	0.0092	0.018	0.021	0.077	4.6×10^{-5}	0.013	0.060
Truck	2,650	15	38	0.0090	0.023	0.0055	0.0013	7.9×10^{-7}	0.029	0.066
Total	3,007	30	68	0.018	0.041	0.026	0.078	4.7×10^{-5}	0.041	0.13

a. Totals might differ from sums of values due to rounding.

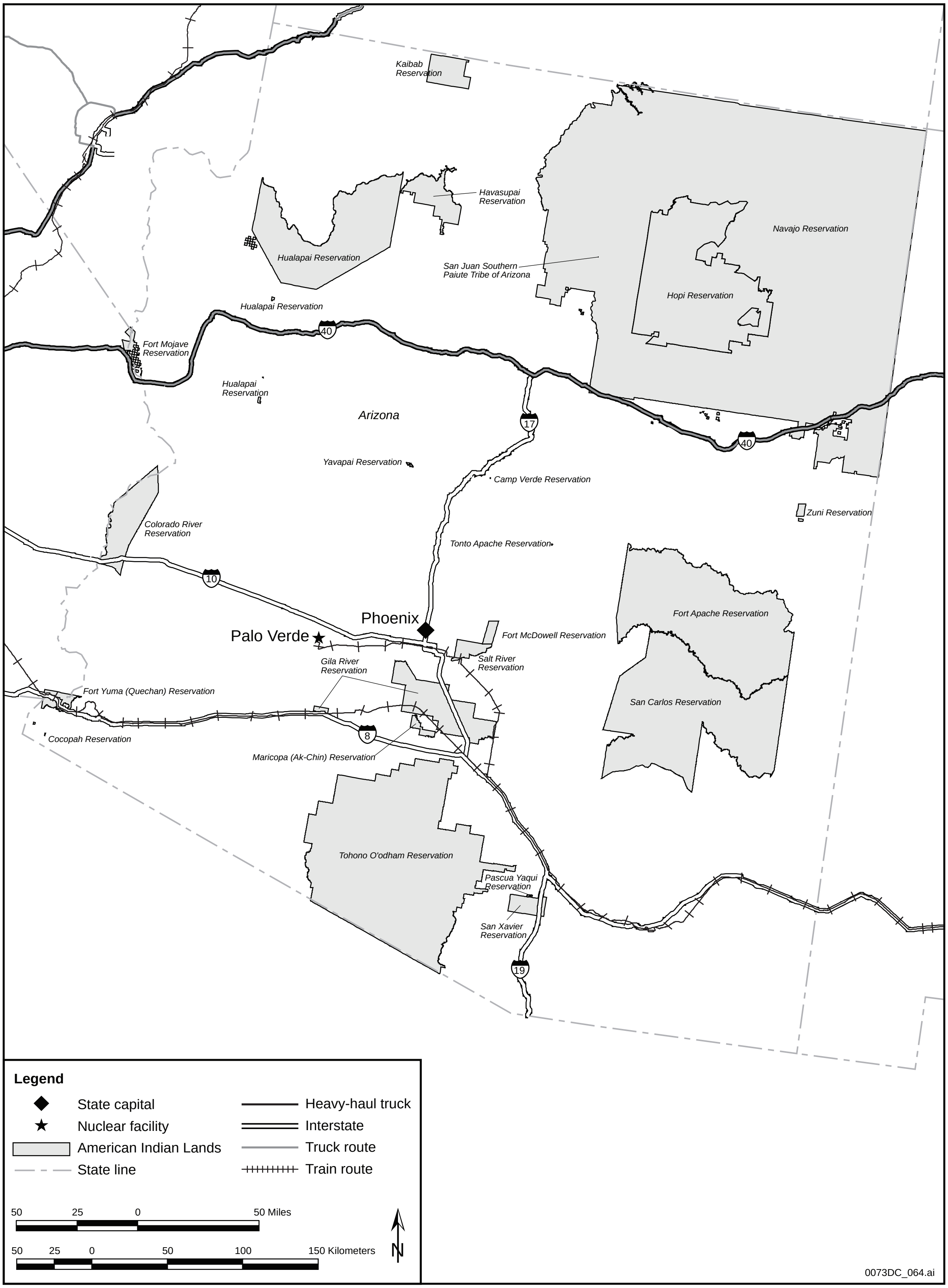

Figure G-4. Representative transportation routes for the State of Arizona.

Table G-24. Estimated transportation impacts for the State of Arkansas.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	227	0.46	11	2.7×10^{-4}	0.0063	6.7×10^{-4}	0.0035	2.1×10^{-6}	0.0026	0.0098
Truck	0	0	0	0	0	0	0	0	0	0
Total	227	0.46	11	2.7×10^{-4}	0.0063	6.7×10^{-4}	0.0035	2.1×10^{-6}	0.0026	0.0098
Mina										
Rail	227	0.46	11	2.7×10^{-4}	0.0063	6.7×10^{-4}	0.0035	2.1×10^{-6}	0.0026	0.0098
Truck	0	0	0	0	0	0	0	0	0	0
Total	227	0.46	11	2.7×10^{-4}	0.0063	6.7×10^{-4}	0.0035	2.1×10^{-6}	0.0026	0.0098

a. Totals might differ from sums of values due to rounding.

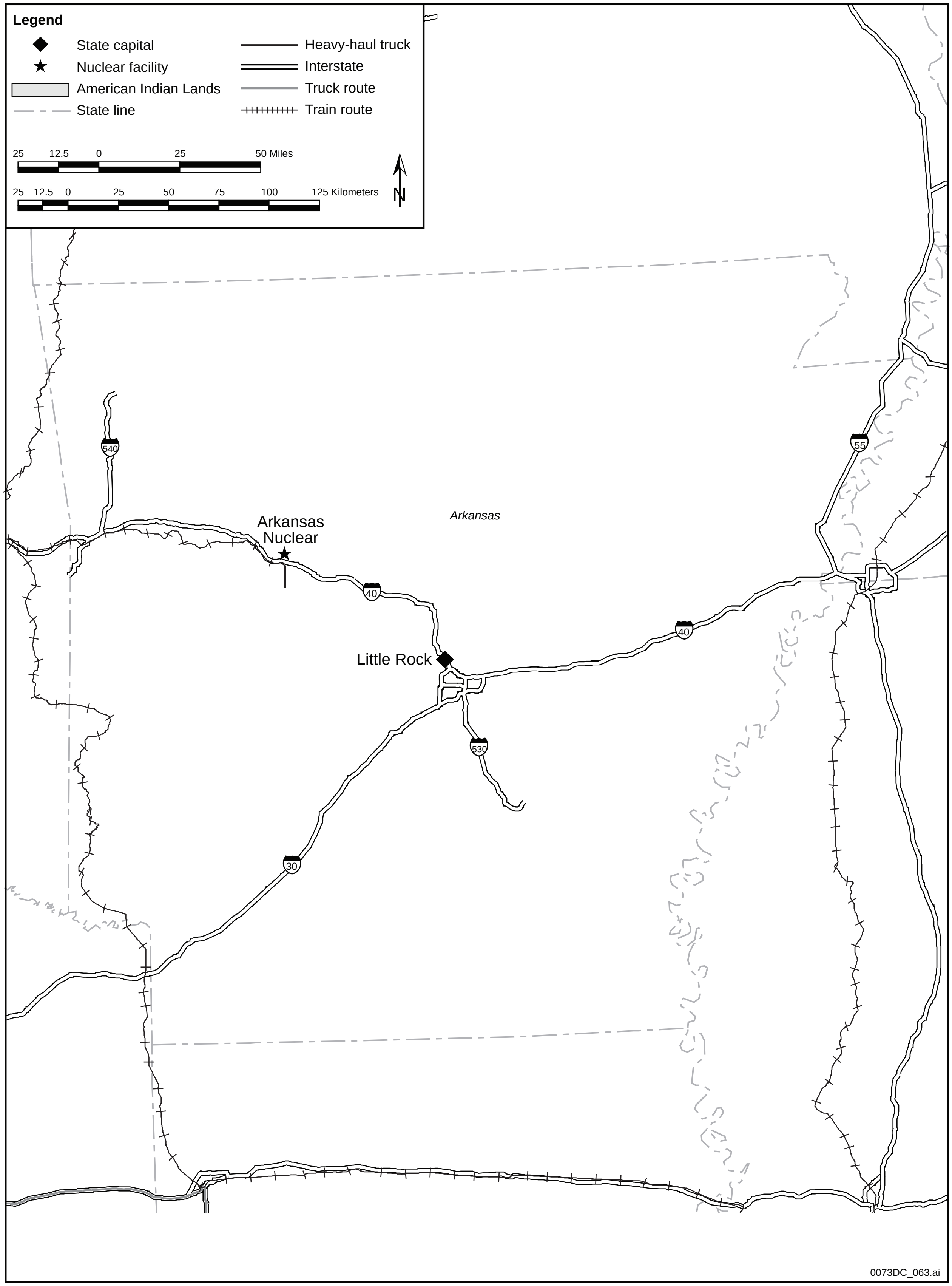

Figure G-5. Representative transportation routes for the State of Arkansas.

Table G-25. Estimated transportation impacts for the State of California.

Rail alignment	No. of Casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	755	35	82	0.021	0.049	0.042	0.16	9.9×10^{-5}	0.032	0.14
Truck	857	7.6	24	0.0045	0.015	0.0010	3.1×10^{-4}	1.9×10^{-7}	0.015	0.036
Total	1,612	43	110	0.026	0.064	0.043	0.16	9.9×10^{-5}	0.047	0.18
Mina										
Rail	1,963	99	160	0.059	0.098	0.12	0.35	2.1×10^{-4}	0.087	0.36
Truck	857	7.6	24	0.0045	0.015	0.0010	3.1×10^{-4}	1.9×10^{-7}	0.015	0.036
Total	2,820	110	190	0.064	0.11	0.12	0.35	2.1×10^{-4}	0.10	0.40

a. Totals might differ from sums of values due to rounding.

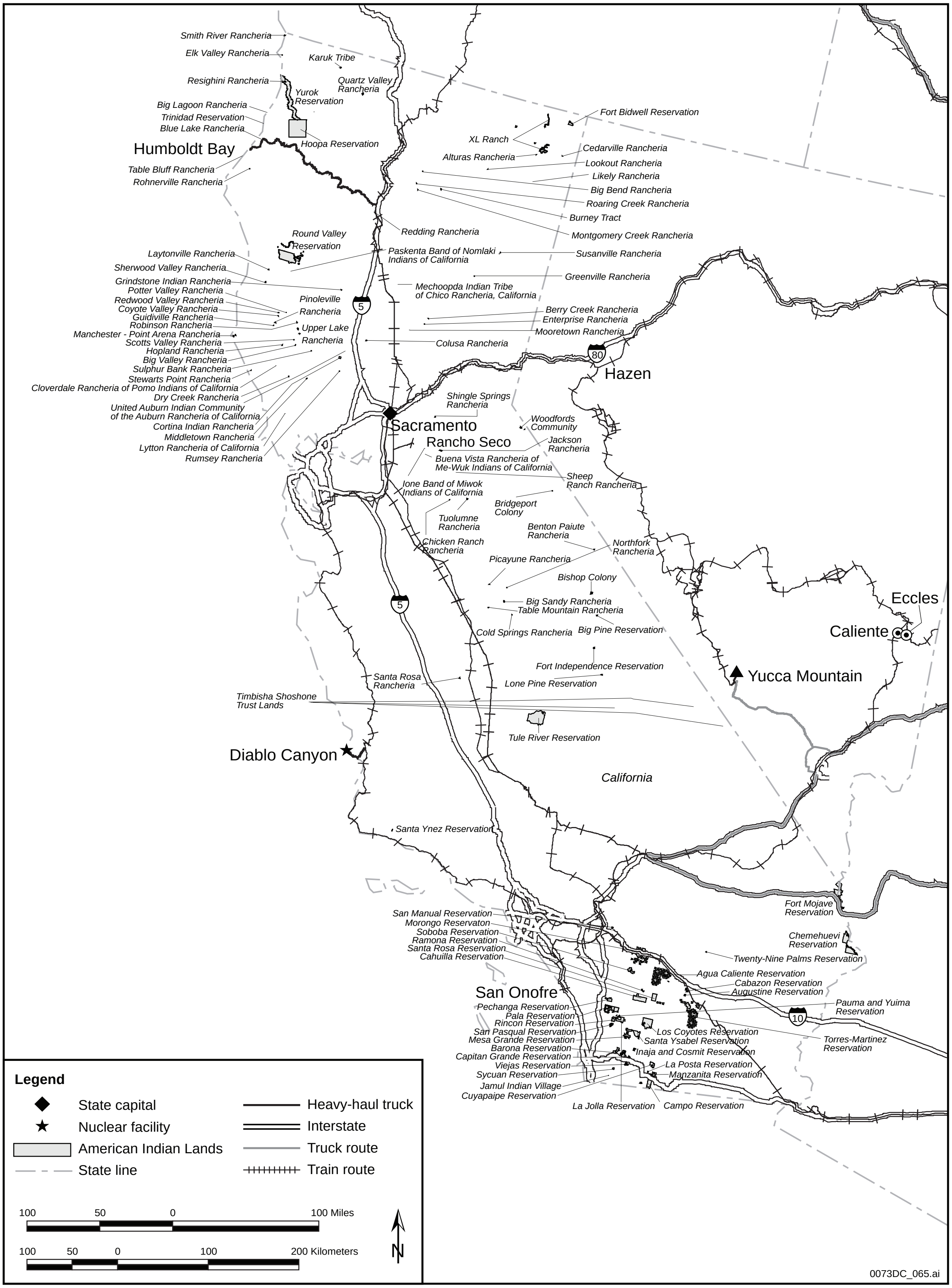

Figure G-6. Representative transportation routes for the State of California.

Table G-26. Estimated transportation impacts for the State of Colorado.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	6,739	6.8	35	0.0041	0.021	0.010	0.055	3.3×10^{-5}	0.024	0.059
Truck	0	0	0	0	0	0	0	0	0	0
Total	6,739	6.8	35	0.0041	0.021	0.010	0.055	3.3×10^{-5}	0.024	0.059
Mina										
Rail	6,838	9.4	43	0.0056	0.026	0.014	0.068	4.1×10^{-5}	0.029	0.075
Truck	0	0	0	0	0	0	0	0	0	0
Total	6,838	9.4	43	0.0056	0.026	0.014	0.068	4.1×10^{-5}	0.029	0.075

a. Totals might differ from sums of values due to rounding.

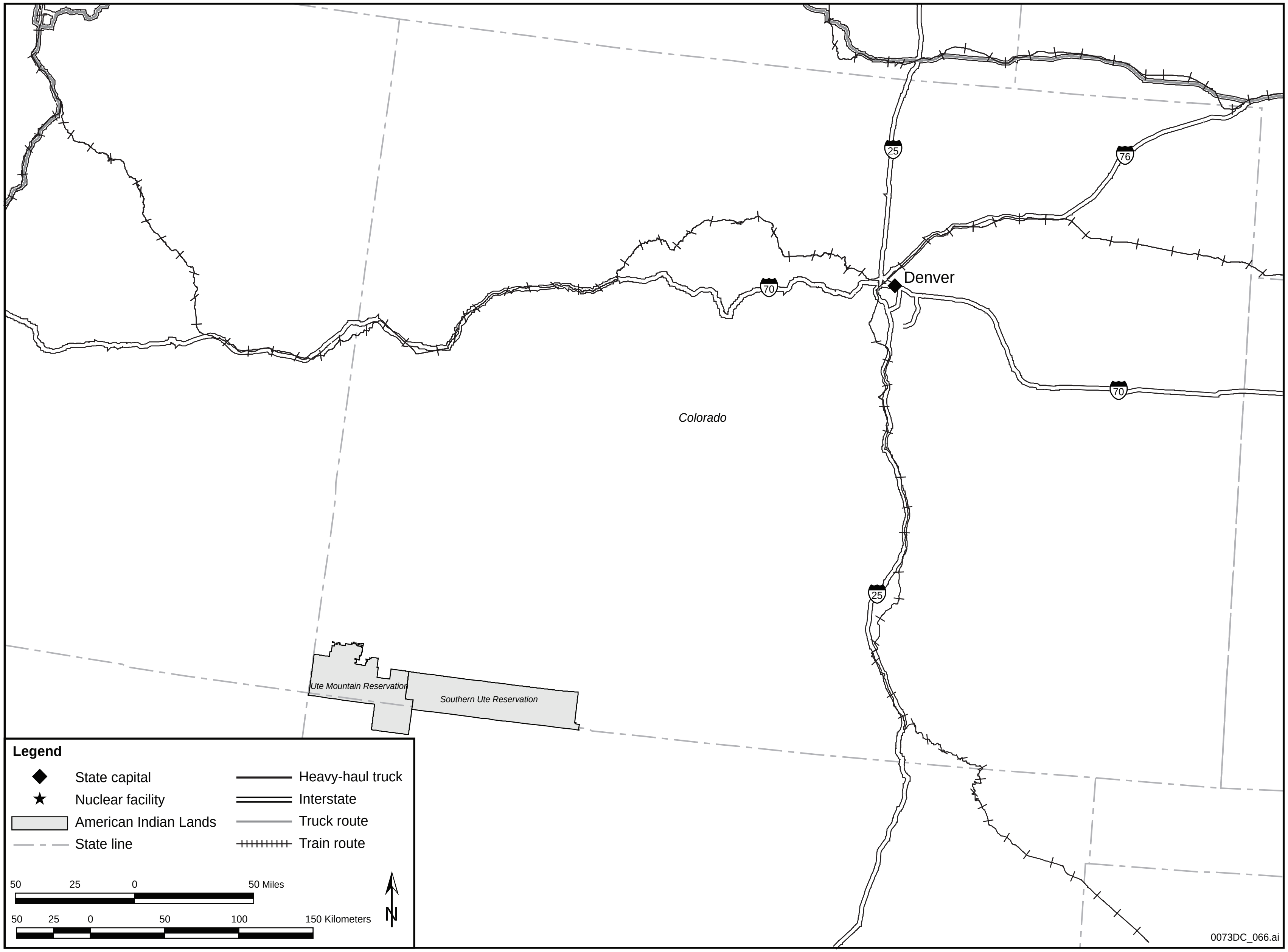

Figure G-7. Representative transportation routes for the State of Colorado.

Table G-27. Estimated transportation impacts for the State of Connecticut.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	216	1.5	19	9.2×10^{-4}	0.012	0.0017	0.0073	4.4×10^{-6}	0.0015	0.016
Truck	344	3.6	3.7	0.0022	0.0022	0.0018	0.0030	1.8×10^{-6}	0.0036	0.0098
Total	560	5.2	23	0.0031	0.014	0.0035	0.010	6.2×10^{-6}	0.0050	0.025
Mina										
Rail	216	1.5	19	9.2×10^{-4}	0.012	0.0017	0.0073	4.4×10^{-6}	0.0015	0.016
Truck	344	3.6	3.7	0.0022	0.0022	0.0018	0.0030	1.8×10^{-6}	0.0036	0.0098
Total	560	5.2	23	0.0031	0.014	0.0035	0.010	6.2×10^{-6}	0.0050	0.025

a. Totals might differ from sums of values due to rounding.

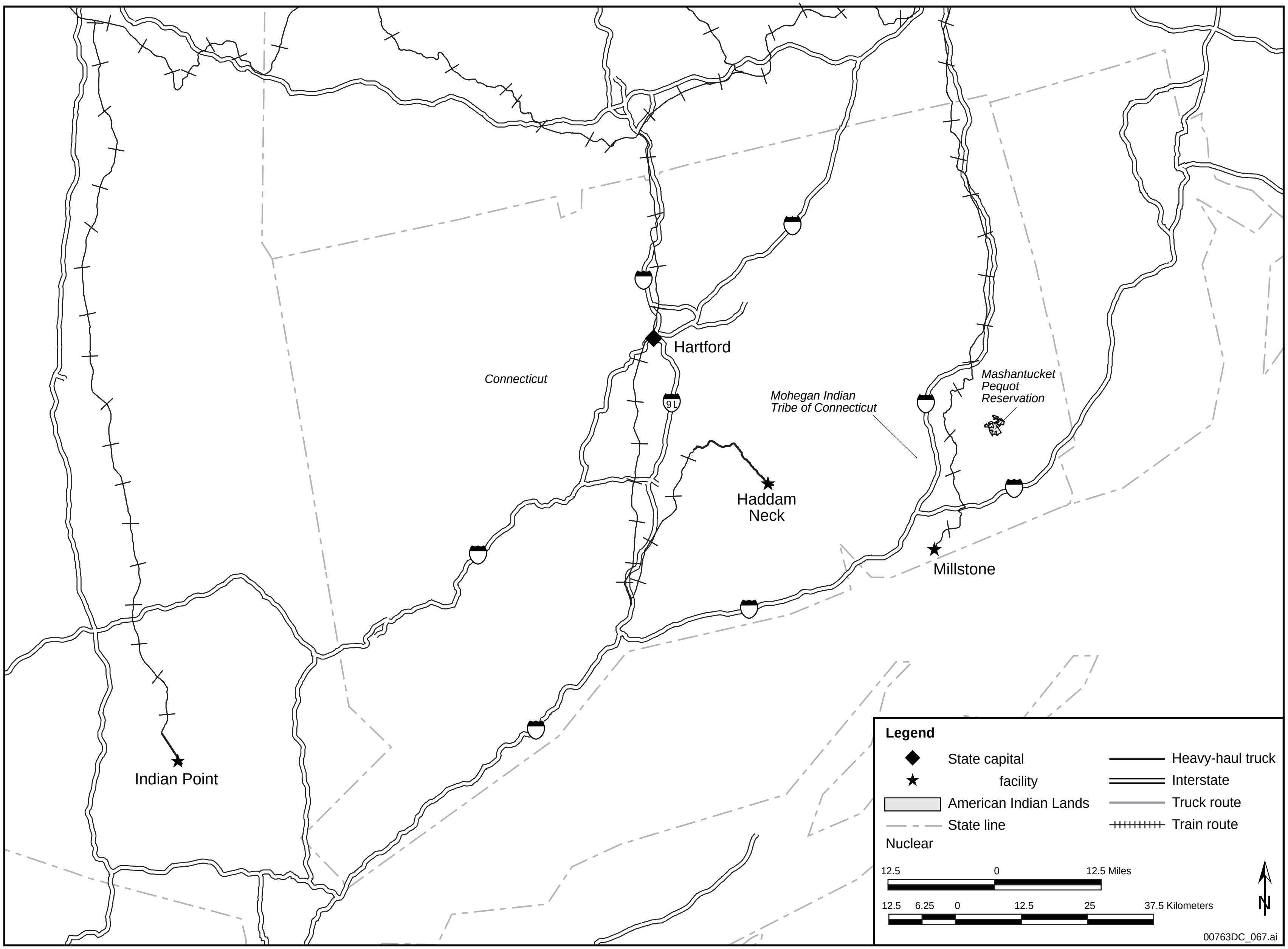

Figure G-8. Representative transportation routes for the State of Connecticut.

Table G-28. Estimated transportation impacts for the District of Columbia.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	255	1.2	0.89	7.0×10^{-4}	5.3×10^{-4}	0.0014	0.0052	3.1×10^{-6}	3.5×10^{-4}	0.0030
Truck	0	0	0	0	0	0	0	0	0	0
Total	255	1.2	0.89	7.0×10^{-4}	5.3×10^{-4}	0.0014	0.0052	3.1×10^{-6}	3.5×10^{-4}	0.0030
Mina										
Rail	255	1.2	0.89	7.0×10^{-4}	5.3×10^{-4}	0.0014	0.0052	3.1×10^{-6}	3.5×10^{-4}	0.0030
Truck	0	0	0	0	0	0	0	0	0	0
Total	255	1.2	0.89	7.0×10^{-4}	5.3×10^{-4}	0.0014	0.0052	3.1×10^{-6}	3.5×10^{-4}	0.0030

a. Totals might differ from sums of values due to rounding.

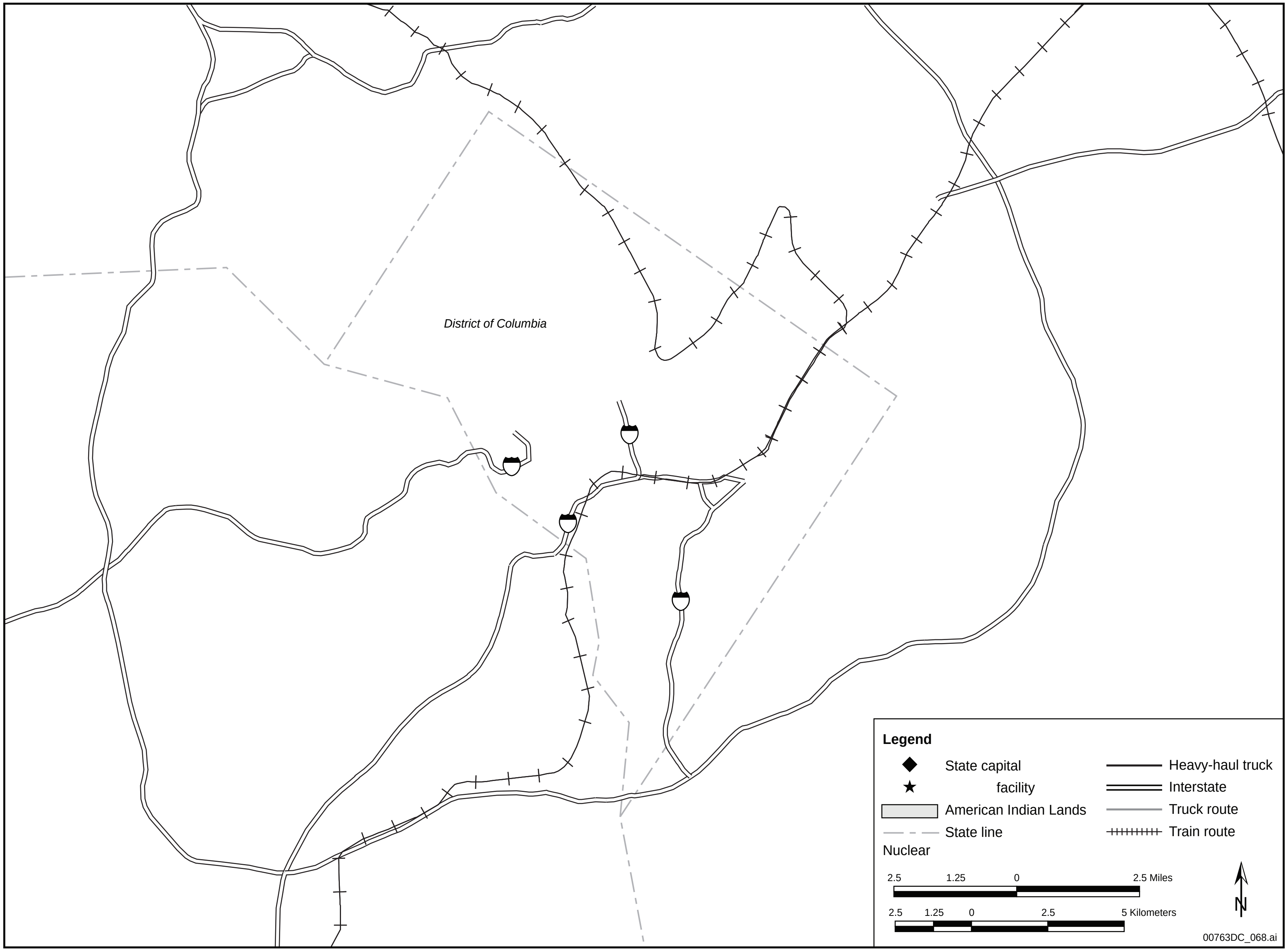

Figure G-9. Representative transportation routes for the District of Columbia.

Table G-29. Estimated transportation impacts for the State of Florida.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	138	13	31	0.0078	0.019	0.013	0.047	2.8×10^{-5}	0.0039	0.043
Truck	857	47	100	0.028	0.060	0.032	0.0040	2.4×10^{-6}	0.040	0.16
Total	995	60	130	0.036	0.079	0.044	0.051	3.1×10^{-5}	0.044	0.20
Mina										
Rail	138	13	31	0.0078	0.019	0.013	0.047	2.8×10^{-5}	0.0039	0.043
Truck	857	47	100	0.028	0.060	0.032	0.0040	2.4×10^{-6}	0.040	0.16
Total	995	60	130	0.036	0.079	0.044	0.051	3.1×10^{-5}	0.044	0.20

a. Totals might differ from sums of values due to rounding.

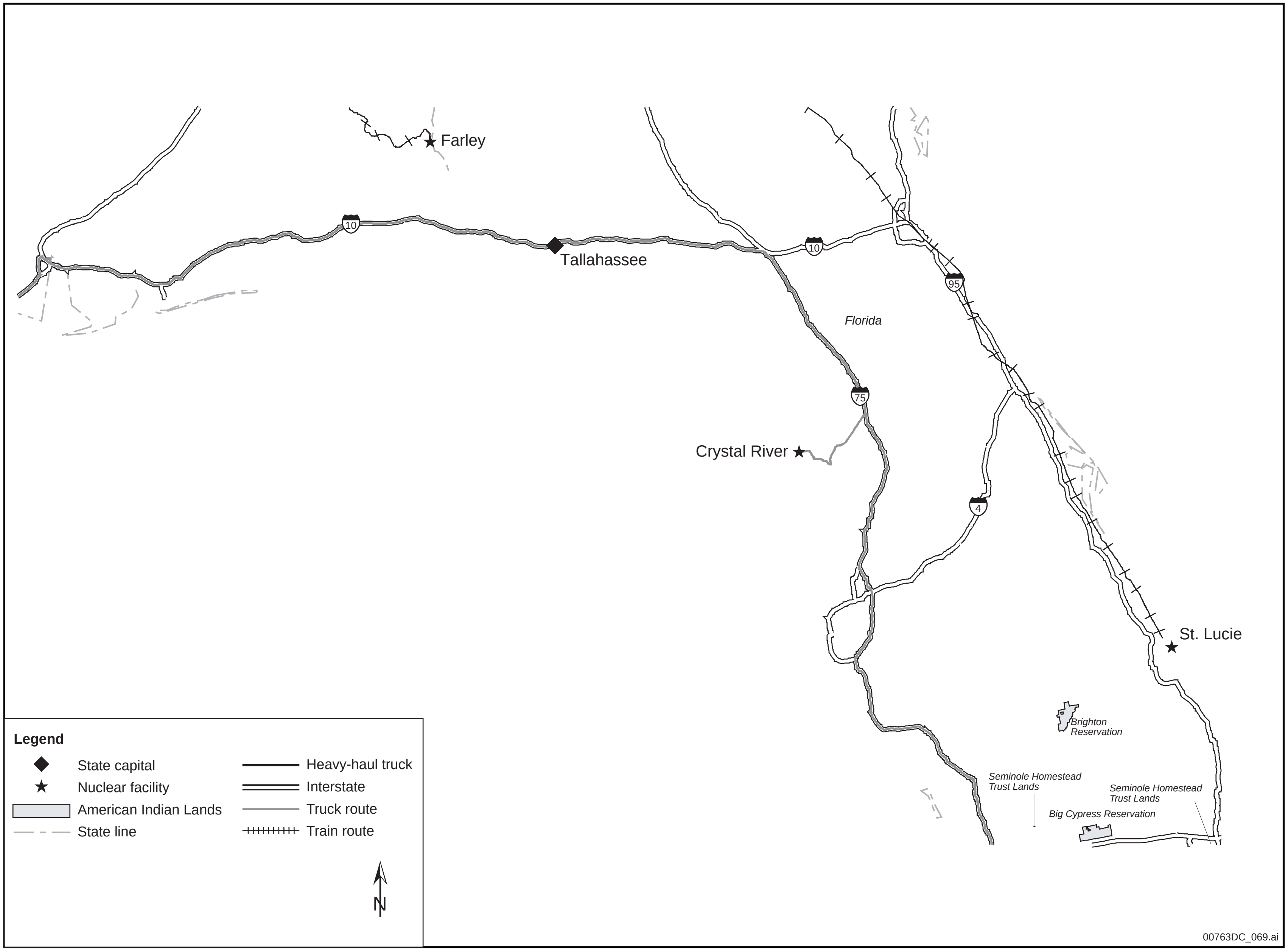

Figure G-10. Representative transportation routes for the State of Florida.

Table G-30. Estimated transportation impacts for the State of Georgia.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	1,672	53	85	0.032	0.051	0.065	0.17	1.0×10^{-4}	0.044	0.19
Truck	0	0	0	0	0	0	0	0	0	0
Total	1,672	53	85	0.032	0.051	0.065	0.17	1.0×10^{-4}	0.044	0.19
Mina										
Rail	1,672	53	85	0.032	0.051	0.065	0.17	1.0×10^{-4}	0.044	0.19
Truck	0	0	0	0	0	0	0	0	0	0
Total	1,672	53	85	0.032	0.051	0.065	0.17	1.0×10^{-4}	0.044	0.19

a. Totals might differ from sums of values due to rounding.

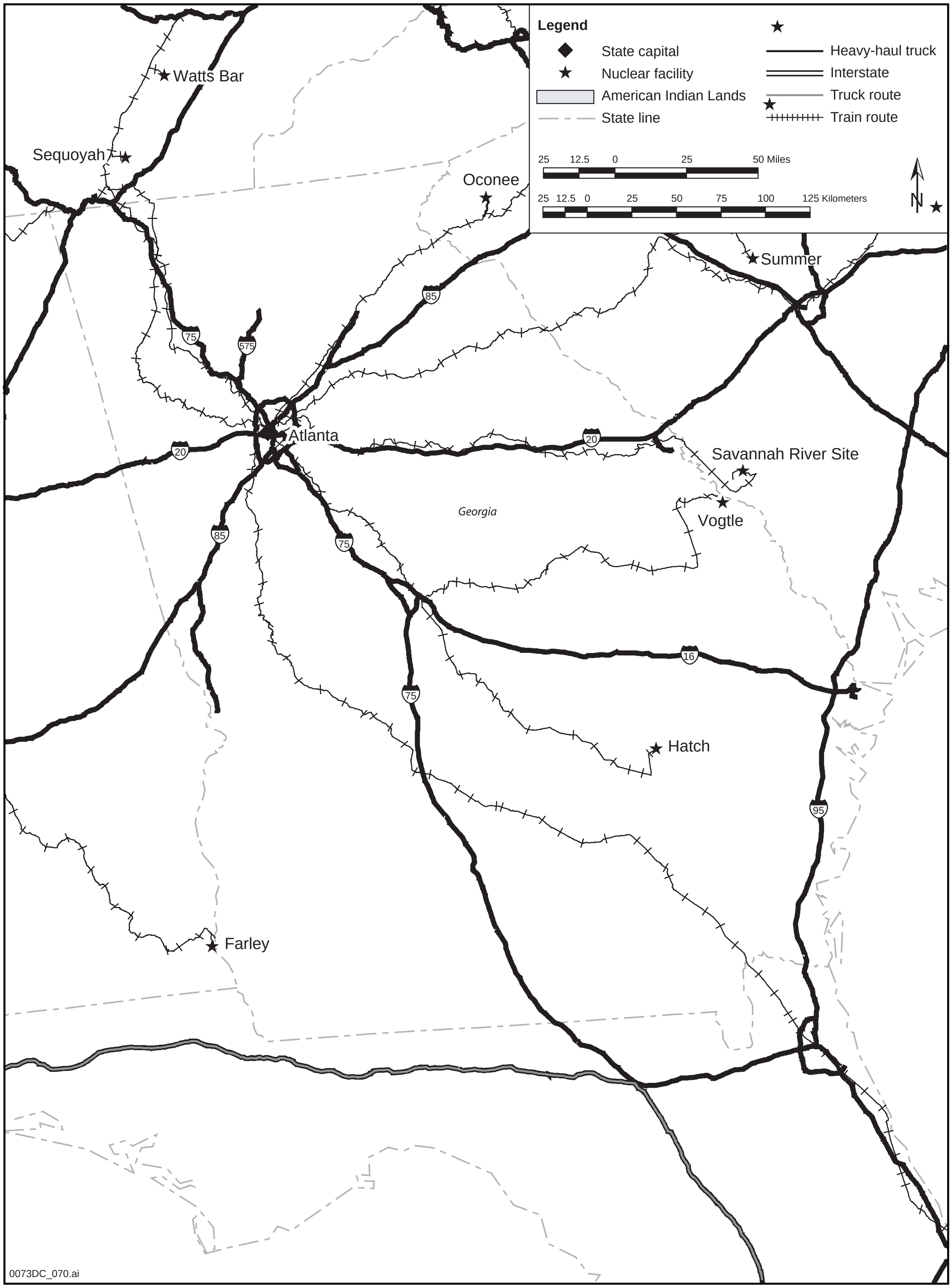

Figure G-11. Representative transportation routes for the State of Georgia.

Table G-31. Estimated transportation impacts for the State of Idaho.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	2,001	28	310	0.017	0.19	0.021	0.015	9.1×10^{-6}	0.046	0.27
Truck	4	0.046	0.15	2.8×10^{-5}	9.0×10^{-5}	1.7×10^{-5}	9.0×10^{-6}	5.4×10^{-9}	5.0×10^{-5}	1.8×10^{-4}
Total	2,005	28	310	0.017	0.19	0.021	0.015	9.1×10^{-6}	0.046	0.27
Mina										
Rail	694	13	270	0.0080	0.16	0.0043	0.0017	1.0×10^{-6}	0.0077	0.18
Truck	4	0.046	0.15	2.8×10^{-5}	9.0×10^{-5}	1.7×10^{-5}	9.0×10^{-6}	5.4×10^{-9}	5.0×10^{-5}	1.8×10^{-4}
Total	698	13	270	0.0080	0.16	0.0044	0.0017	1.0×10^{-6}	0.0077	0.18

a. Totals might differ from sums of values due to rounding.

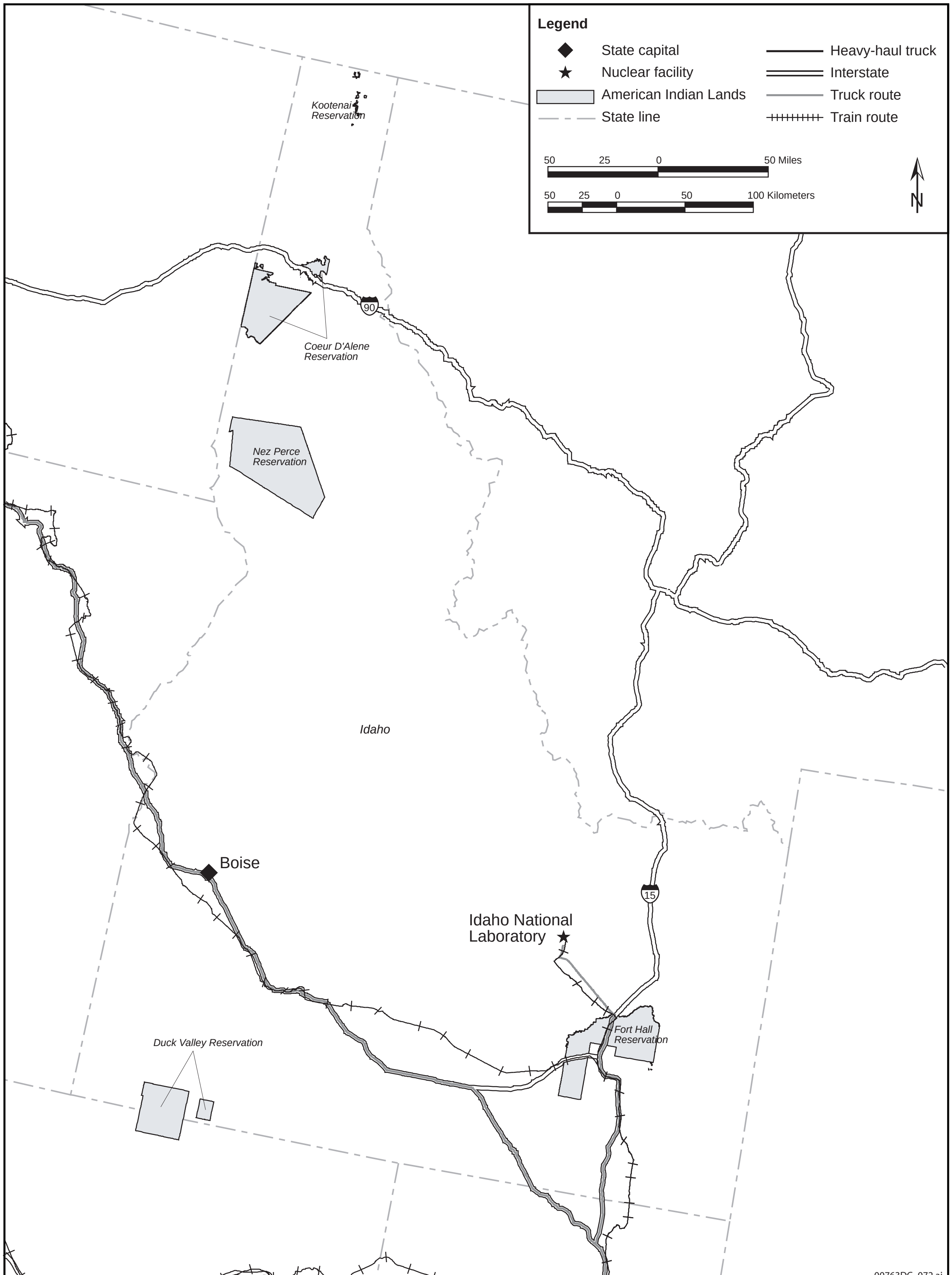

Figure G-12. Representative transportation routes for the State of Idaho.

Table G-32. Estimated transportation impacts for the State of Illinois.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	6,069	75	200	0.045	0.12	0.094	0.47	2.8×10^{-4}	0.091	0.35
Truck	1,752	15	46	0.0090	0.028	0.0044	0.0020	1.2×10^{-6}	0.021	0.062
Total	7,821	90	250	0.054	0.15	0.099	0.47	2.8×10^{-4}	0.11	0.41
Mina										
Rail	6,069	75	200	0.045	0.12	0.094	0.47	2.8×10^{-4}	0.091	0.35
Truck	1,752	15	46	0.0090	0.028	0.0044	0.0020	1.2×10^{-6}	0.021	0.062
Total	7,821	90	250	0.054	0.15	0.099	0.47	2.8×10^{-4}	0.11	0.41

a. Totals might differ from sums of values due to rounding.

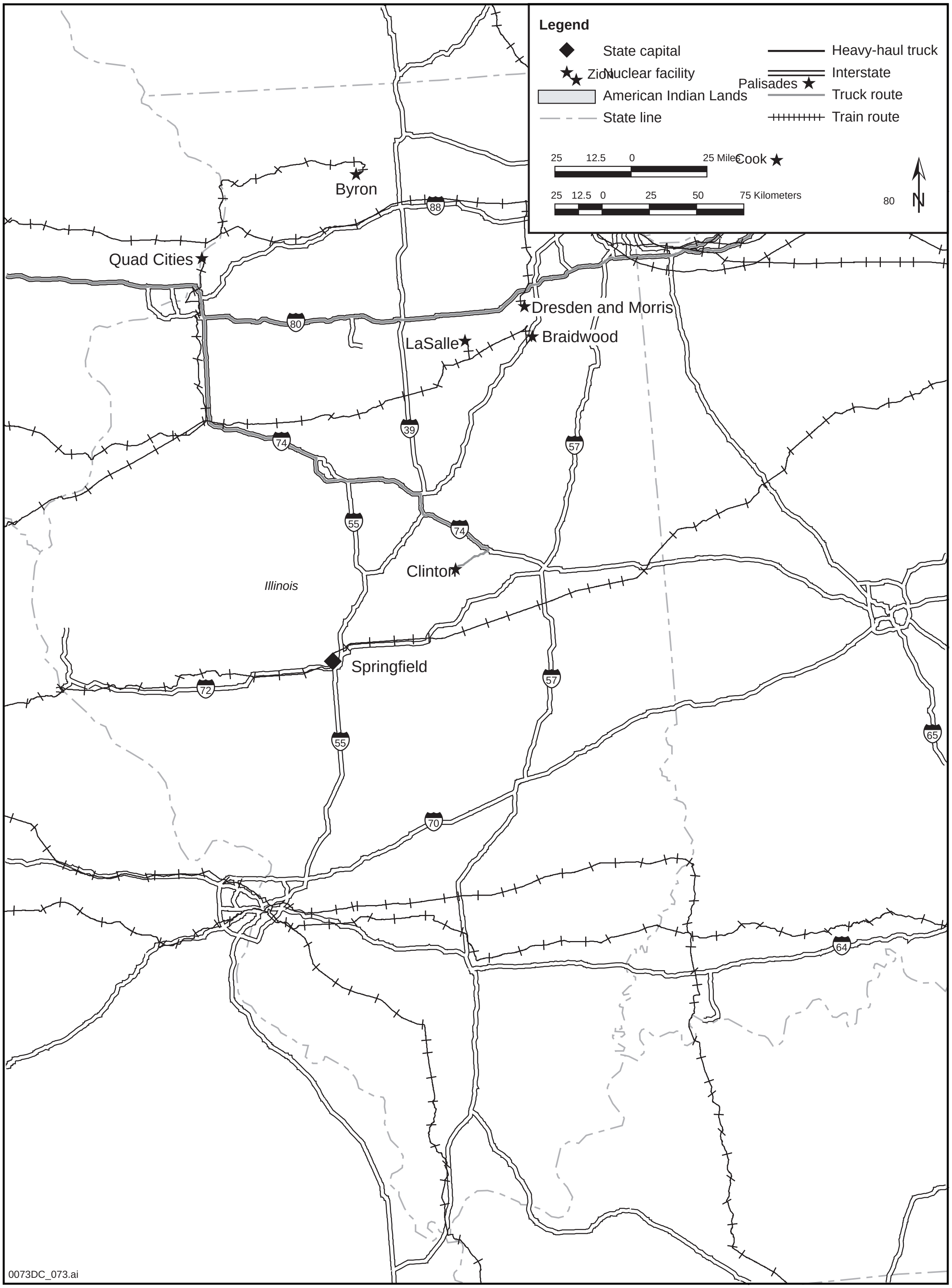

Figure G-13. Representative transportation routes for the State of Illinois.

Table G-33. Estimated transportation impacts for the State of Indiana.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	4,887	27	86	0.016	0.052	0.036	0.18	1.1×10^{-4}	0.055	0.16
Truck	1,425	9.1	15	0.0055	0.0088	0.0035	0.0015	9.0×10^{-7}	0.0089	0.027
Total	6,312	36	100	0.021	0.061	0.039	0.19	1.1×10^{-4}	0.064	0.19
Mina										
Rail	4,887	27	86	0.016	0.052	0.036	0.18	1.1×10^{-4}	0.055	0.16
Truck	1,425	9.1	15	0.0055	0.0088	0.0035	0.0015	9.0×10^{-7}	0.0089	0.027
Total	6,312	36	100	0.021	0.061	0.039	0.19	1.1×10^{-4}	0.064	0.19

a. Totals might differ from sums of values due to rounding.

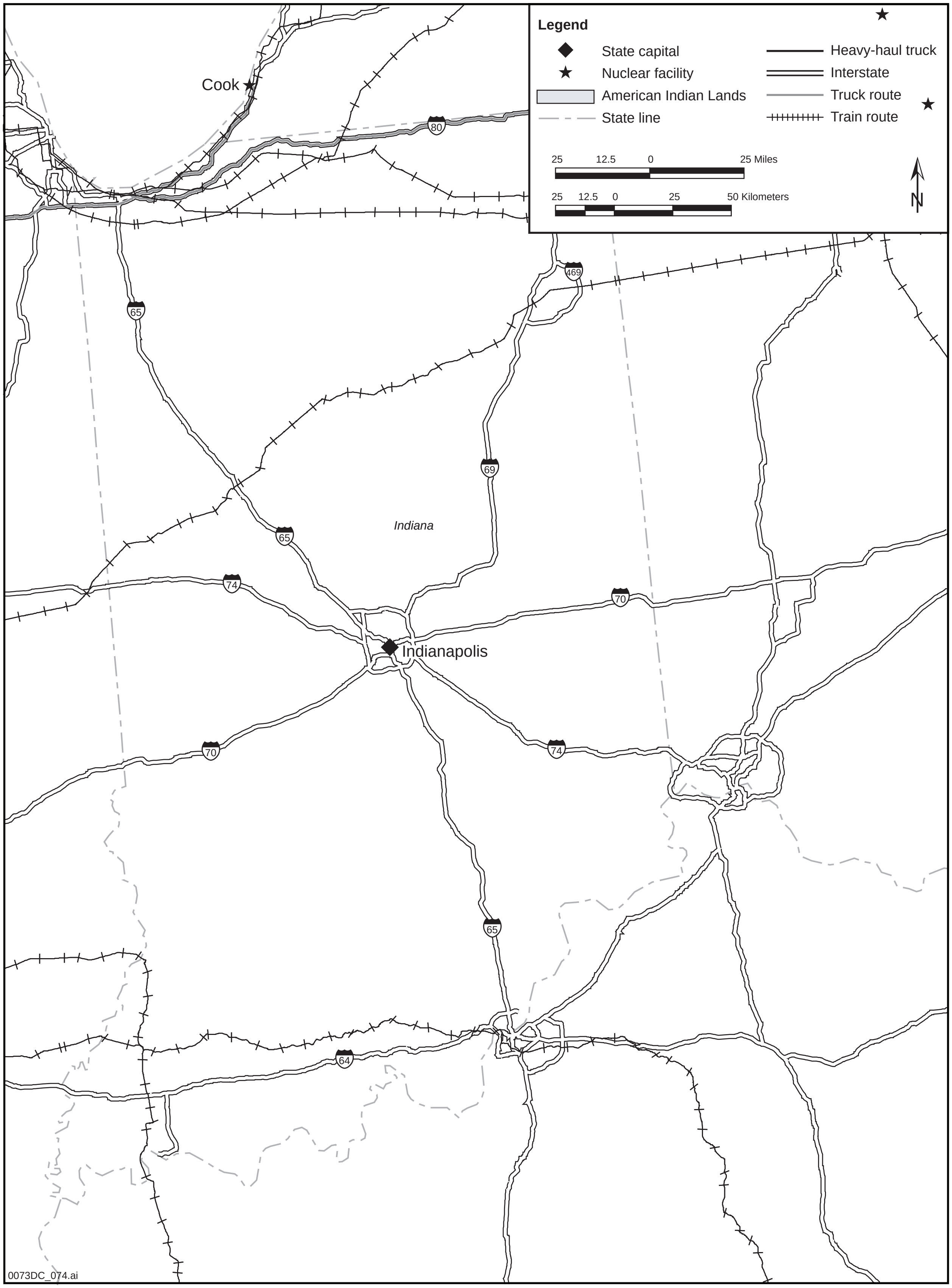

Figure G-14. Representative transportation routes for the State of Indiana.

Table G-34. Estimated transportation impacts for the State of Iowa.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	3,066	13	150	0.0079	0.089	0.020	0.19	1.2×10^{-4}	0.096	0.21
Truck	1,789	22	59	0.013	0.035	0.0037	0.0011	6.5×10^{-7}	0.044	0.096
Total	4,855	35	210	0.021	0.12	0.023	0.19	1.2×10^{-4}	0.14	0.31
Mina										
Rail	3,066	13	150	0.0079	0.089	0.020	0.19	1.2×10^{-4}	0.096	0.21
Truck	1,789	22	59	0.013	0.035	0.0037	0.0011	6.5×10^{-7}	0.044	0.096
Total	4,855	35	210	0.021	0.12	0.023	0.19	1.2×10^{-4}	0.14	0.31

a. Totals might differ from sums of values due to rounding.

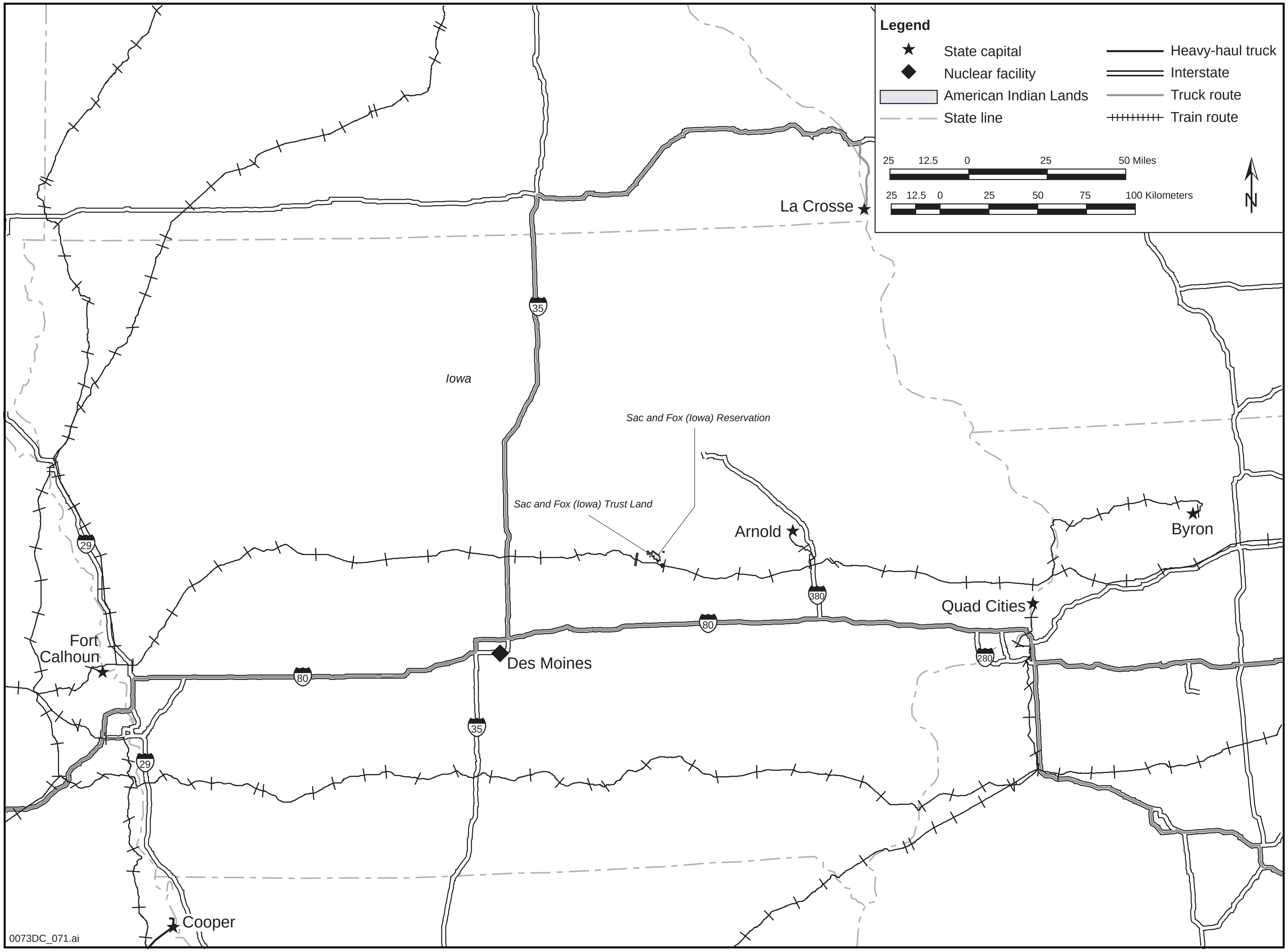

Figure G-15. Representative transportation routes for the State of Iowa.

Table G-35. Estimated transportation impacts for the State of Kansas.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	3,574	8.7	90	0.0052	0.054	0.012	0.066	3.9×10^{-5}	0.061	0.13
Truck	0	0	0	0	0	0	0	0	0	0
Total	3,574	8.7	90	0.0052	0.054	0.012	0.066	3.9×10^{-5}	0.061	0.13
Mina										
Rail	3,574	8.7	90	0.0052	0.054	0.012	0.066	3.9×10^{-5}	0.061	0.13
Truck	0	0	0	0	0	0	0	0	0	0
Total	3,574	8.7	90	0.0052	0.054	0.012	0.066	3.9×10^{-5}	0.061	0.13

a. Totals might differ from sums of values due to rounding.

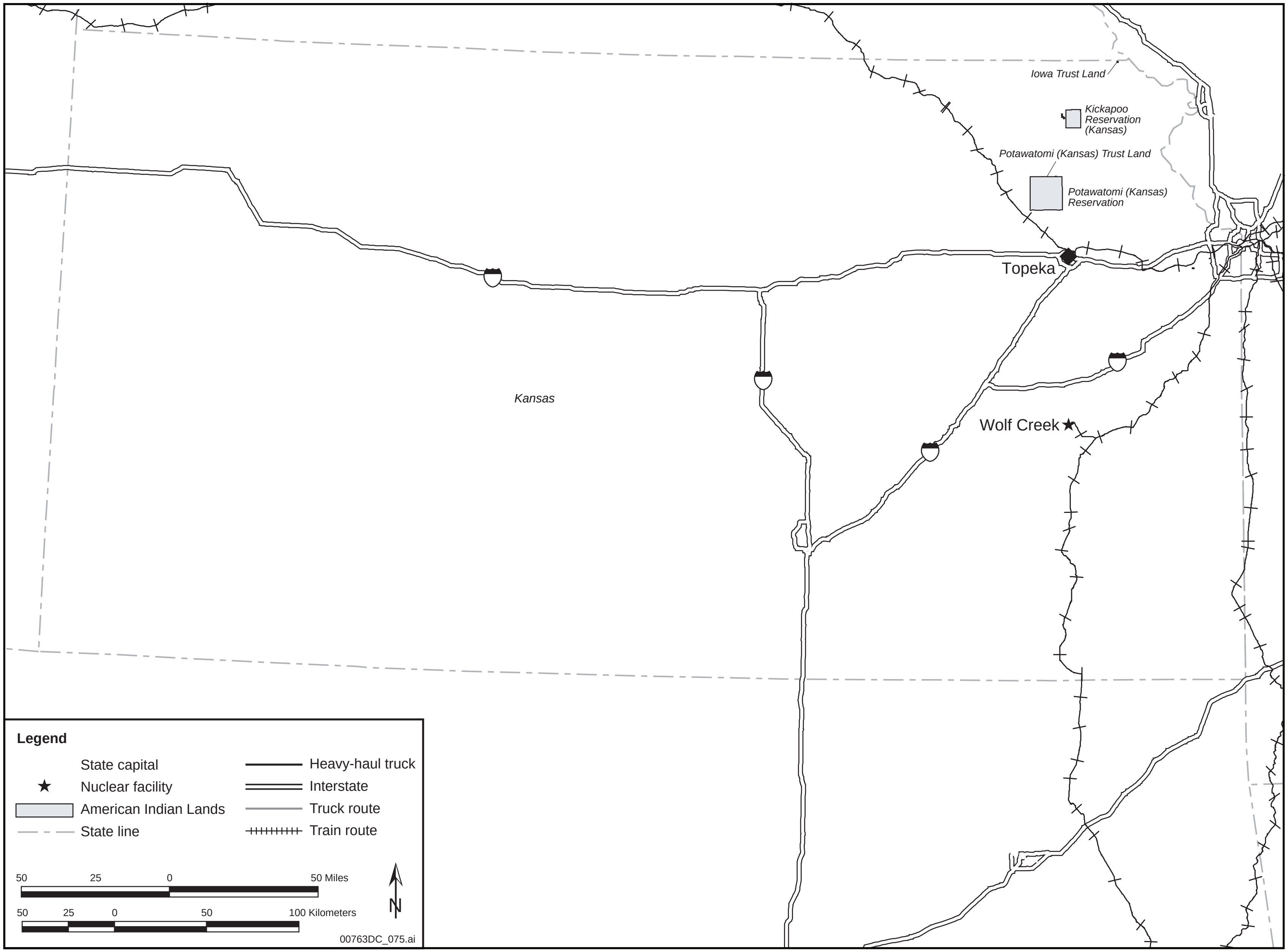

Figure G-16. Representative transportation routes for the State of Kansas.

Table G-36. Estimated transportation impacts for the Commonwealth of Kentucky.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	2,663	14	50	0.0086	0.030	0.020	0.077	4.6×10^{-5}	0.032	0.090
Truck	0	0	0	0	0	0	0	0	0	0
Total	2,663	14	50	0.0086	0.030	0.020	0.077	4.6×10^{-5}	0.032	0.090
Mina										
Rail	2,663	14	50	0.0086	0.030	0.020	0.077	4.6×10^{-5}	0.032	0.090
Truck	0	0	0	0	0	0	0	0	0	0
Total	2,663	14	50	0.0086	0.030	0.020	0.077	4.6×10^{-5}	0.032	0.090

a. Totals might differ from sums of values due to rounding.

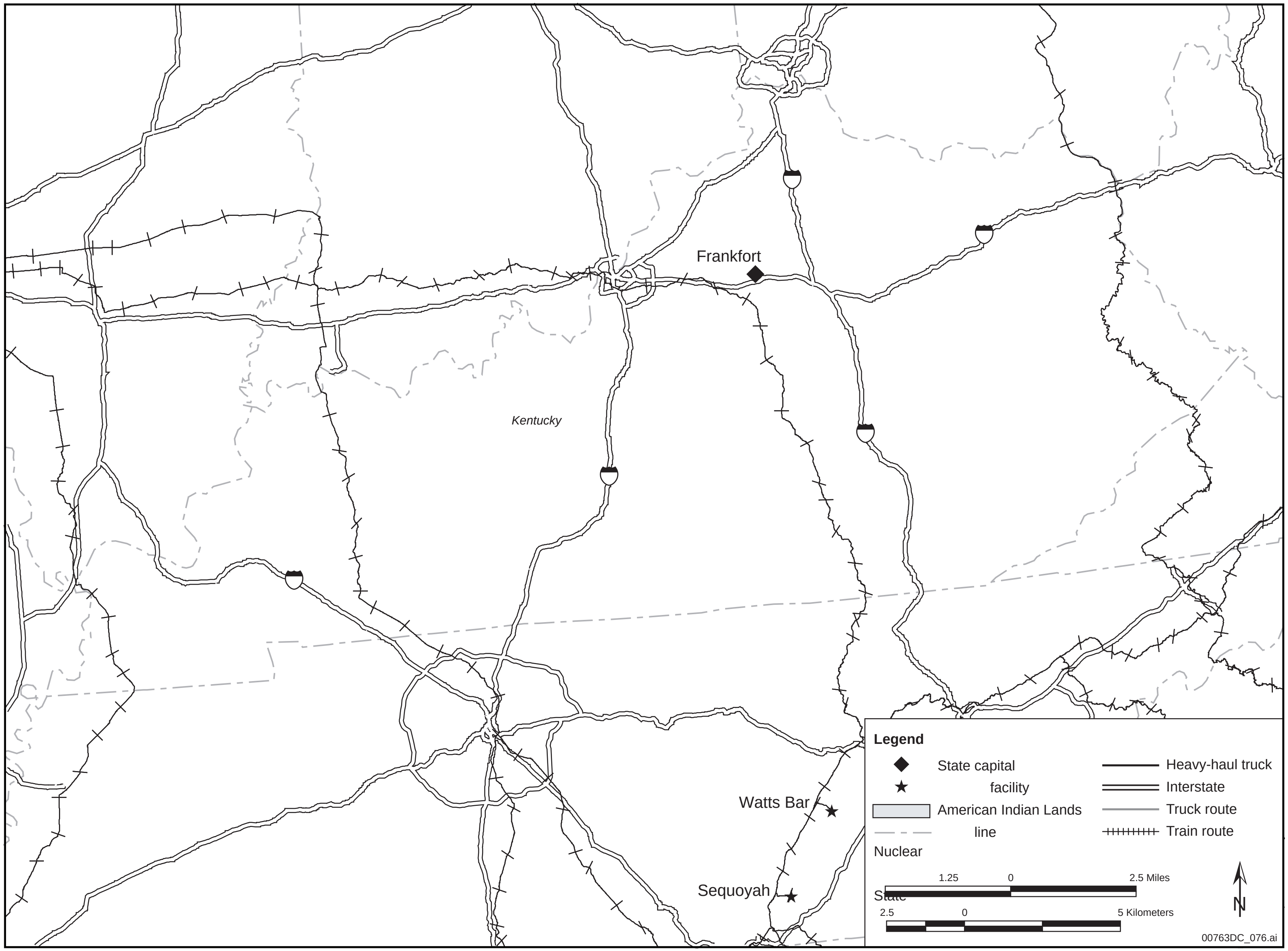

Figure G-17. Representative transportation routes for the Commonwealth of Kentucky.

Table G-37. Estimated transportation impacts for the State of Louisiana.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	233	1.3	14	7.8×10^{-4}	0.0082	0.0019	0.0098	5.9×10^{-6}	0.0043	0.015
Truck	857	17	35	0.010	0.021	0.0054	0.0022	1.3×10^{-6}	0.025	0.062
Total	1,090	19	48	0.011	0.029	0.0073	0.012	7.2×10^{-6}	0.029	0.077
Mina										
Rail	233	1.3	14	7.8×10^{-4}	0.0082	0.0019	0.0098	5.9×10^{-6}	0.0043	0.015
Truck	857	17	35	0.010	0.021	0.0054	0.0022	1.3×10^{-6}	0.025	0.062
Total	1,090	19	48	0.011	0.029	0.0073	0.012	7.2×10^{-6}	0.029	0.077

a. Totals might differ from sums of values due to rounding.

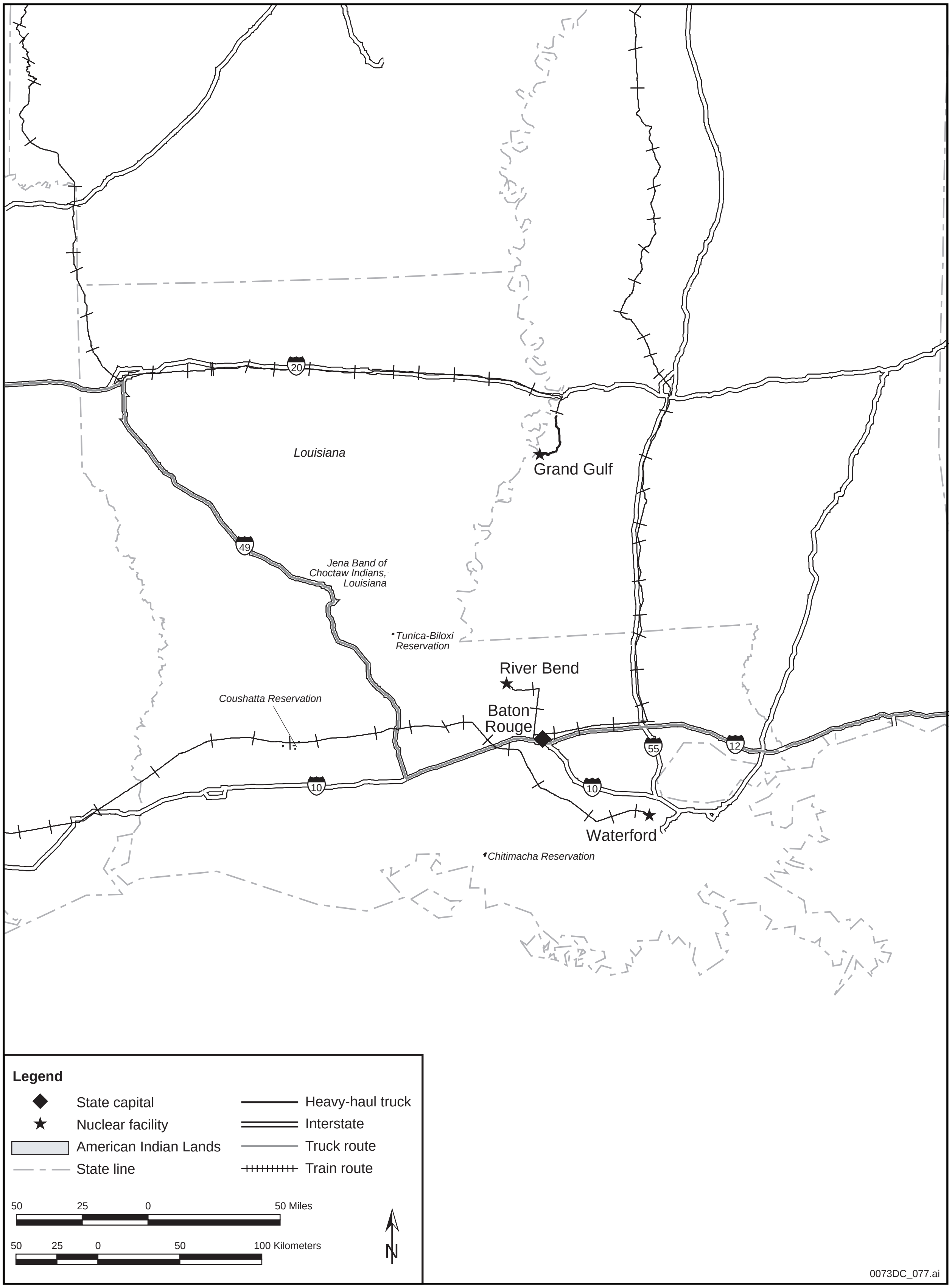

Figure G-18. Representative transportation routes for the State of Louisiana.

Table G-38. Estimated transportation impacts for the State of Maine.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	60	0.38	4.1	2.3×10^{-4}	0.0025	5.0×10^{-4}	0.0021	1.3×10^{-6}	5.3×10^{-4}	0.0037
Truck	0	0	0	0	0	0	0	0	0	0
Total	60	0.38	4.1	2.3×10^{-4}	0.0025	5.0×10^{-4}	0.0021	1.3×10^{-6}	5.3×10^{-4}	0.0037
Mina										
Rail	60	0.38	4.1	2.3×10^{-4}	0.0025	5.0×10^{-4}	0.0021	1.3×10^{-6}	5.3×10^{-4}	0.0037
Truck	0	0	0	0	0	0	0	0	0	0
Total	60	0.38	4.1	2.3×10^{-4}	0.0025	5.0×10^{-4}	0.0021	1.3×10^{-6}	5.3×10^{-4}	0.0037

a. Totals might differ from sums of values due to rounding.

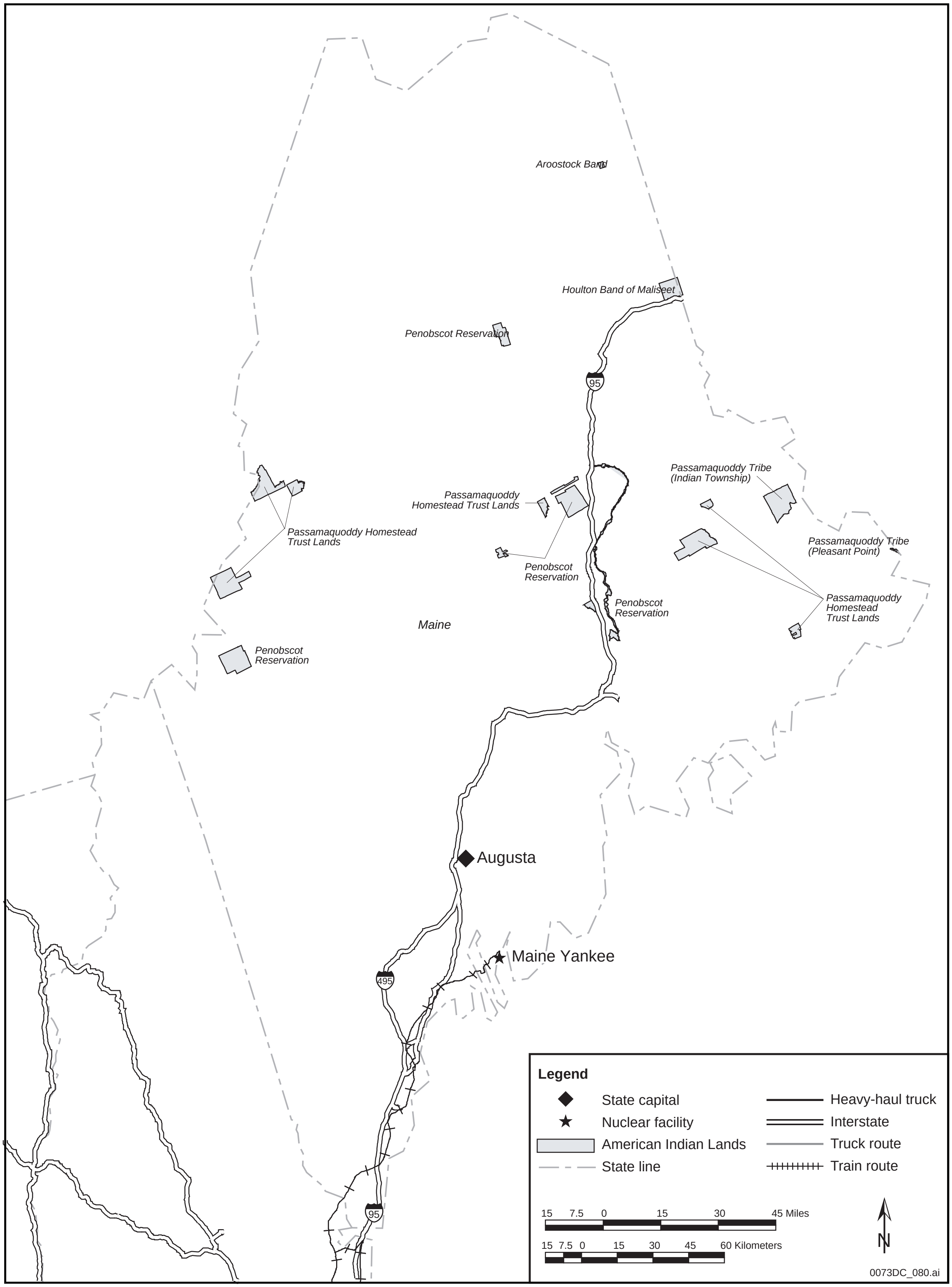

Figure G-19. Representative transportation routes for the State of Maine.

Table G-39. Estimated transportation impacts for the State of Maryland.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	255	7.9	30	0.0047	0.018	0.0075	0.029	1.8×10^{-5}	0.0039	0.034
Truck	0	0	0	0	0	0	0	0	0	0
Total	255	7.9	30	0.0047	0.018	0.0075	0.029	1.8×10^{-5}	0.0039	0.034
Mina										
Rail	255	7.9	30	0.0047	0.018	0.0075	0.029	1.8×10^{-5}	0.0039	0.034
Truck	0	0	0	0	0	0	0	0	0	0
Total	255	7.9	30	0.0047	0.018	0.0075	0.029	1.8×10^{-5}	0.0039	0.034

a. Totals might differ from sums of values due to rounding.

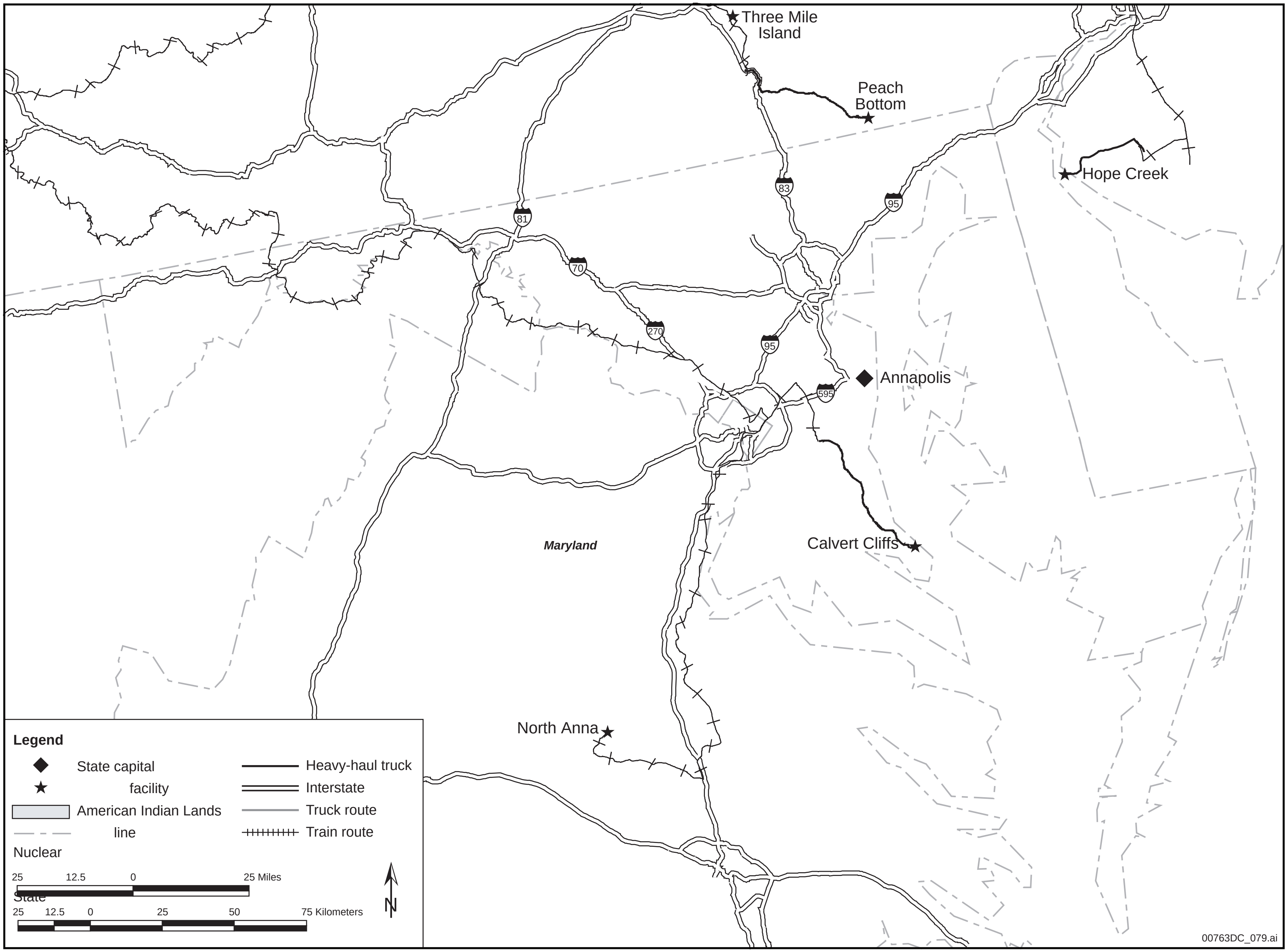

Figure G-20. Representative transportation routes for the State of Maryland.

Table G-40. Estimated transportation impacts for the Commonwealth of Massachusetts.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	415	4.8	12	0.0029	0.0071	0.0064	0.028	1.7×10^{-5}	0.0053	0.022
Truck	344	2.5	19	0.0015	0.012	8.9×10^{-4}	1.4×10^{-4}	8.4×10^{-8}	0.0013	0.015
Total	759	7.3	31	0.0044	0.019	0.0072	0.028	1.7×10^{-5}	0.0066	0.037
Mina										
Rail	415	4.8	12	0.0029	0.0071	0.0064	0.028	1.7×10^{-5}	0.0053	0.022
Truck	344	2.5	19	0.0015	0.012	8.9×10^{-4}	1.4×10^{-4}	8.4×10^{-8}	0.0013	0.015
Total	759	7.3	31	0.0044	0.019	0.0072	0.028	1.7×10^{-5}	0.0066	0.037

a. Totals might differ from sums of values due to rounding.

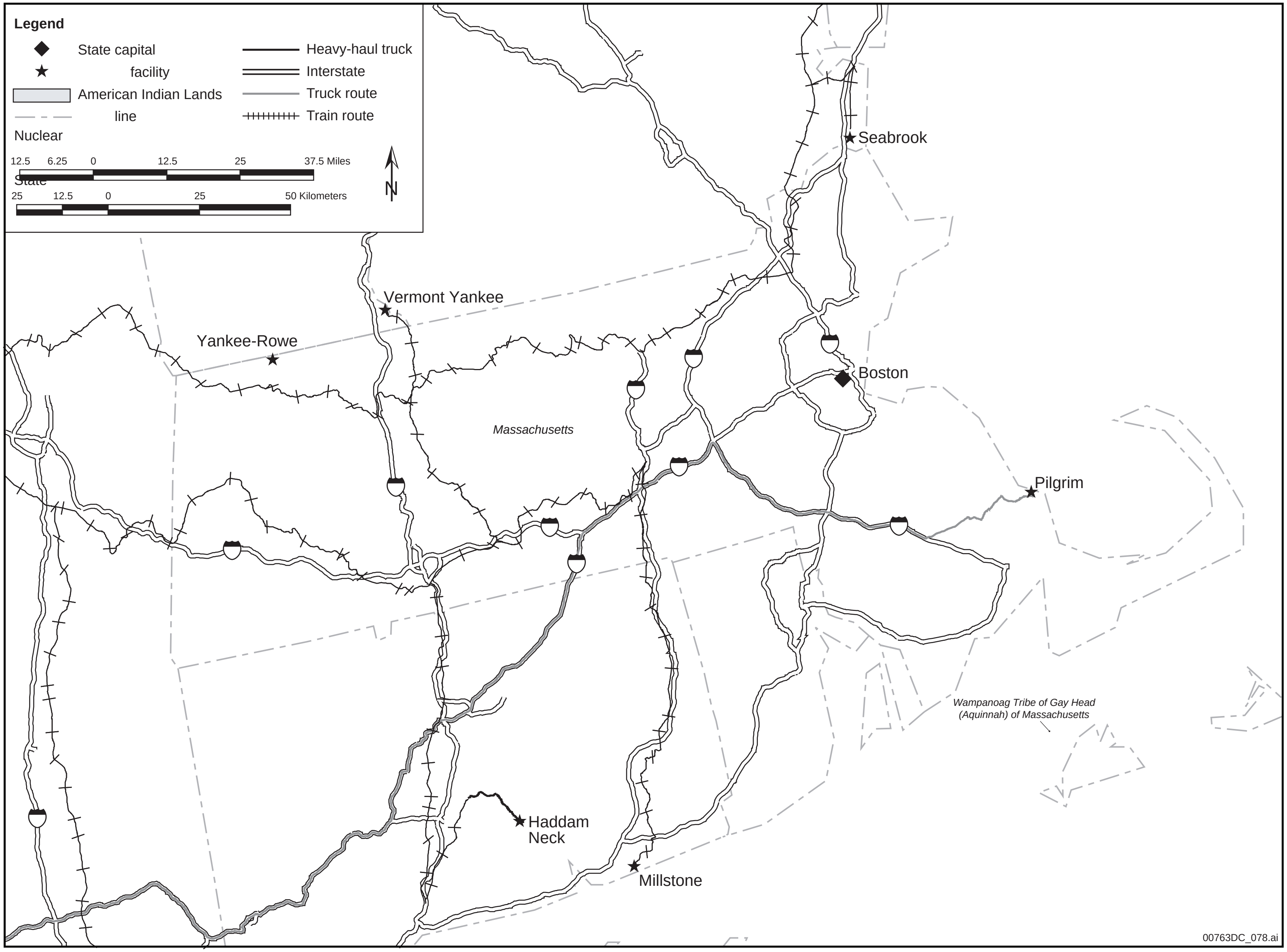

Figure G-21. Representative transportation routes for the Commonwealth of Massachusetts.

Table G-41. Estimated transportation impacts for the State of Michigan.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	132	2.3	20	0.0014	0.012	0.0023	0.013	7.8×10^{-6}	0.0025	0.018
Truck	768	0.66	37	4.0×10^{-4}	0.022	1.4×10^{-4}	7.5×10^{-5}	4.5×10^{-8}	0.0012	0.024
Total	900	2.9	57	0.0018	0.034	0.0024	0.013	7.9×10^{-6}	0.0038	0.042
Mina										
Rail	132	2.3	20	0.0014	0.012	0.0023	0.013	7.8×10^{-6}	0.0025	0.018
Truck	768	0.66	37	4.0×10^{-4}	0.022	1.4×10^{-4}	7.5×10^{-5}	4.5×10^{-8}	0.0012	0.024
Total	900	2.9	57	0.0018	0.034	0.0024	0.013	7.9×10^{-6}	0.0038	0.042

a. Totals might differ from sums of values due to rounding.

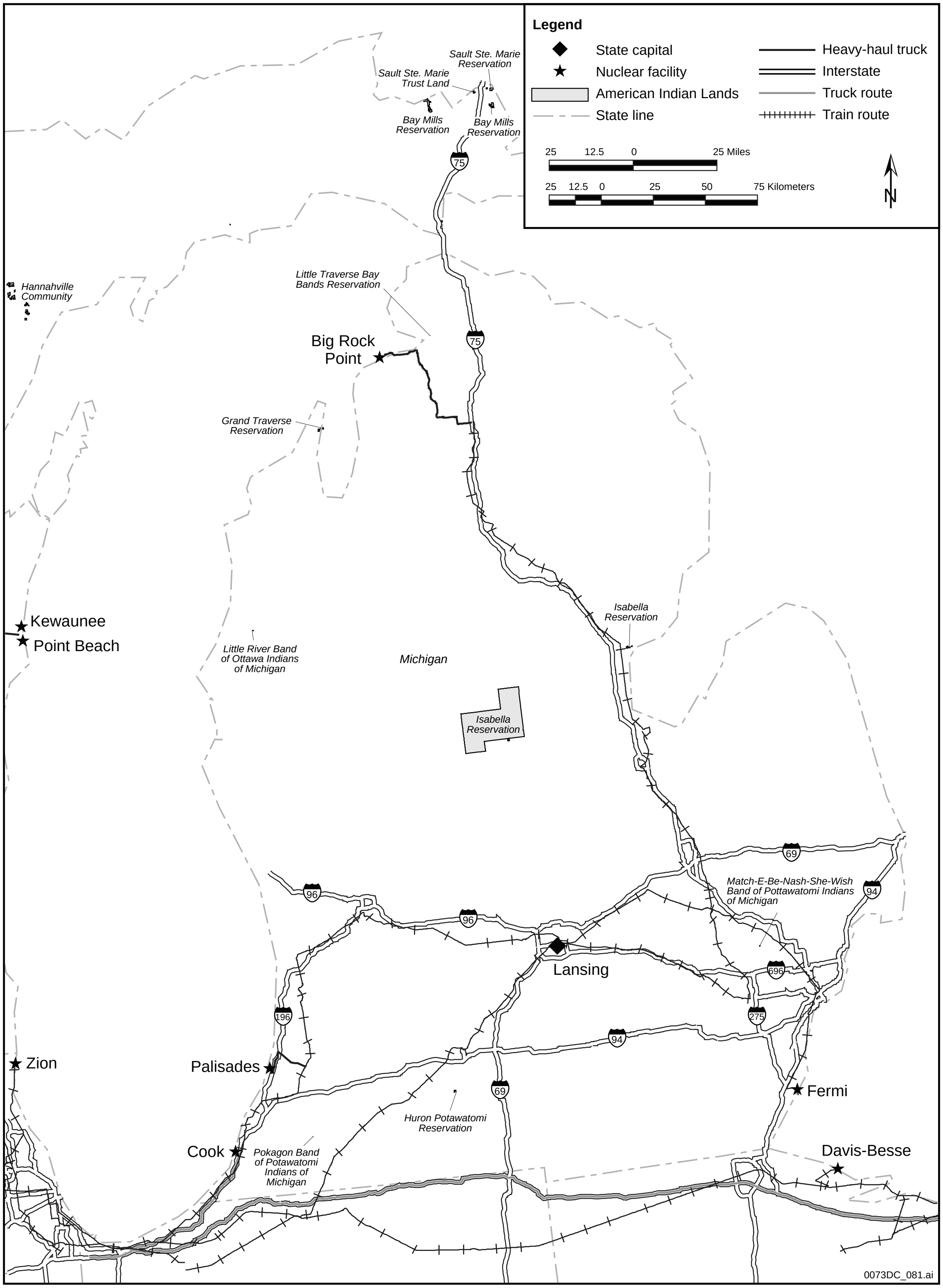

Figure G-22. Representative transportation routes for the State of Michigan.

Table G-42. Estimated transportation impacts for the State of Minnesota.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	153	1.5	14	9.0×10^{-4}	0.0083	0.0021	0.011	6.3×10^{-6}	0.0036	0.015
Truck	37	0.18	0.51	1.1×10^{-4}	3.1×10^{-4}	3.3×10^{-5}	1.2×10^{-5}	7.0×10^{-9}	2.3×10^{-4}	6.7×10^{-4}
Total	190	1.7	14	0.0010	0.0086	0.0021	0.011	6.3×10^{-6}	0.0038	0.016
Mina										
Rail	153	1.5	14	9.0×10^{-4}	0.0083	0.0021	0.011	6.3×10^{-6}	0.0036	0.015
Truck	37	0.18	0.51	1.1×10^{-4}	3.1×10^{-4}	3.3×10^{-5}	1.2×10^{-5}	7.0×10^{-9}	2.3×10^{-4}	6.7×10^{-4}
Total	190	1.7	14	0.0010	0.0086	0.0021	0.011	6.3×10^{-6}	0.0038	0.016

a. Totals might differ from sums of values due to rounding.

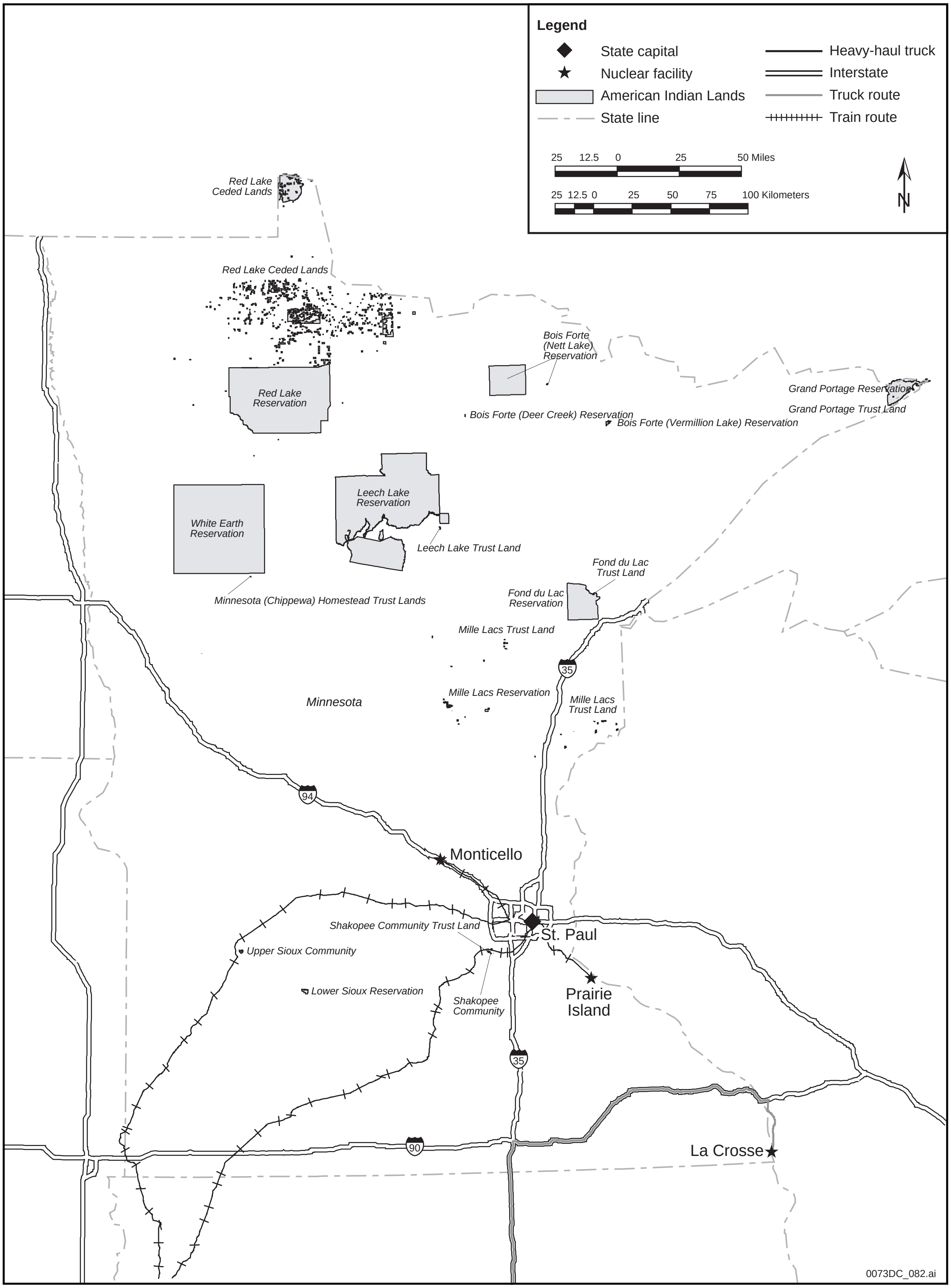

Figure G-23. Representative transportation routes for the State of Minnesota.

Table G-43. Estimated transportation impacts for the State of Mississippi.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	170	1.2	22	7.0×10^{-4}	0.013	7.4×10^{-4}	0.0042	2.5×10^{-6}	0.0026	0.017
Truck	857	3.3	7.2	0.0020	0.0043	8.5×10^{-4}	7.5×10^{-5}	4.5×10^{-8}	0.0030	0.010
Total	1,027	4.5	29	0.0027	0.017	0.0016	0.0043	2.6×10^{-6}	0.0055	0.027
Mina										
Rail	170	1.2	22	7.0×10^{-4}	0.013	7.4×10^{-4}	0.0042	2.5×10^{-6}	0.0026	0.017
Truck	857	3.3	7.2	0.0020	0.0043	8.5×10^{-4}	7.5×10^{-5}	4.5×10^{-8}	0.0030	0.010
Total	1,027	4.5	29	0.0027	0.017	0.0016	0.0043	2.6×10^{-6}	0.0055	0.027

a. Totals might differ from sums of values due to rounding.

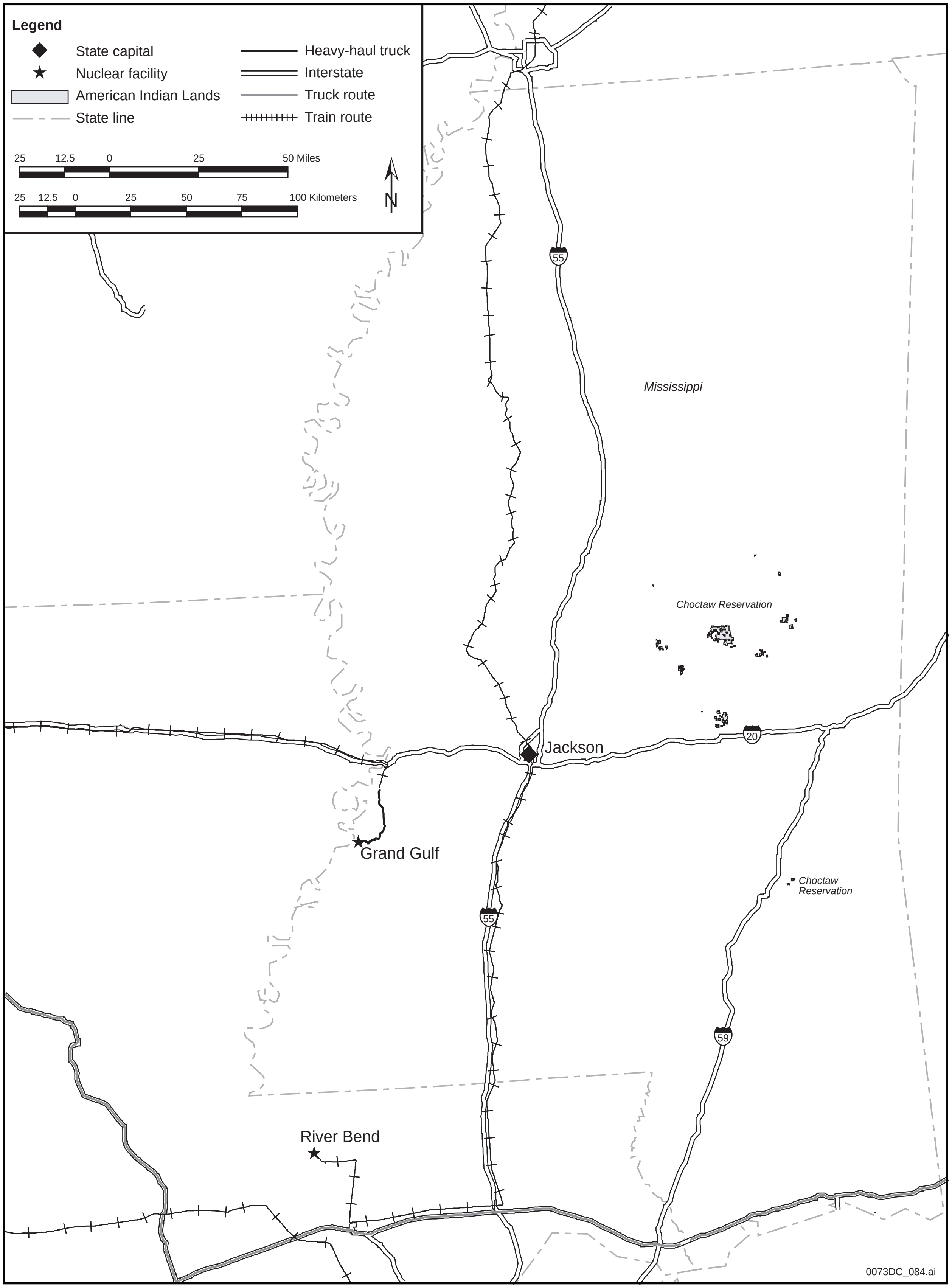

Figure G-24. Representative transportation routes for the State of Mississippi.

Table G-44. Estimated transportation impacts for the State of Missouri.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	3,574	41	140	0.024	0.083	0.052	0.19	1.2×10^{-4}	0.082	0.24
Truck	0	0	0	0	0	0	0	0	0	0
Total	3,574	41	140	0.024	0.083	0.052	0.19	1.2×10^{-4}	0.082	0.24
Mina										
Rail	3,574	41	140	0.024	0.083	0.052	0.19	1.2×10^{-4}	0.082	0.24
Truck	0	0	0	0	0	0	0	0	0	0
Total	3,574	41	140	0.024	0.083	0.052	0.19	1.2×10^{-4}	0.082	0.24

a. Totals might differ from sums of values due to rounding.

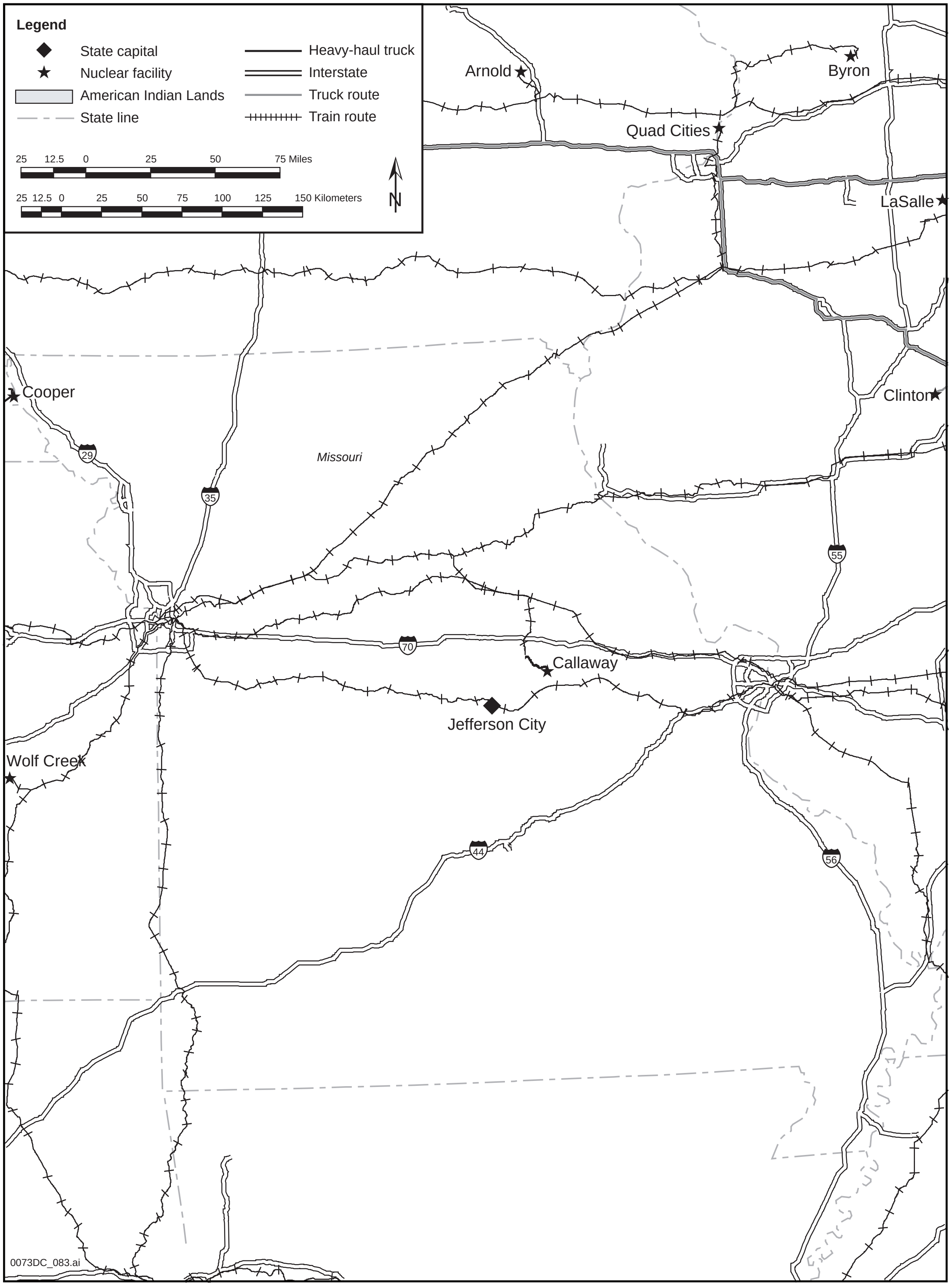

Figure G-25. Representative transportation routes for the State of Missouri.

Table G-45. Estimated transportation impacts for the State of Nebraska.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	6,739	37	400	0.022	0.24	0.052	0.35	2.1×10^{-4}	0.27	0.59
Truck	1,789	30	88	0.018	0.053	0.0042	0.0030	1.8×10^{-6}	0.083	0.16
Total	8,528	67	490	0.040	0.30	0.056	0.35	2.1×10^{-4}	0.35	0.74
Mina										
Rail	6,739	37	400	0.022	0.24	0.052	0.35	2.1×10^{-4}	0.27	0.59
Truck	1,789	30	88	0.018	0.053	0.0042	0.0030	1.8×10^{-6}	0.083	0.16
Total	8,528	67	490	0.040	0.30	0.056	0.35	2.1×10^{-4}	0.35	0.74

a. Totals might differ from sums of values due to rounding.

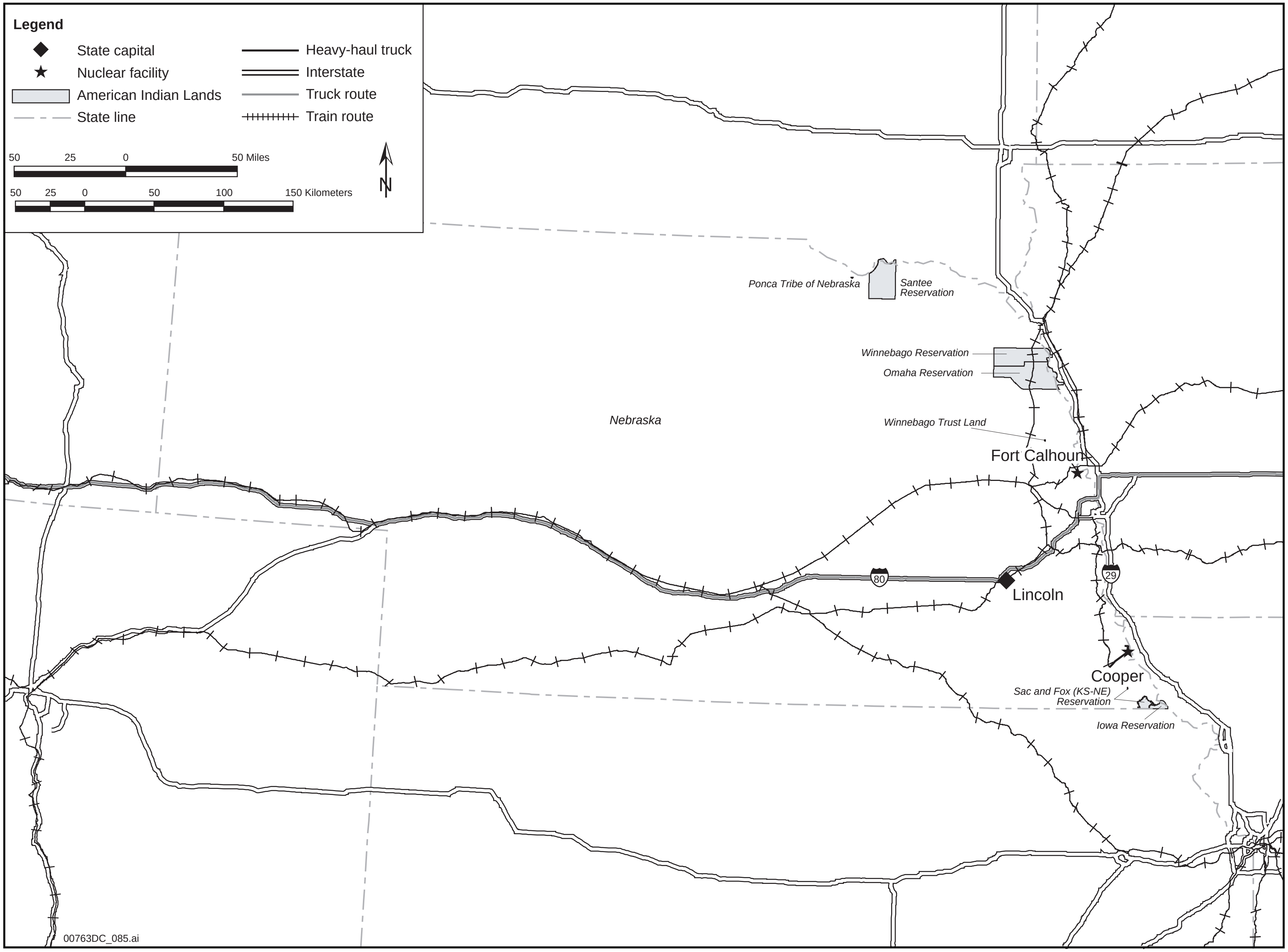

Figure G-26. Representative transportation routes for the State of Nebraska.

Table G-46. Estimated transportation impacts for the State of Nevada.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	9,495	16	680	0.0096	0.41	0.020	0.075	4.5×10^{-5}	0.34	0.78
Truck	2,650	21	95	0.012	0.057	0.0046	0.0032	1.9×10^{-6}	0.050	0.12
Total	12,145	37	770	0.022	0.46	0.024	0.078	4.7×10^{-5}	0.39	0.90
Mina										
Rail	9,495	30	1,500	0.018	0.88	0.037	0.10	6.3×10^{-5}	0.58	1.5
Truck	2,650	21	95	0.012	0.057	0.0046	0.0032	1.9×10^{-6}	0.050	0.12
Total	12,145	50	1,600	0.030	0.94	0.042	0.11	6.5×10^{-5}	0.63	1.6

a. Totals might differ from sums of values due to rounding.

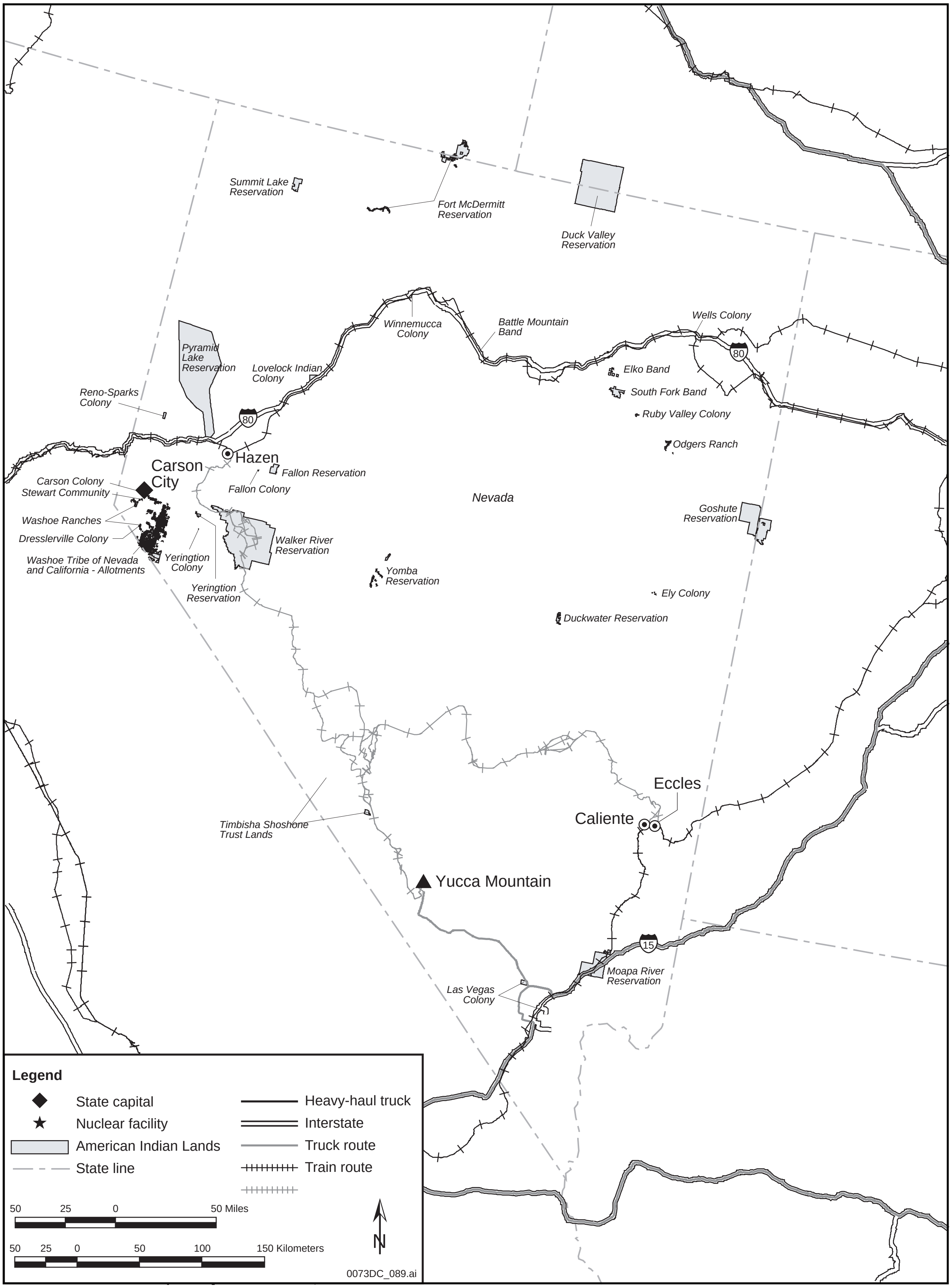

Figure G-27. Representative transportation routes for the State of Nevada.

Table G-47. Estimated transportation impacts for the State of New Hampshire.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	110	0.41	3.4	2.5×10^{-4}	0.0020	5.6×10^{-4}	0.0023	1.4×10^{-6}	4.0×10^{-4}	0.0032
Truck	0	0	0	0	0	0	0	0	0	0
Total	110	0.41	3.4	2.5×10^{-4}	0.0020	5.6×10^{-4}	0.0023	1.4×10^{-6}	4.0×10^{-4}	0.0032
Mina										
Rail	110	0.41	3.4	2.5×10^{-4}	0.0020	5.6×10^{-4}	0.0023	1.4×10^{-6}	4.0×10^{-4}	0.0032
Truck	0	0	0	0	0	0	0	0	0	0
Total	110	0.41	3.4	2.5×10^{-4}	0.0020	5.6×10^{-4}	0.0023	1.4×10^{-6}	4.0×10^{-4}	0.0032

a. Totals might differ from sums of values due to rounding.

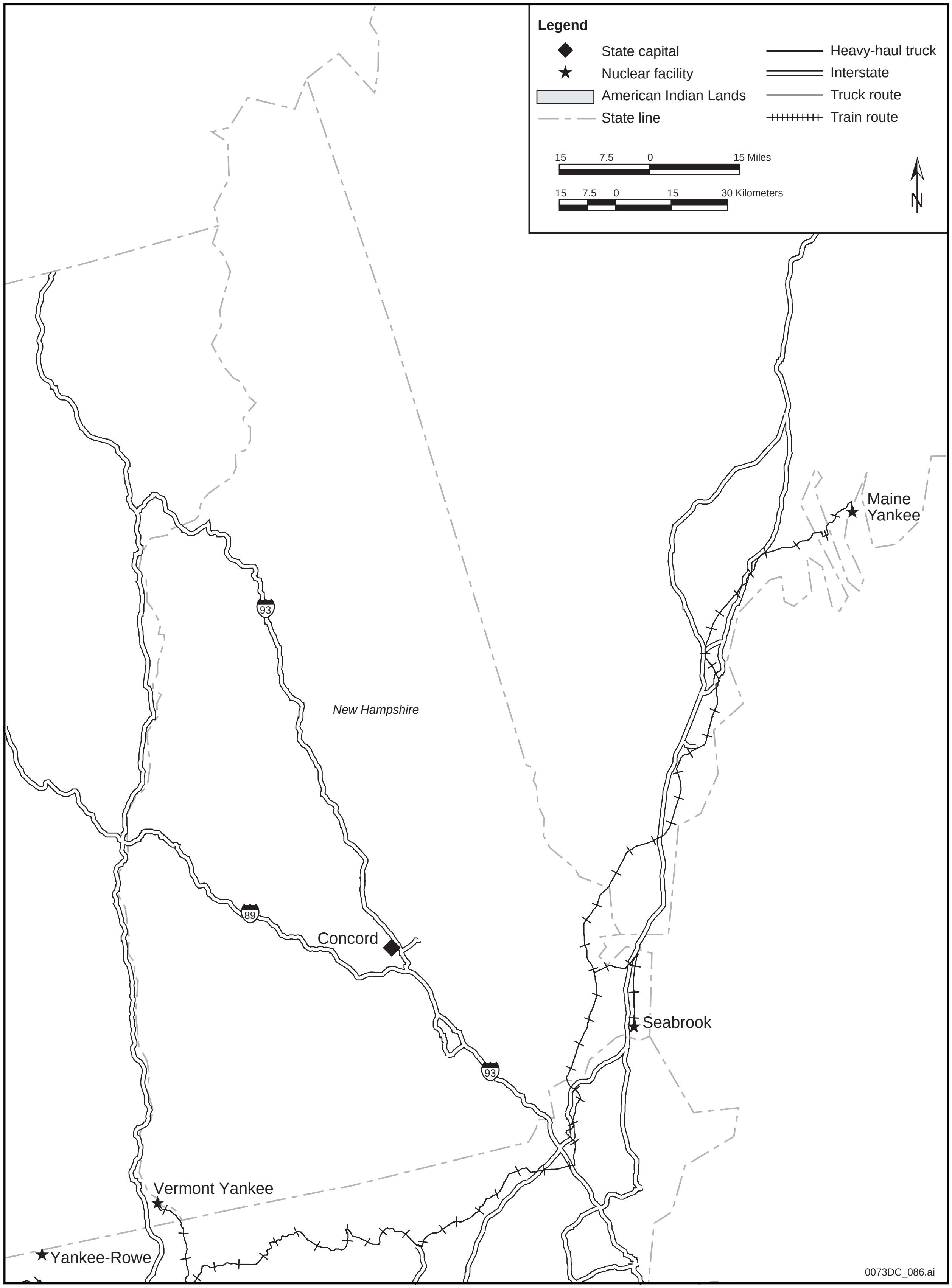

Figure G-28. Representative transportation routes for the State of New Hampshire.

Table G-48. Estimated transportation impacts for the State of New Jersey.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	276	8.2	56	0.0049	0.033	0.0066	0.031	1.9×10^{-5}	0.0031	0.048
Truck	0	0	0	0	0	0	0	0	0	0
Total	276	8.2	56	0.0049	0.033	0.0066	0.031	1.9×10^{-5}	0.0031	0.048
Mina										
Rail	276	8.2	56	0.0049	0.033	0.0066	0.031	1.9×10^{-5}	0.0031	0.048
Truck	0	0	0	0	0	0	0	0	0	0
Total	276	8.2	56	0.0049	0.033	0.0066	0.031	1.9×10^{-5}	0.0031	0.048

a. Totals might differ from sums of values due to rounding.

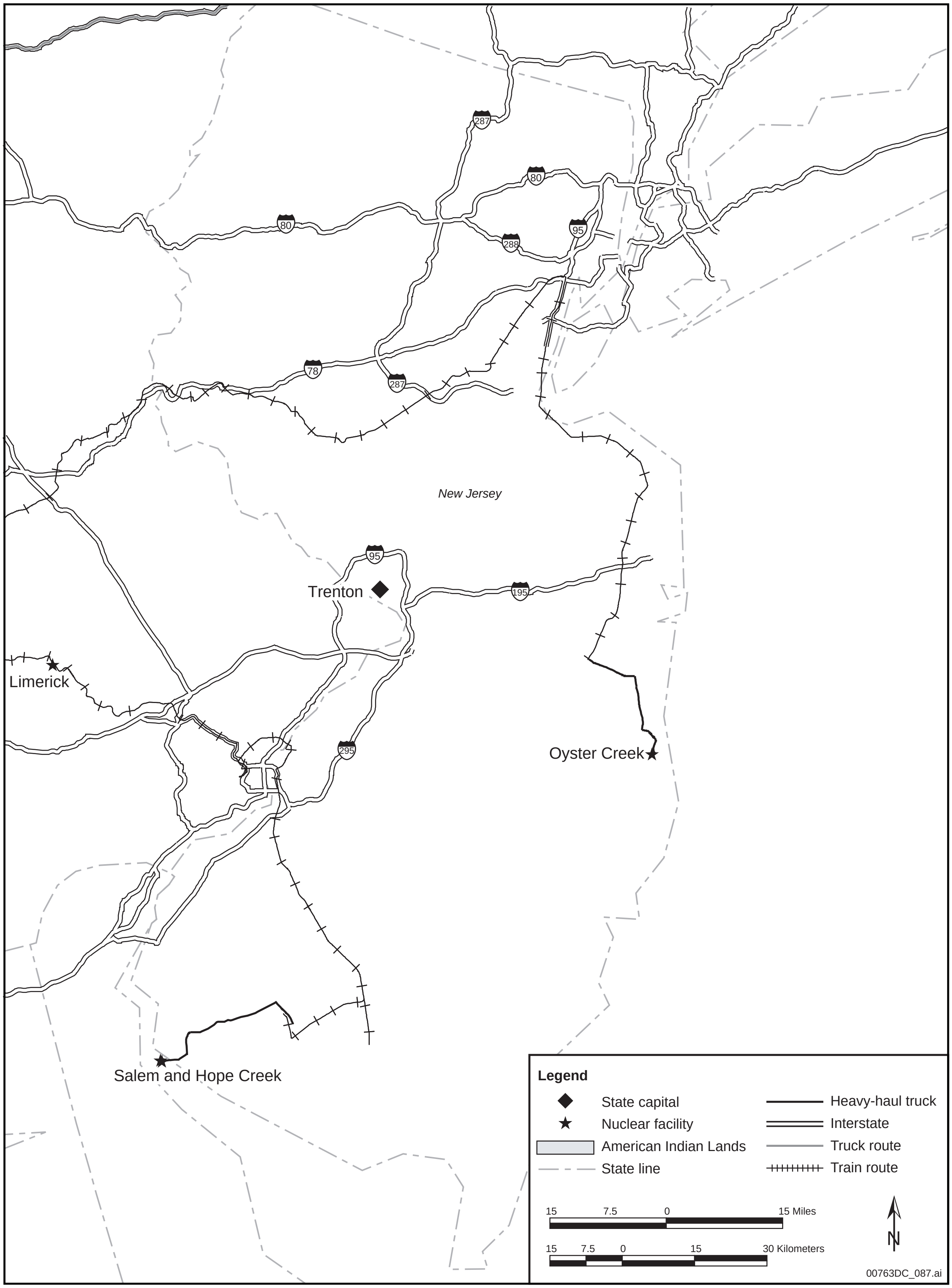

Figure G-29. Representative transportation routes for the State of New Jersey.

Table G-49. Estimated transportation impacts for the State of New Mexico.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	257	0.24	6.0	1.5×10^{-4}	0.0036	3.6×10^{-4}	0.0014	8.6×10^{-7}	0.0043	0.0084
Truck	857	13	34	0.0078	0.020	0.0027	5.7×10^{-4}	3.4×10^{-7}	0.029	0.060
Total	1,114	13	40	0.0080	0.024	0.0031	0.0020	1.2×10^{-6}	0.033	0.069
Mina										
Rail	257	0.17	4.8	9.9×10^{-5}	0.0029	2.5×10^{-4}	9.8×10^{-4}	5.9×10^{-7}	0.0034	0.0067
Truck	857	13	34	0.0078	0.020	0.0027	5.7×10^{-4}	3.4×10^{-7}	0.029	0.060
Total	1,114	13	39	0.0079	0.023	0.0030	0.0015	9.3×10^{-7}	0.033	0.067

a. Totals might differ from sums of values due to rounding.

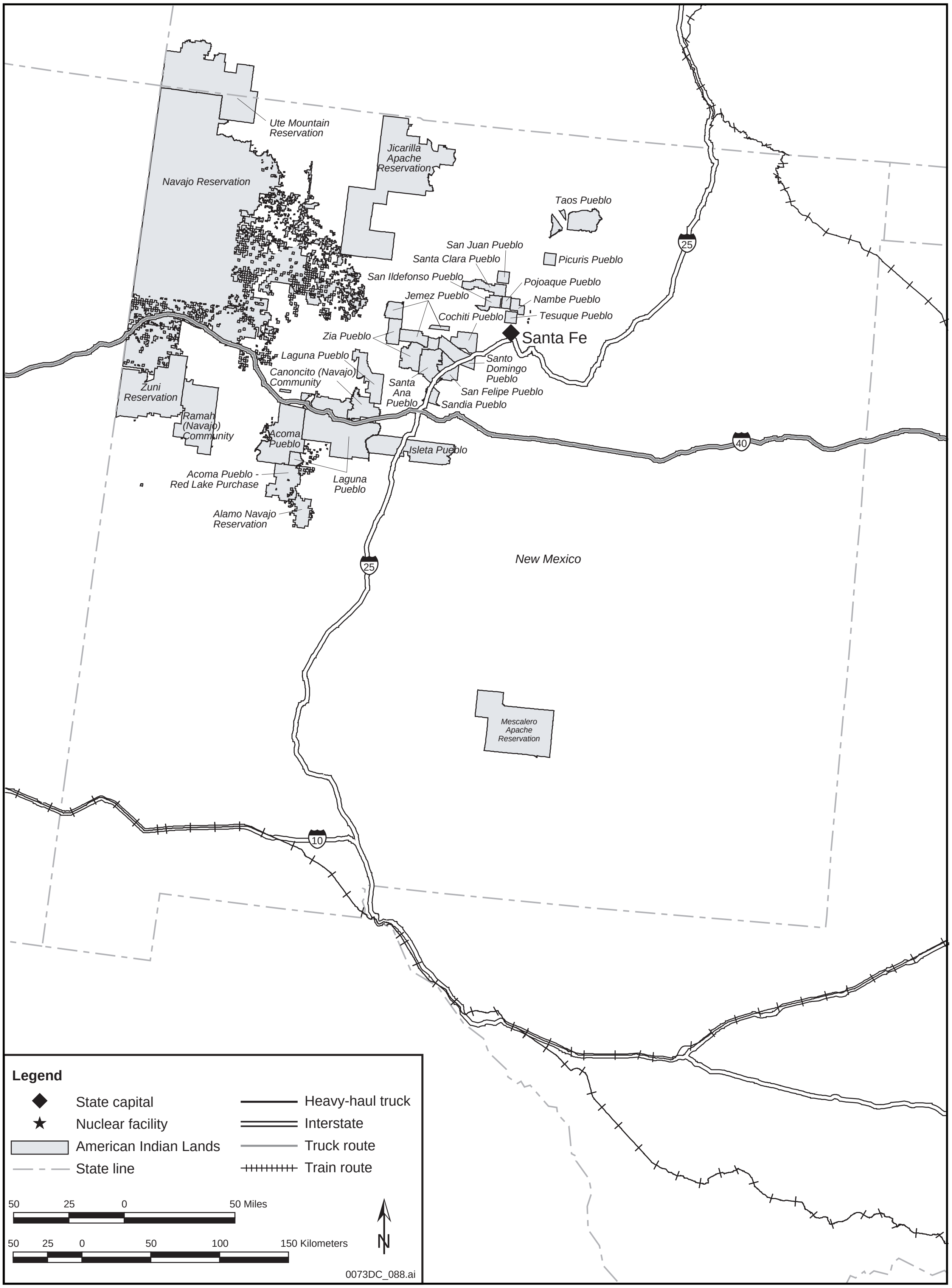

Figure G-30. Representative transportation routes for the State of New Mexico.

Table G-50. Estimated transportation impacts for the State of New York.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	827	14	85	0.0084	0.051	0.018	0.083	5.0×10^{-5}	0.029	0.11
Truck	657	5.4	23	0.0032	0.014	0.0020	0.0013	7.7×10^{-7}	0.0072	0.026
Total	1,484	19	110	0.012	0.065	0.020	0.085	5.1×10^{-5}	0.036	0.13
Mina										
Rail	827	14	85	0.0084	0.051	0.018	0.083	5.0×10^{-5}	0.029	0.11
Truck	657	5.4	23	0.0032	0.014	0.0020	0.0013	7.7×10^{-7}	0.0072	0.026
Total	1,484	19	110	0.012	0.065	0.020	0.085	5.1×10^{-5}	0.036	0.13

a. Totals might differ from sums of values due to rounding.

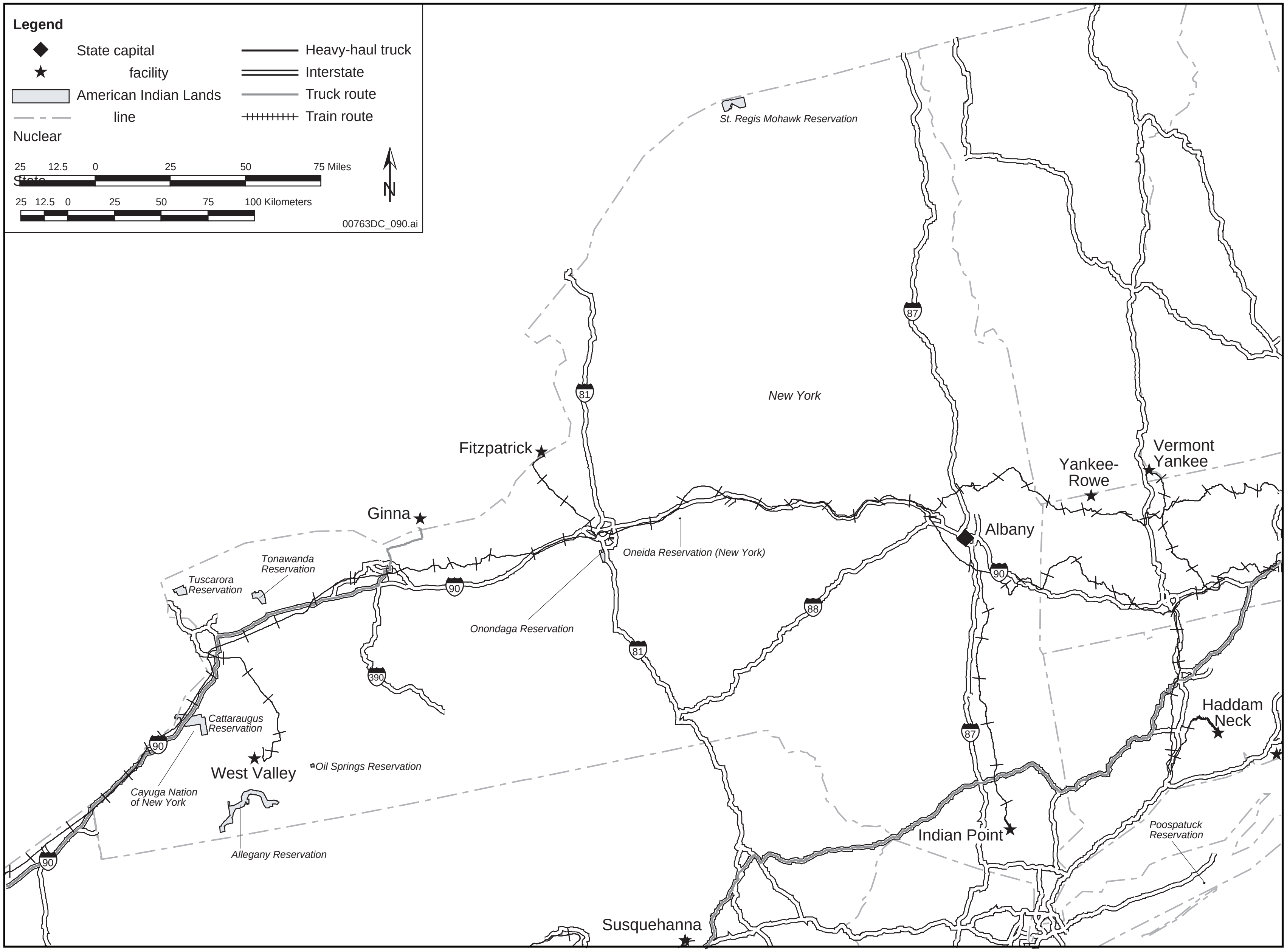

Figure G-31. Representative transportation routes for the State of New York.

Table G-51. Estimated transportation impacts for the State of North Carolina.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	502	7.1	35	0.0042	0.021	0.011	0.045	2.7×10^{-5}	0.0094	0.046
Truck	0	0	0	0	0	0	0	0	0	0
Total	502	7.1	35	0.0042	0.021	0.011	0.045	2.7×10^{-5}	0.0094	0.046
Mina										
Rail	502	7.1	35	0.0042	0.021	0.011	0.045	2.7×10^{-5}	0.0094	0.046
Truck	0	0	0	0	0	0	0	0	0	0
Total	502	7.1	35	0.0042	0.021	0.011	0.045	2.7×10^{-5}	0.0094	0.046

a. Totals might differ from sums of values due to rounding.

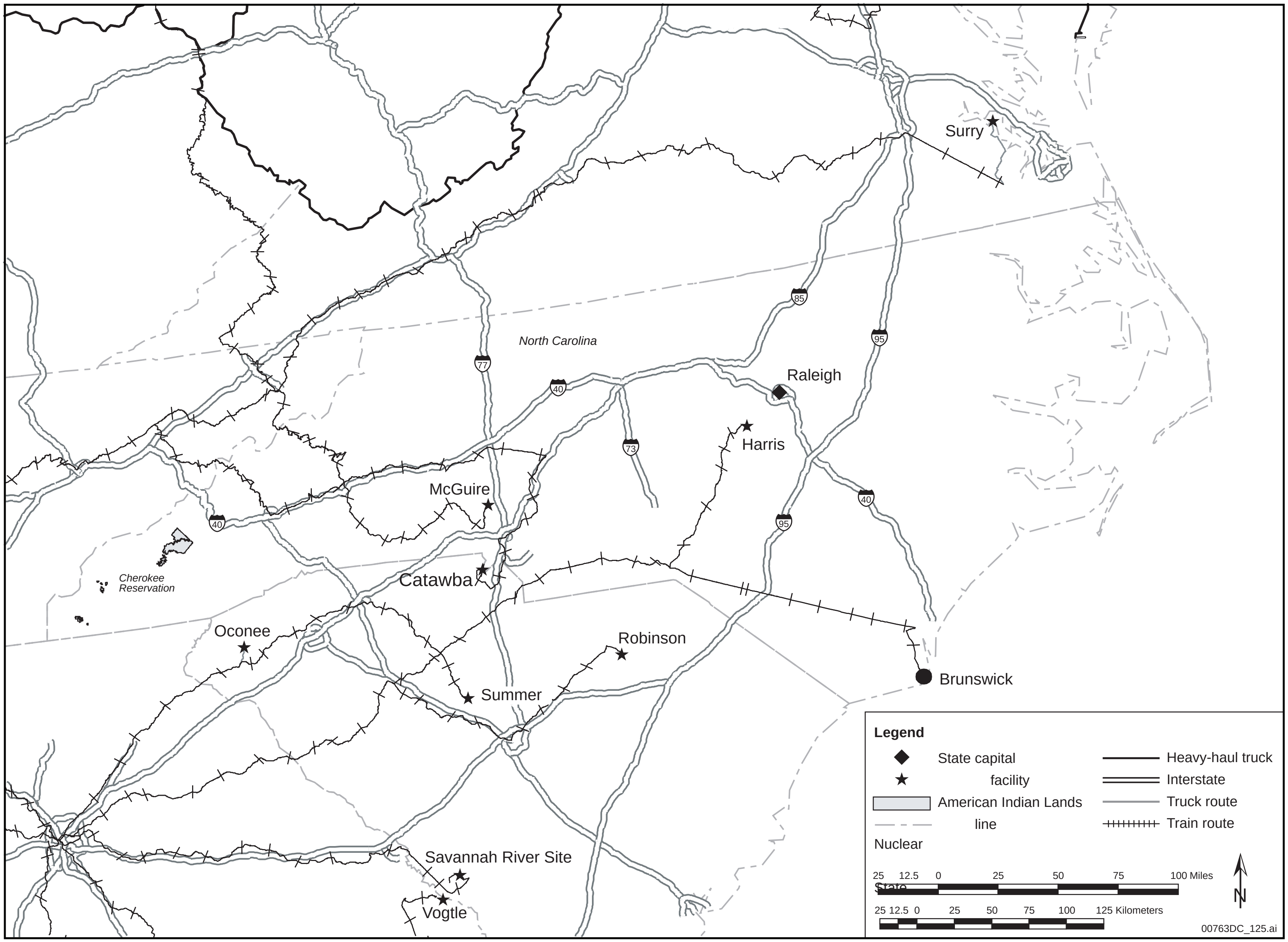

Figure G-32. Representative transportation routes for the State of North Carolina.

Table G-52. Estimated transportation impacts for the State of Ohio.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	2,314	37	100	0.022	0.062	0.049	0.25	1.5×10^{-4}	0.058	0.19
Truck	657	9.8	18	0.0059	0.011	0.0031	9.6×10^{-4}	5.8×10^{-7}	0.0085	0.028
Total	2,971	47	120	0.028	0.073	0.052	0.25	1.5×10^{-4}	0.066	0.22
Mina										
Rail	2,314	37	100	0.022	0.062	0.049	0.25	1.5×10^{-4}	0.058	0.19
Truck	657	9.8	18	0.0059	0.011	0.0031	9.6×10^{-4}	5.8×10^{-7}	0.0085	0.028
Total	2,971	47	120	0.028	0.073	0.052	0.25	1.5×10^{-4}	0.066	0.22

a. Totals might differ from sums of values due to rounding.

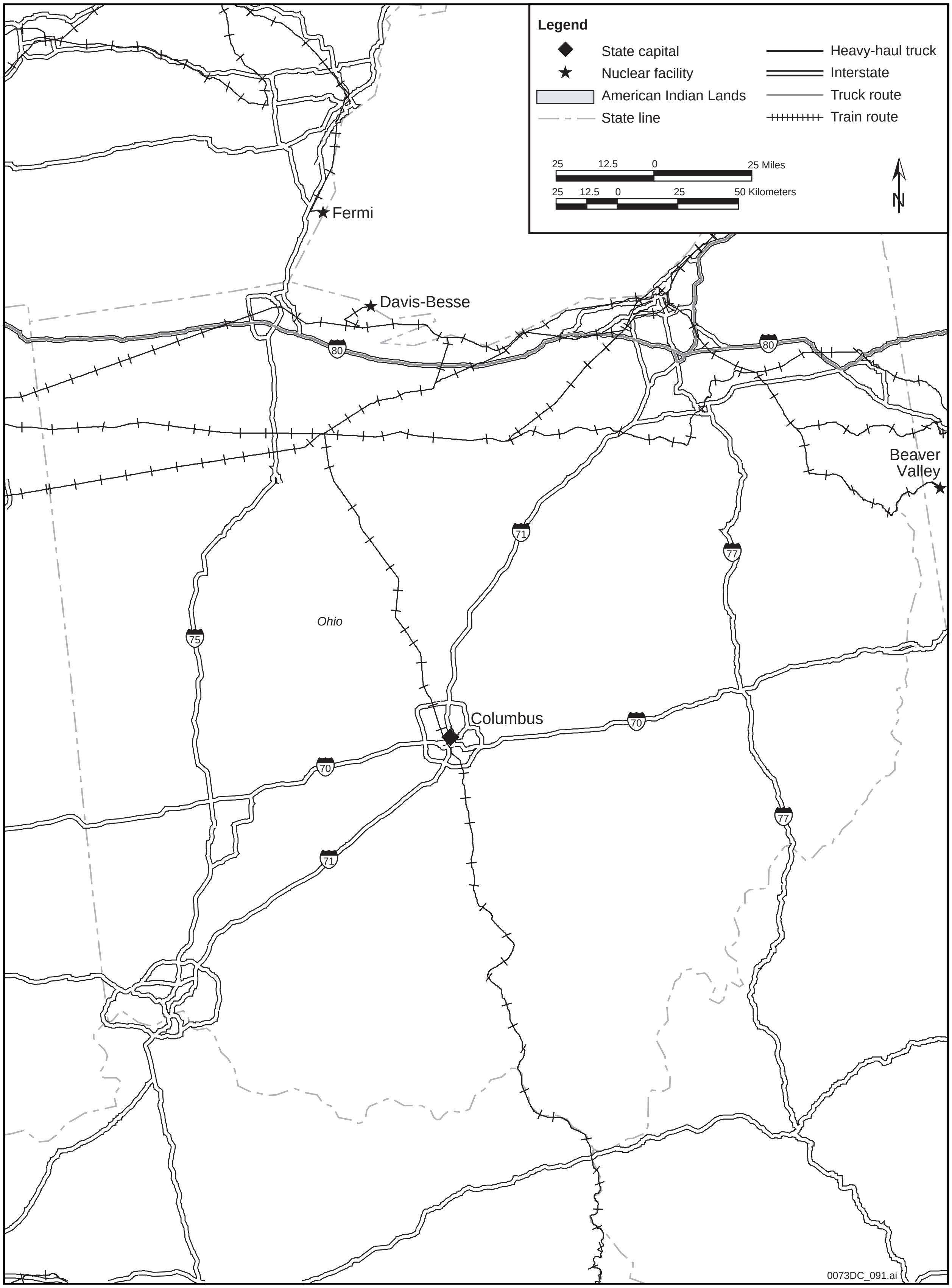

Figure G-33. Representative transportation routes for the State of Ohio.

Table G-53. Estimated transportation impacts for the State of Oklahoma.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	227	0.61	4.9	3.7×10^{-4}	0.0029	0.0010	0.0048	2.9×10^{-6}	0.0033	0.0076
Truck	857	12	26	0.0069	0.015	0.0035	0.0018	1.1×10^{-6}	0.024	0.050
Total	1,084	12	31	0.0073	0.018	0.0045	0.0066	3.9×10^{-6}	0.027	0.057
Mina										
Rail	227	0.61	4.9	3.7×10^{-4}	0.0029	0.0010	0.0048	2.9×10^{-6}	0.0033	0.0076
Truck	857	12	26	0.0069	0.015	0.0035	0.0018	1.1×10^{-6}	0.024	0.050
Total	1,084	12	31	0.0073	0.018	0.0045	0.0066	3.9×10^{-6}	0.027	0.057

a. Totals might differ from sums of values due to rounding.

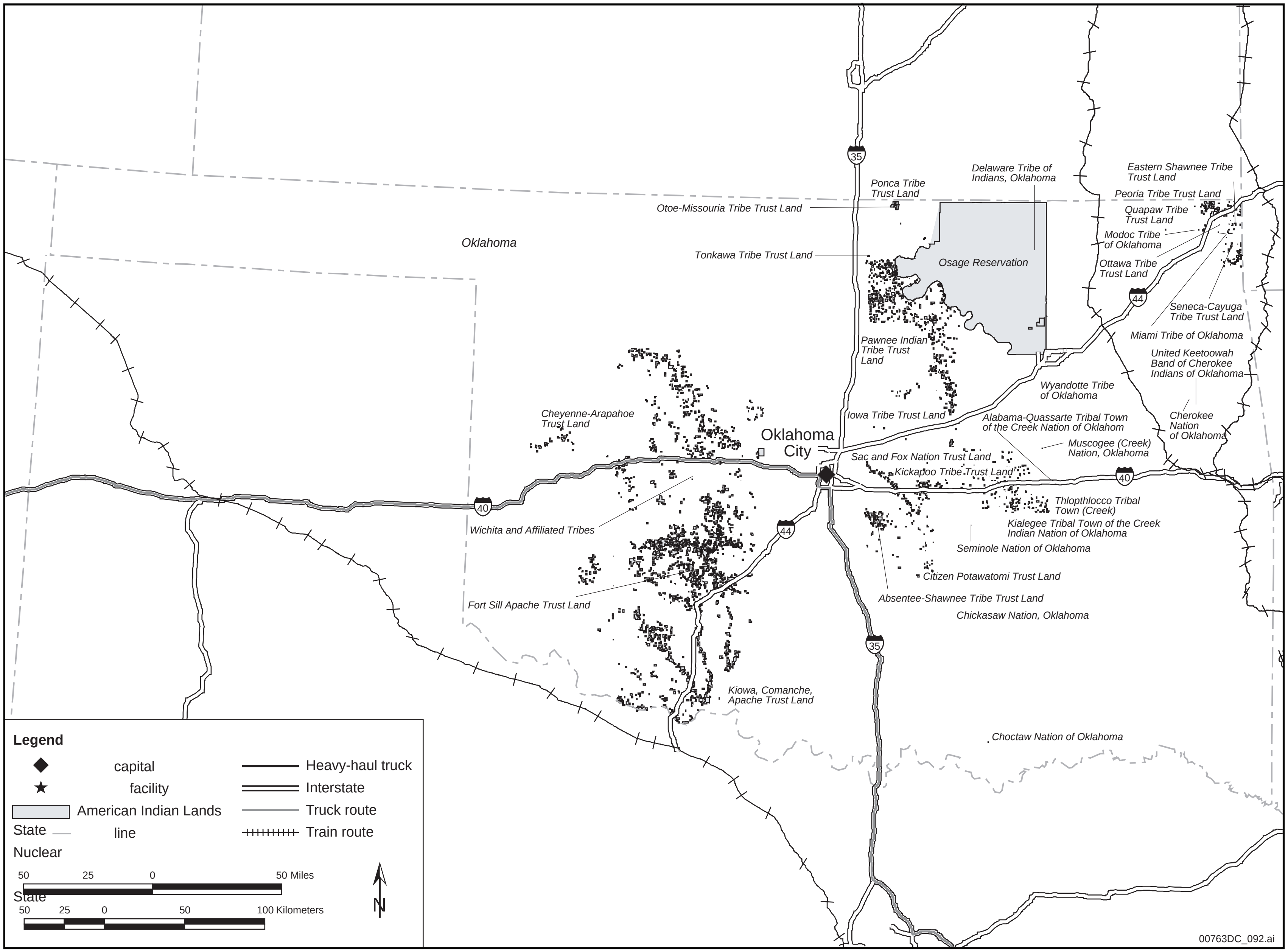

Figure G-34. Representative transportation routes for the State of Oklahoma.

Table G-54. Estimated transportation impacts for the State of Oregon.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	1,307	7.7	33	0.0046	0.020	0.0091	0.012	7.3×10^{-6}	0.025	0.058
Truck	3	0.024	0.067	1.5×10^{-5}	4.0×10^{-5}	5.7×10^{-6}	2.3×10^{-6}	1.4×10^{-9}	8.5×10^{-5}	1.5×10^{-4}
Total	1,310	7.7	33	0.0046	0.020	0.0091	0.012	7.3×10^{-6}	0.025	0.058
Mina										
Rail	1,307	9.4	53	0.0056	0.032	0.012	0.016	9.3×10^{-6}	0.042	0.091
Truck	3	0.024	0.067	1.5×10^{-5}	4.0×10^{-5}	5.7×10^{-6}	2.3×10^{-6}	1.4×10^{-9}	8.5×10^{-5}	1.5×10^{-4}
Total	1,310	9.4	53	0.0056	0.032	0.012	0.016	9.3×10^{-6}	0.042	0.091

a. Totals might differ from sums of values due to rounding.

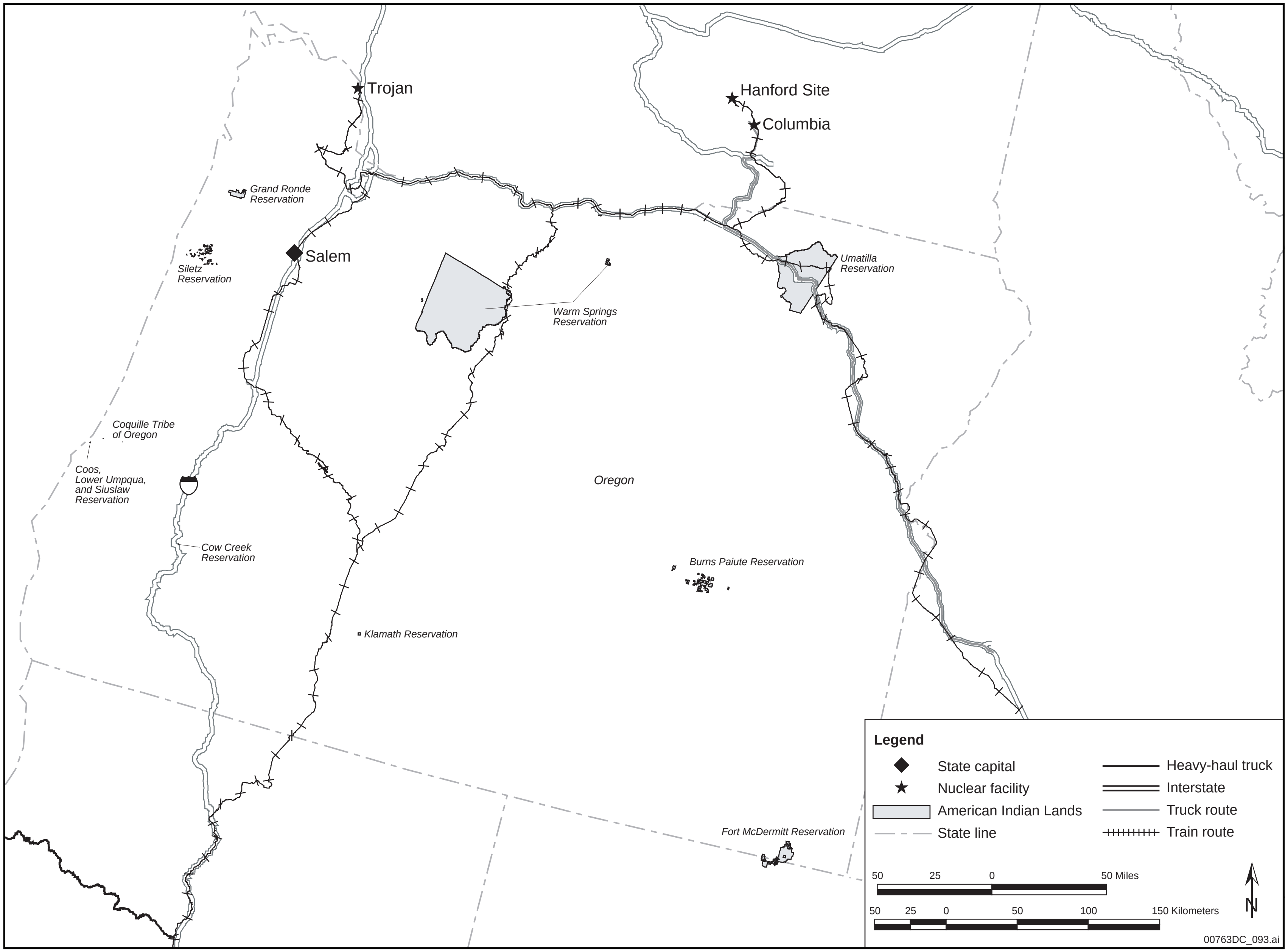

Figure G-35. Representative transportation routes for the State of Oregon.

Table G-55. Estimated transportation impacts for the Commonwealth of Pennsylvania.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	2,036	39	130	0.023	0.080	0.047	0.24	1.4×10^{-4}	0.042	0.19
Truck	657	6.1	15	0.0037	0.0087	0.0012	0.0012	7.1×10^{-7}	0.013	0.027
Total	2,693	45	150	0.027	0.089	0.048	0.24	1.4×10^{-4}	0.056	0.22
Mina										
Rail	2,036	39	130	0.023	0.080	0.047	0.24	1.4×10^{-4}	0.042	0.19
Truck	657	6.1	15	0.0037	0.0087	0.0012	0.0012	7.1×10^{-7}	0.013	0.027
Total	2,693	45	150	0.027	0.089	0.048	0.24	1.4×10^{-4}	0.056	0.22

a. Totals might differ from sums of values due to rounding.

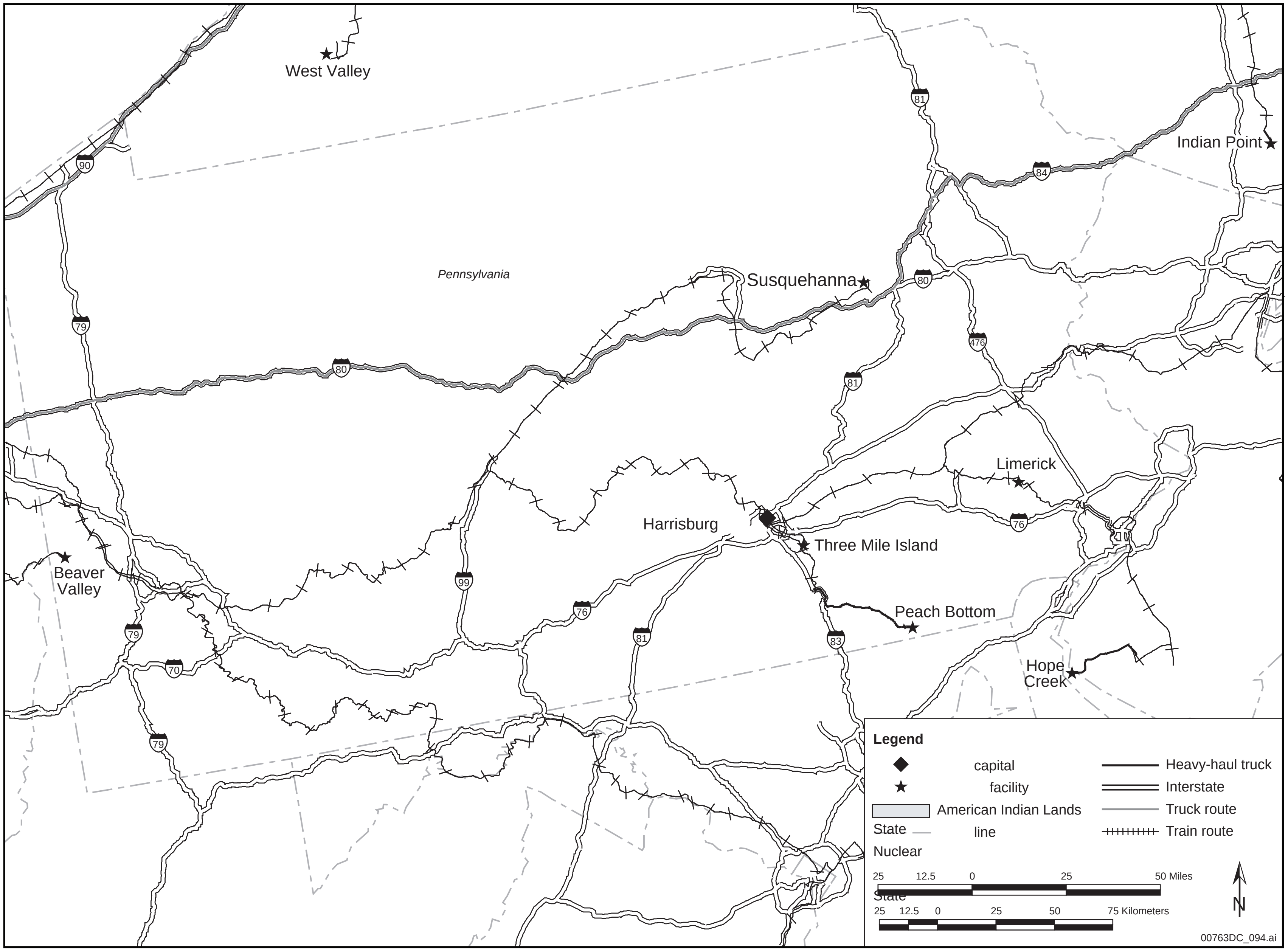

Figure G-36. Representative transportation routes for the Commonwealth of Pennsylvania.

Table G-56. Estimated transportation impacts for the State of South Carolina.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	1,365	4.6	93	0.0027	0.056	0.0035	0.015	8.8×10^{-6}	0.0083	0.070
Truck	0	0	0	0	0	0	0	0	0	0
Total	1,365	4.6	93	0.0027	0.056	0.0035	0.015	8.8×10^{-6}	0.0083	0.070
Mina										
Rail	1,365	4.6	93	0.0027	0.056	0.0035	0.015	8.8×10^{-6}	0.0083	0.070
Truck	0	0	0	0	0	0	0	0	0	0
Total	1,365	4.6	93	0.0027	0.056	0.0035	0.015	8.8×10^{-6}	0.0083	0.070

a. Totals might differ from sums of values due to rounding.

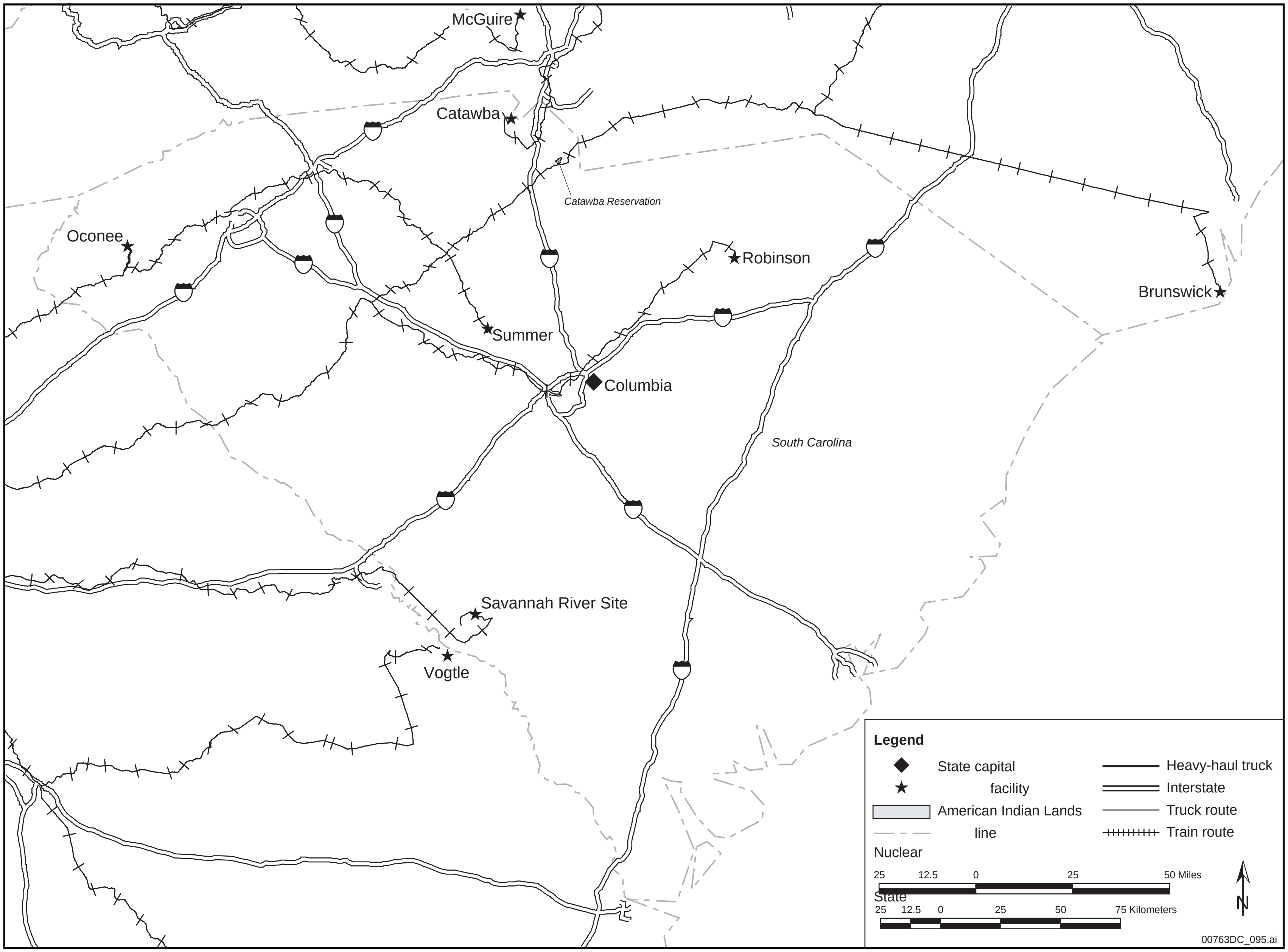

Figure G-37. Representative transportation routes for the State of South Carolina.

Table G-57. Estimated transportation impacts for the State of South Dakota.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	44	0.0045	0.081	2.7×10^{-6}	4.9×10^{-5}	8.1×10^{-6}	5.6×10^{-5}	3.4×10^{-8}	5.6×10^{-5}	1.2×10^{-4}
Truck	0	0	0	0	0	0	0	0	0	0
Total	44	0.0045	0.081	2.7×10^{-6}	4.9×10^{-5}	8.1×10^{-6}	5.6×10^{-5}	3.4×10^{-8}	5.6×10^{-5}	1.2×10^{-4}
Mina										
Rail	44	0.0045	0.081	2.7×10^{-6}	4.9×10^{-5}	8.1×10^{-6}	5.6×10^{-5}	3.4×10^{-8}	5.6×10^{-5}	1.2×10^{-4}
Truck	0	0	0	0	0	0	0	0	0	0
Total	44	0.0045	0.081	2.7×10^{-6}	4.9×10^{-5}	8.1×10^{-6}	5.6×10^{-5}	3.4×10^{-8}	5.6×10^{-5}	1.2×10^{-4}

a. Totals might differ from sums of values due to rounding.

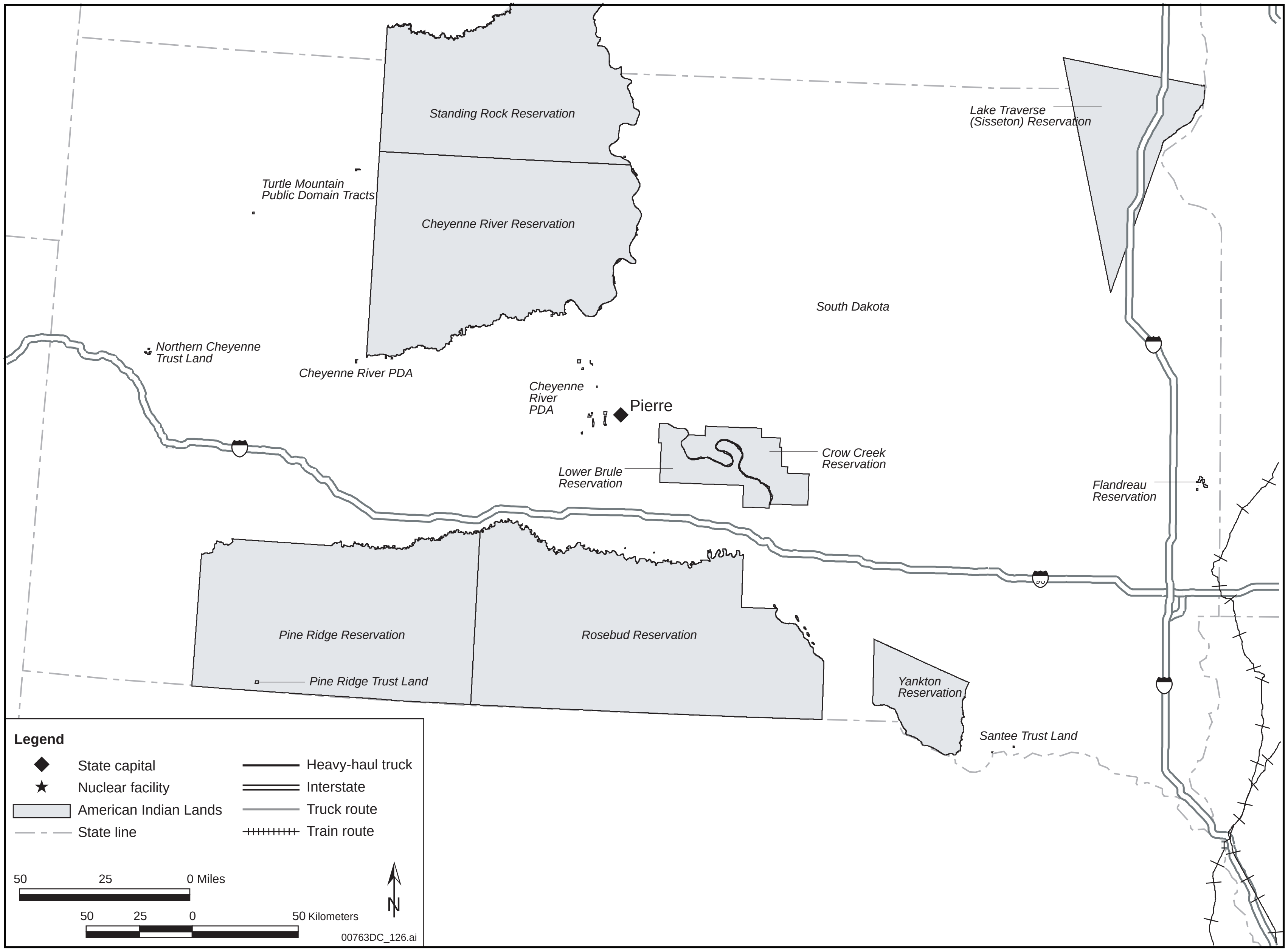

Figure G-38. Representative transportation routes for the State of South Dakota.

Table G-58. Estimated transportation impacts for the State of Tennessee.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	2,663	29	70	0.018	0.042	0.039	0.12	7.1×10^{-5}	0.040	0.14
Truck	0	0	0	0	0	0	0	0	0	0
Total	2,663	29	70	0.018	0.042	0.039	0.12	7.1×10^{-5}	0.040	0.14
Mina										
Rail	2,663	29	70	0.018	0.042	0.039	0.12	7.1×10^{-5}	0.040	0.14
Truck	0	0	0	0	0	0	0	0	0	0
Total	2,663	29	70	0.018	0.042	0.039	0.12	7.1×10^{-5}	0.040	0.14

a. Totals might differ from sums of values due to rounding.

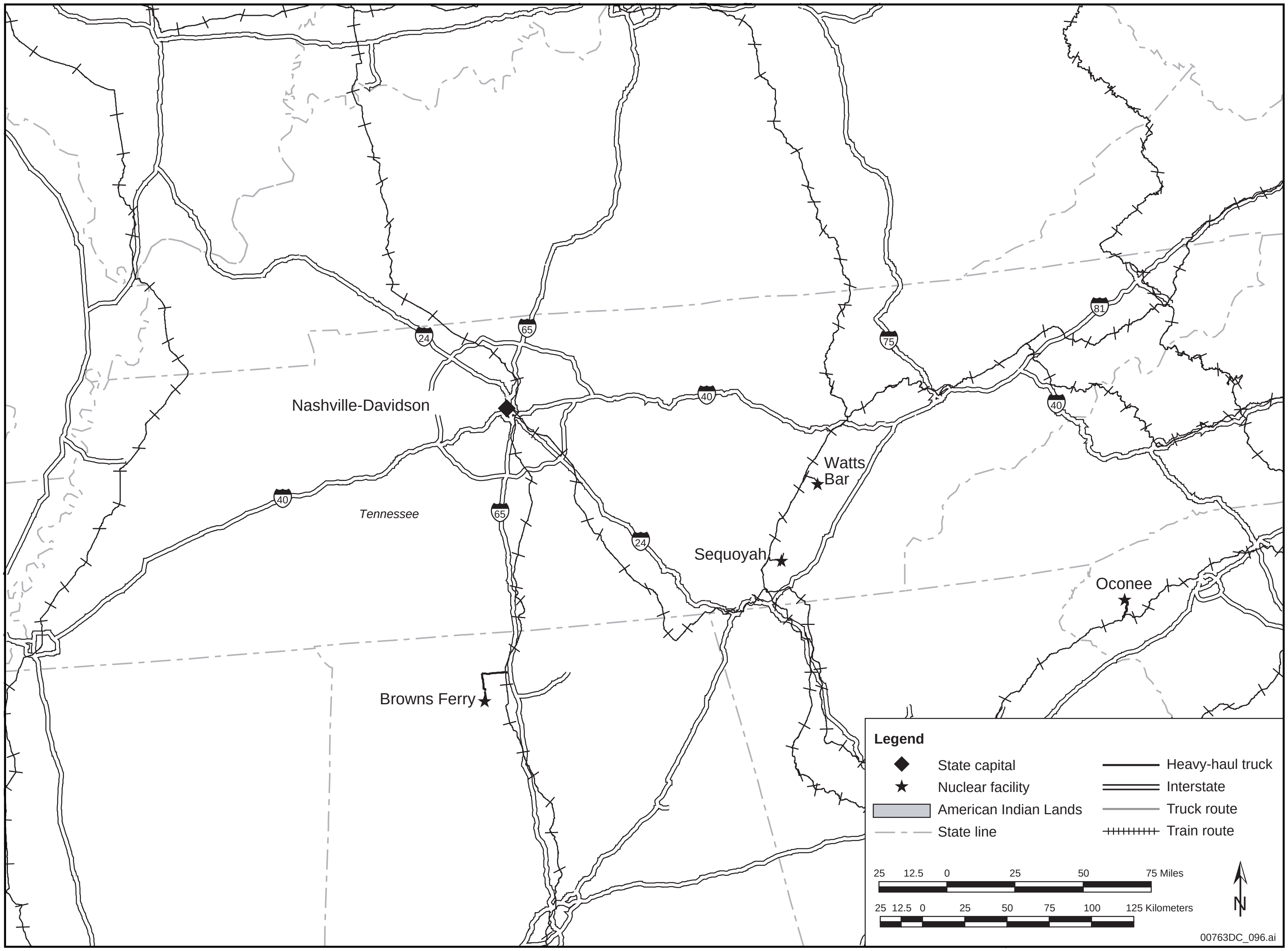

Figure G-39. Representative transportation routes for the State of Tennessee.

Table G-59. Estimated transportation impacts for the State of Texas.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	357	15	41	0.0087	0.025	0.020	0.076	4.6×10^{-5}	0.021	0.074
Truck	857	30	39	0.018	0.023	0.019	0.021	1.2×10^{-5}	0.035	0.096
Total	1,214	44	80	0.027	0.048	0.039	0.097	5.8×10^{-5}	0.056	0.17
Mina										
Rail	357	12	39	0.0073	0.023	0.017	0.064	3.8×10^{-5}	0.019	0.066
Truck	857	30	39	0.018	0.023	0.019	0.021	1.2×10^{-5}	0.035	0.096
Total	1,214	42	78	0.025	0.047	0.035	0.085	5.1×10^{-5}	0.055	0.16

a. Totals might differ from sums of values due to rounding.

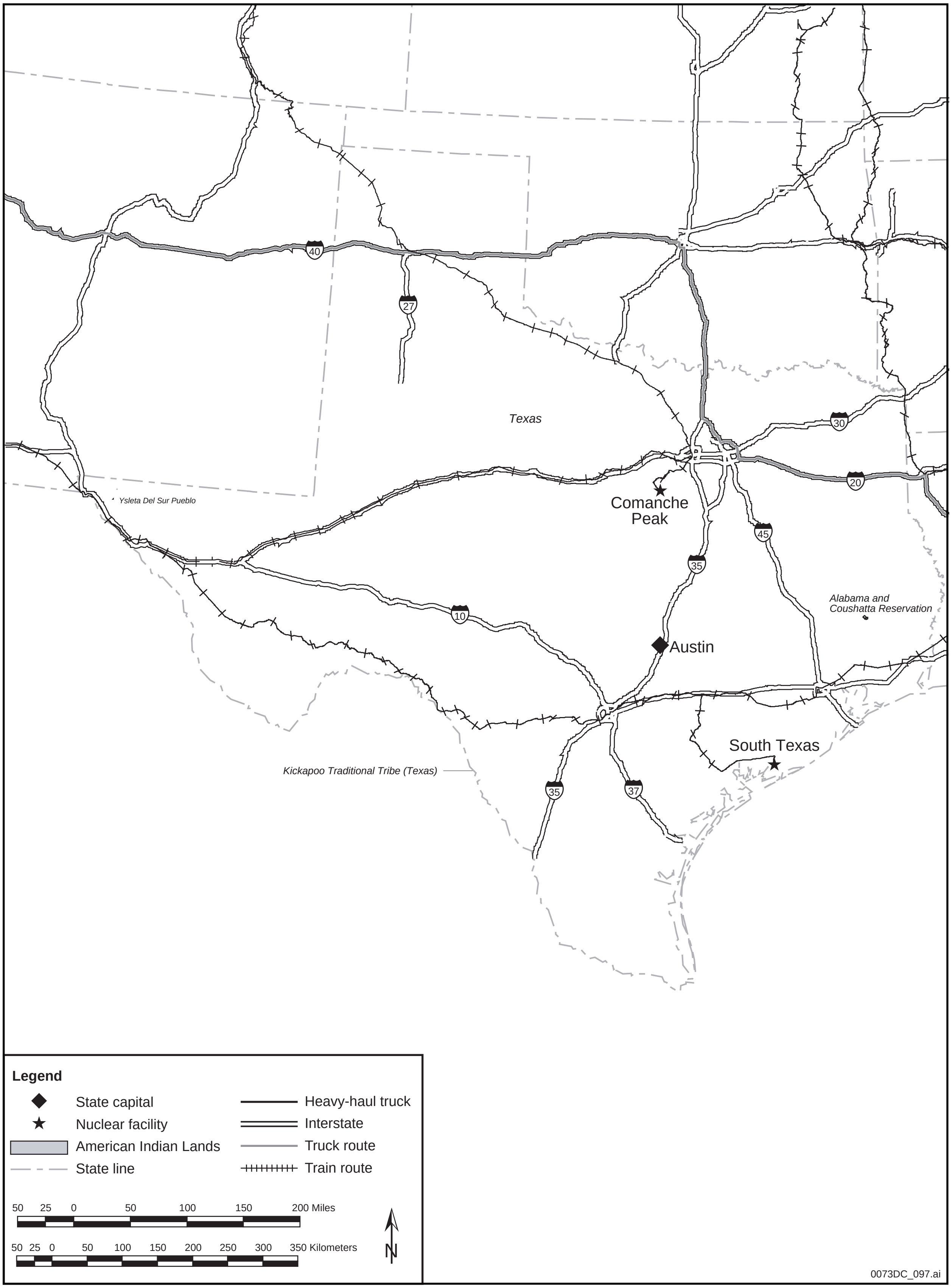

Figure G-40. Representative transportation routes for the State of Texas.

Table G-60. Estimated transportation impacts for the State of Utah.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	8,740	190	950	0.12	0.57	0.23	0.80	4.8×10^{-4}	0.31	1.2
Truck	1,793	50	73	0.030	0.044	0.030	0.016	9.5×10^{-6}	0.063	0.17
Total	10,533	240	1,000	0.15	0.62	0.26	0.81	4.9×10^{-4}	0.38	1.4
Mina										
Rail	7,532	33	420	0.020	0.25	0.045	0.19	1.1×10^{-4}	0.14	0.45
Truck	1,793	50	73	0.030	0.044	0.030	0.016	9.5×10^{-6}	0.063	0.17
Total	9,325	83	490	0.050	0.30	0.075	0.21	1.2×10^{-4}	0.20	0.62

a. Totals might differ from sums of values due to rounding.

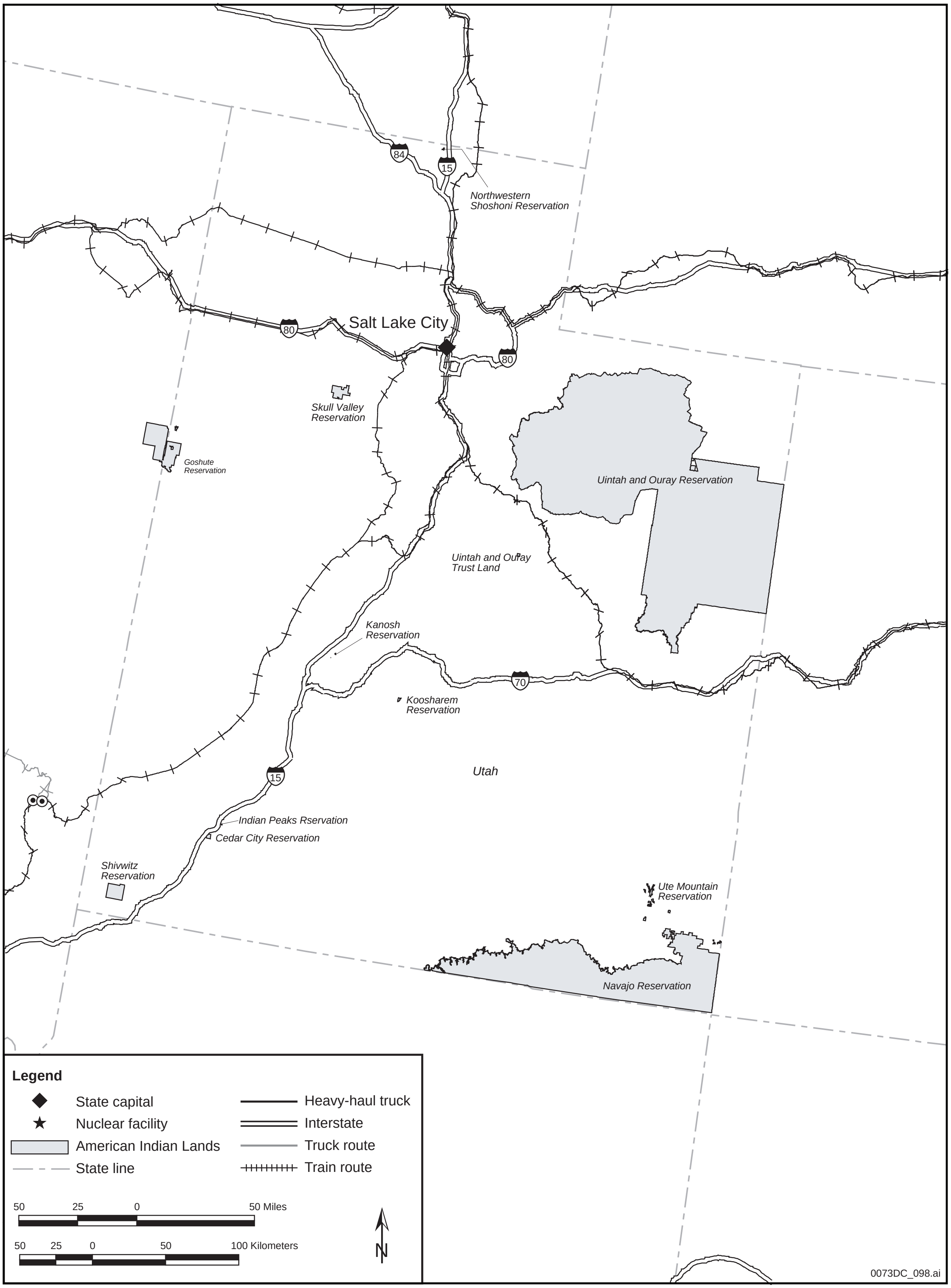

Figure G-41. Representative transportation routes for the State of Utah.

Table G-61. Estimated transportation impacts for the State of Vermont.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	199	0.087	4.2	5.2×10^{-5}	0.0025	3.9×10^{-5}	2.1×10^{-4}	1.3×10^{-7}	1.9×10^{-4}	0.0028
Truck	0	0	0	0	0	0	0	0	0	0
Total	199	0.087	4.2	5.2×10^{-5}	0.0025	3.9×10^{-5}	2.1×10^{-4}	1.3×10^{-7}	1.9×10^{-4}	0.0028
Mina										
Rail	199	0.087	4.2	5.2×10^{-5}	0.0025	3.9×10^{-5}	2.1×10^{-4}	1.3×10^{-7}	1.9×10^{-4}	0.0028
Truck	0	0	0	0	0	0	0	0	0	0
Total	199	0.087	4.2	5.2×10^{-5}	0.0025	3.9×10^{-5}	2.1×10^{-4}	1.3×10^{-7}	1.9×10^{-4}	0.0028

a. Totals might differ from sums of values due to rounding.

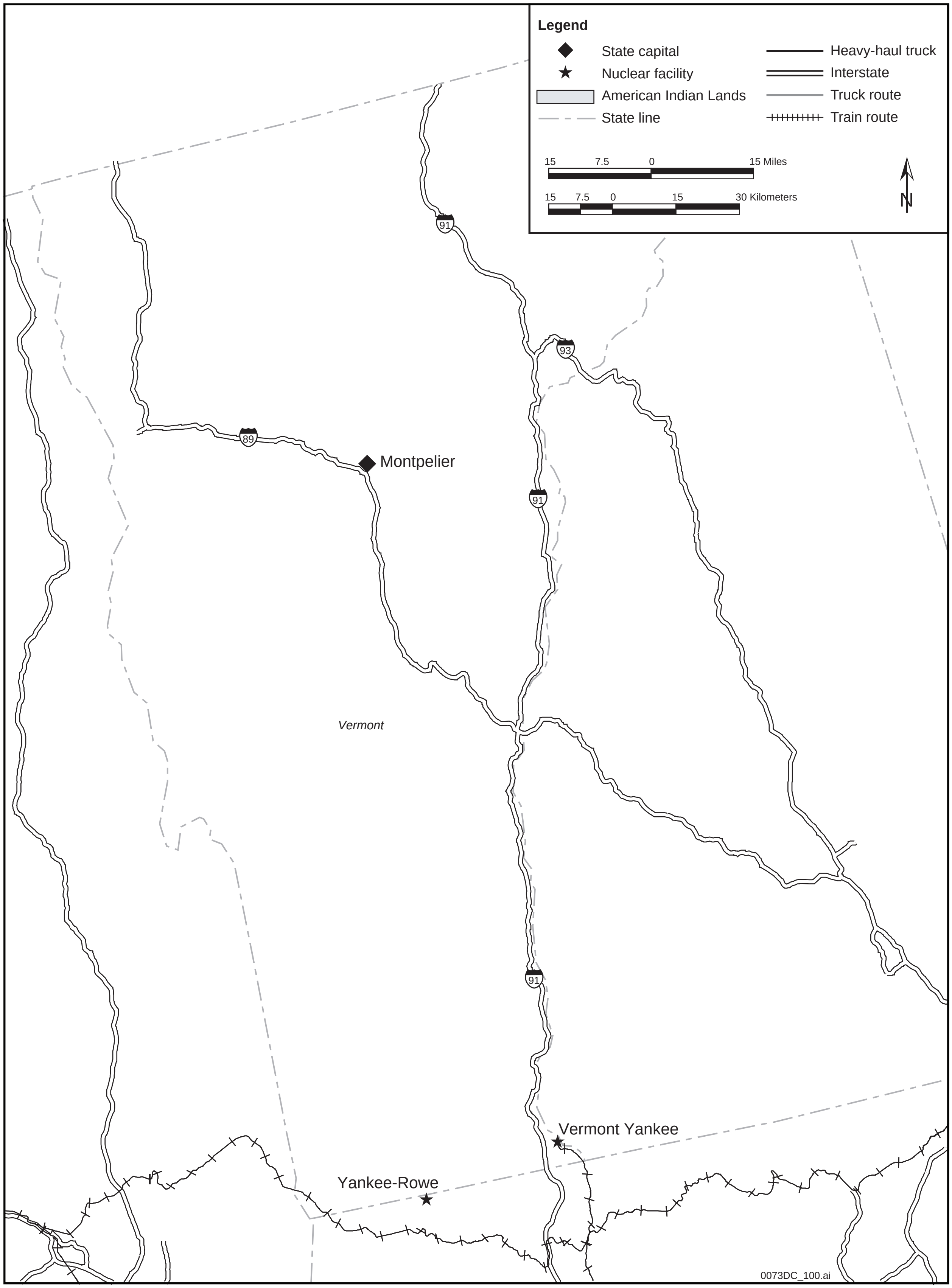

Figure G-42. Representative transportation routes for the State of Vermont.

Table G-62. Estimated transportation impacts for the Commonwealth of Virginia.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	390	5.9	40	0.0036	0.024	0.0060	0.023	1.4×10^{-5}	0.0078	0.041
Truck	0	0	0	0	0	0	0	0	0	0
Total	390	5.9	40	0.0036	0.024	0.0060	0.023	1.4×10^{-5}	0.0078	0.041
Mina										
Rail	390	5.9	40	0.0036	0.024	0.0060	0.023	1.4×10^{-5}	0.0078	0.041
Truck	0	0	0	0	0	0	0	0	0	0
Total	390	5.9	40	0.0036	0.024	0.0060	0.023	1.4×10^{-5}	0.0078	0.041

a. Totals might differ from sums of values due to rounding.

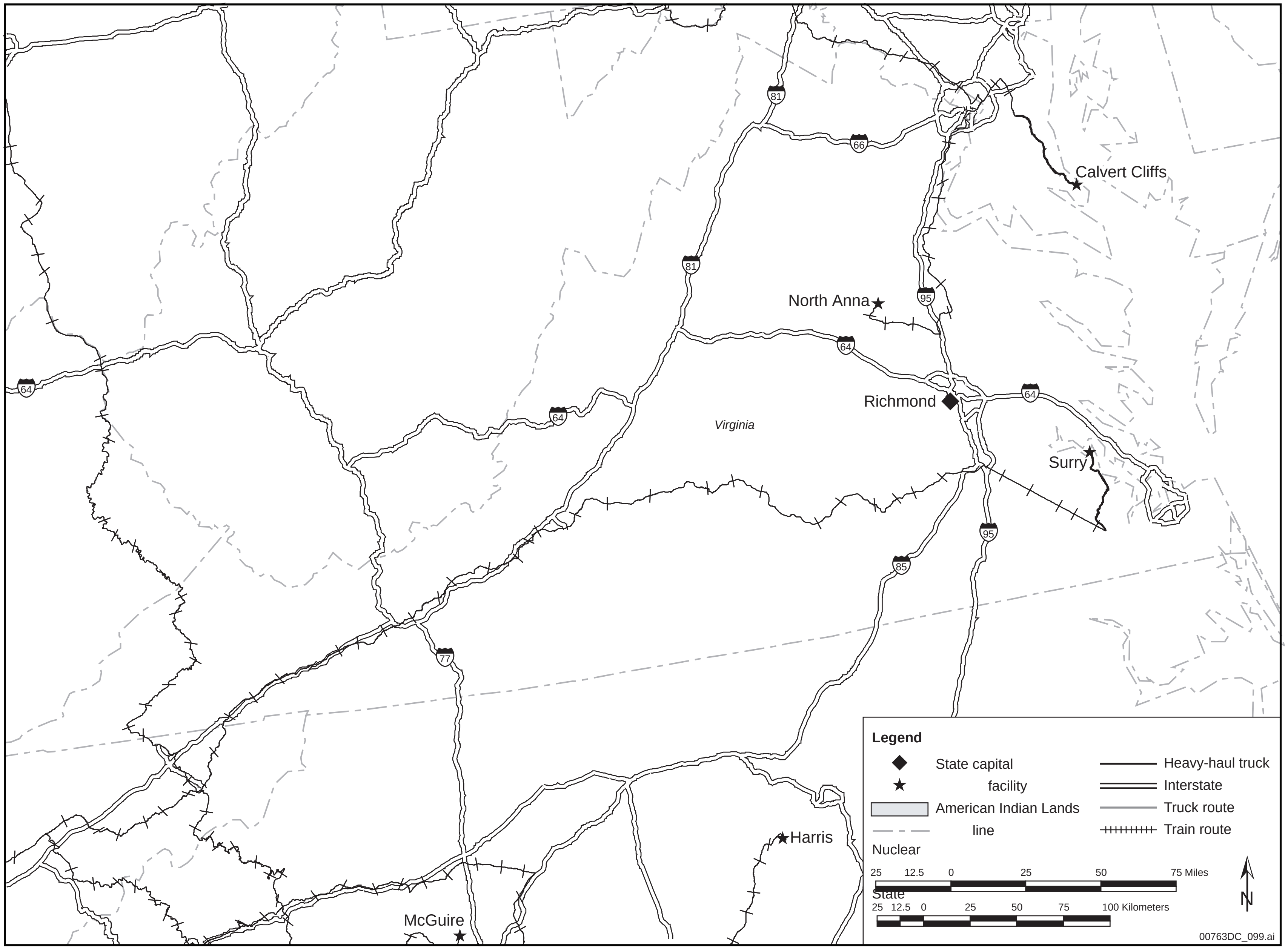

Figure G-43. Representative transportation routes for the Commonwealth of Virginia.

Table G-63. Estimated transportation impacts for the State of Washington.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	1,274	7.9	73	0.0047	0.044	0.0066	0.0045	2.7×10^{-6}	0.0066	0.062
Truck	3	0.0098	0.15	5.9×10^{-6}	9.3×10^{-5}	4.9×10^{-6}	2.4×10^{-6}	1.4×10^{-9}	6.8×10^{-6}	1.1×10^{-4}
Total	1,277	7.9	73	0.0047	0.044	0.0066	0.0045	2.7×10^{-6}	0.0066	0.062
Mina										
Rail	1,274	7.9	73	0.0047	0.044	0.0066	0.0045	2.7×10^{-6}	0.0066	0.062
Truck	3	0.0098	0.15	5.9×10^{-6}	9.3×10^{-5}	4.9×10^{-6}	2.4×10^{-6}	1.4×10^{-9}	6.8×10^{-6}	1.1×10^{-4}
Total	1,277	7.9	73	0.0047	0.044	0.0066	0.0045	2.7×10^{-6}	0.0066	0.062

a. Totals might differ from sums of values due to rounding.

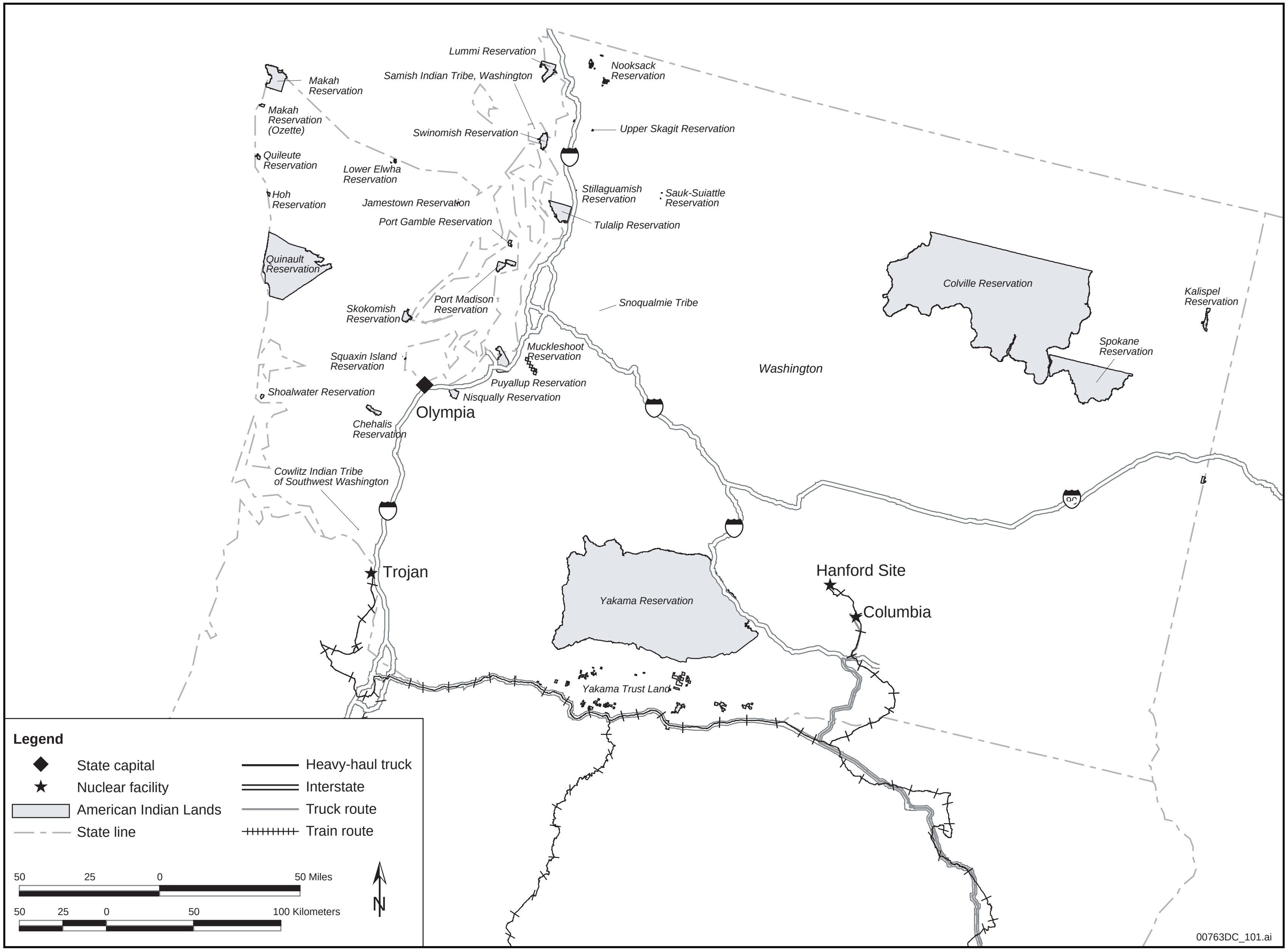

Figure G-44. Representative transportation routes for the State of Washington.

Table G-64. Estimated transportation impacts for the State of West Virginia.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	255	0.30	3.3	1.8×10^{-4}	0.0020	4.6×10^{-4}	0.0018	1.1×10^{-6}	0.0022	0.0048
Truck	0	0	0	0	0	0	0	0	0	0
Total	255	0.30	3.3	1.8×10^{-4}	0.0020	4.6×10^{-4}	0.0018	1.1×10^{-6}	0.0022	0.0048
Mina										
Rail	255	0.30	3.3	1.8×10^{-4}	0.0020	4.6×10^{-4}	0.0018	1.1×10^{-6}	0.0022	0.0048
Truck	0	0	0	0	0	0	0	0	0	0
Total	255	0.30	3.3	1.8×10^{-4}	0.0020	4.6×10^{-4}	0.0018	1.1×10^{-6}	0.0022	0.0048

a. Totals might differ from sums of values due to rounding.

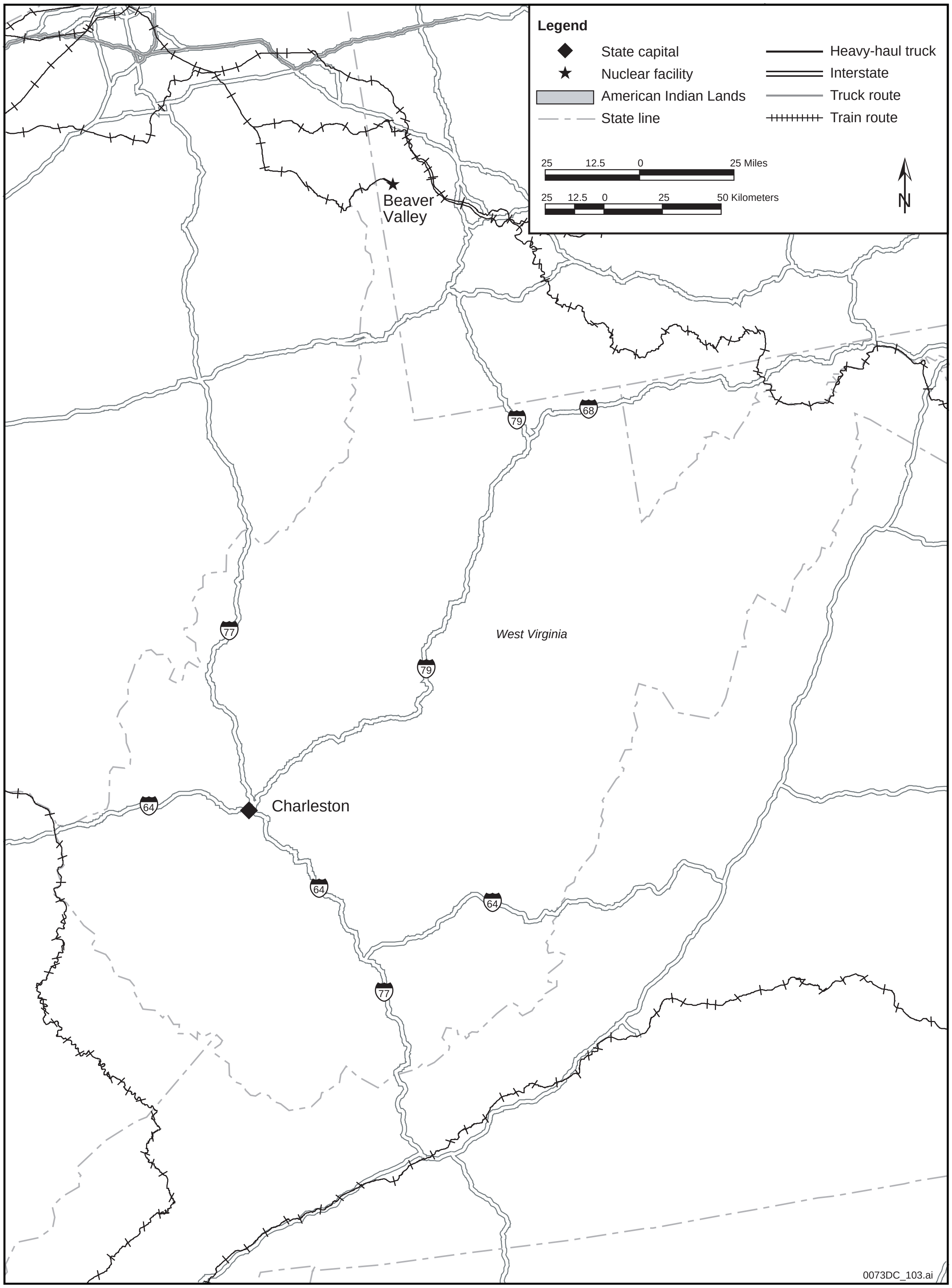

Figure G-45. Representative transportation routes for the State of West Virginia.

Table G-65. Estimated transportation impacts for the State of Wisconsin.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	152	3.5	33	0.0021	0.020	0.0031	0.013	7.6×10^{-6}	0.0038	0.029
Truck	37	0.089	1.8	5.3×10^{-5}	0.0011	4.4×10^{-5}	3.7×10^{-5}	2.2×10^{-8}	7.5×10^{-5}	0.0012
Total	189	3.5	34	0.0021	0.021	0.0031	0.013	7.6×10^{-6}	0.0038	0.030
Mina										
Rail	152	3.5	33	0.0021	0.020	0.0031	0.013	7.6×10^{-6}	0.0038	0.029
Truck	37	0.089	1.8	5.3×10^{-5}	0.0011	4.4×10^{-5}	3.7×10^{-5}	2.2×10^{-8}	7.5×10^{-5}	0.0012
Total	189	3.5	34	0.0021	0.021	0.0031	0.013	7.6×10^{-6}	0.0038	0.030

a. Totals might differ from sums of values due to rounding.

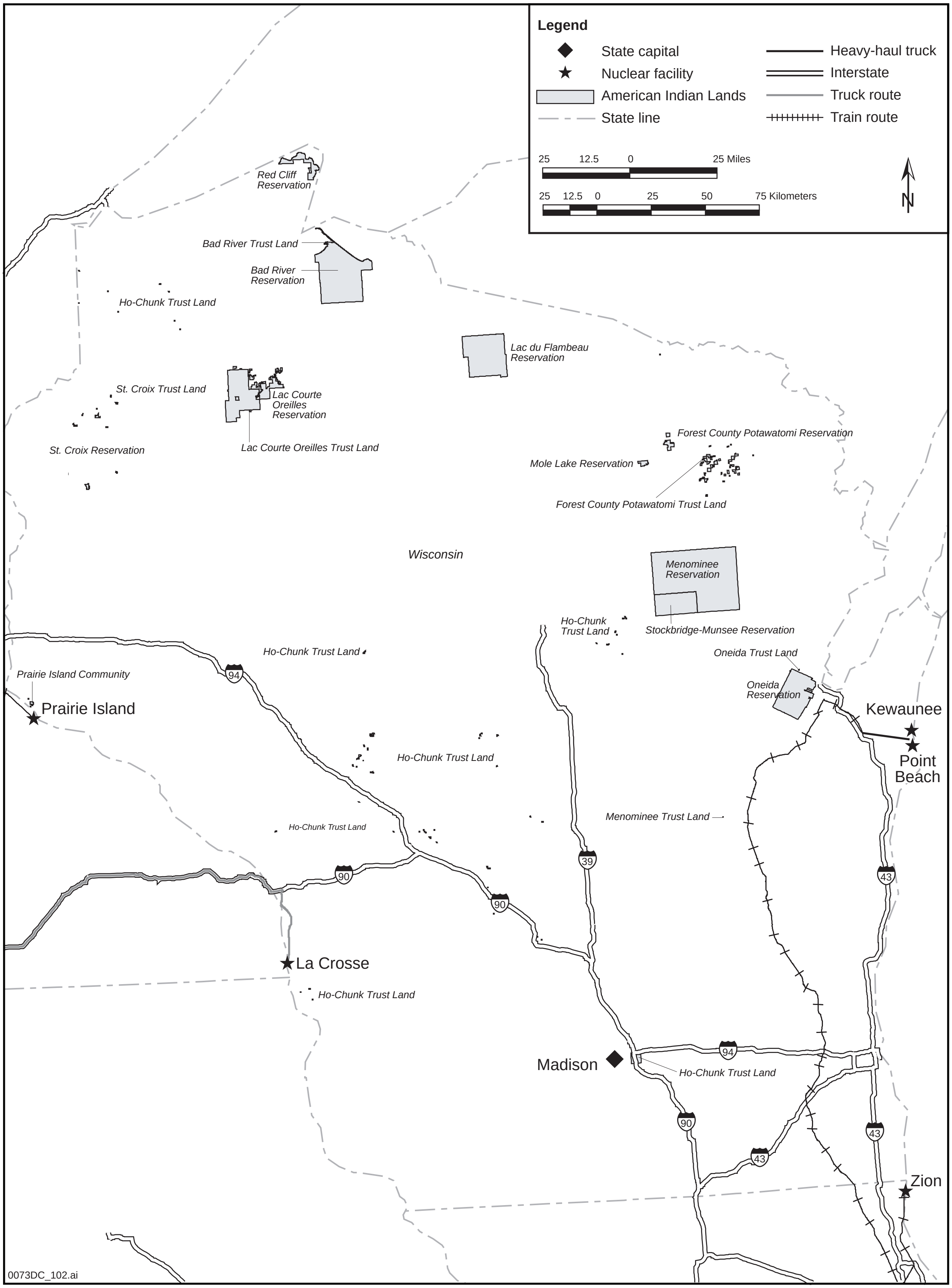

Figure G-46. Representative transportation routes for the State of Wisconsin.

Table G-66. Estimated transportation impacts for the State of Wyoming.

Rail alignment	No. of casks	Members of the public radiation dose (person-rem)	Involved workers radiation dose (person-rem)	Members of the public (latent cancer fatalities)	Involved workers (latent cancer fatalities)	Vehicle emission fatalities	Radiological accident dose risk (person-rem)	Radiological accident risk (latent cancer fatalities)	Traffic fatalities	Total fatalities
Caliente										
Rail	6,354	18	390	0.011	0.23	0.025	0.11	6.4×10^{-5}	0.28	0.55
Truck	1,789	23	77	0.014	0.046	0.0022	0.0027	1.6×10^{-6}	0.062	0.12
Total	8,143	41	470	0.025	0.28	0.027	0.11	6.5×10^{-5}	0.34	0.67
Mina										
Rail	6,354	18	390	0.011	0.23	0.025	0.11	6.4×10^{-5}	0.28	0.55
Truck	1,789	23	77	0.014	0.046	0.0022	0.0027	1.6×10^{-6}	0.062	0.12
Total	8,143	41	470	0.025	0.28	0.027	0.11	6.5×10^{-5}	0.34	0.67

a. Totals might differ from sums of values due to rounding.

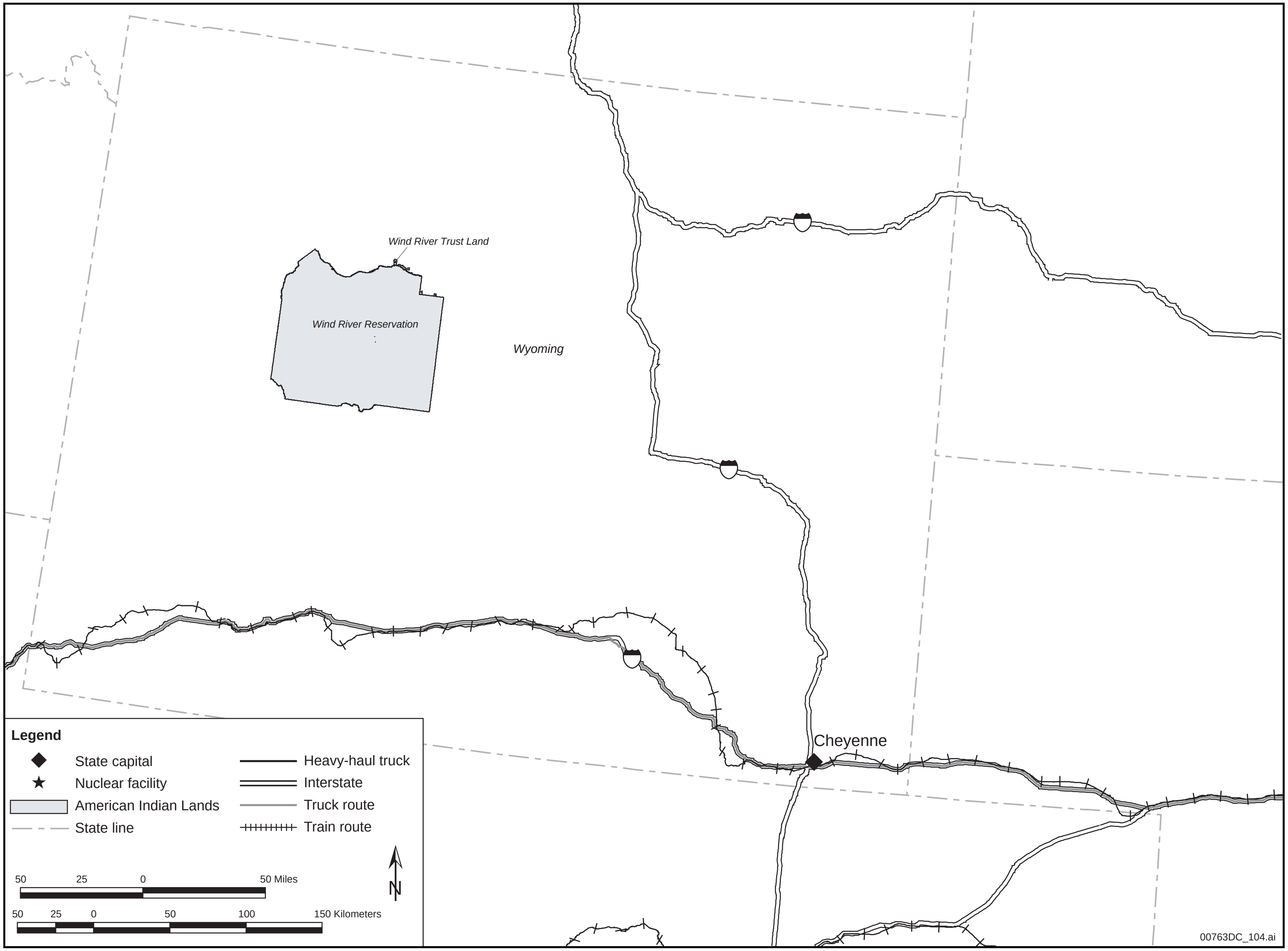

Figure G-47. Representative transportation routes for the State of Wyoming.

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Appendix H

Supplemental Transportation Information

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H. SUPPLEMENTAL TRANSPORTATION INFORMATION

H.1 Introduction

The U.S. Department of Energy (DOE or the Department) developed this appendix to provide general background information on transportation-related topics. Although this information is not essential for analysis of potential impacts from the transportation of spent nuclear fuel and high-level radioactive waste to a repository at Yucca Mountain, Nevada, it will help readers to understand how the transportation system would operate within the regulatory framework for the transportation of these materials. Section H.2 discusses transportation regulations, Section H.3 describes the components of a transportation system, and Section H.4 discusses operational practices. Section H.5 describes cask safety and testing. Section H.6 discusses emergency response, and Section H.7 describes available assistance for state, local, and American Indian tribal governments for emergency response planning. Section H.8 discusses DOE plans for transportation security, and Section H.9 describes potential liability under the *Price-Anderson Act* [Section 170 of the *Atomic Energy Act*, as amended (42 U.S.C. 2011 et seq.)].

Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation, the component elements of which have not been separated by reprocessing. In this document, the term refers to the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies and includes commercial spent nuclear fuel (including mixed-oxide fuel) from civilian nuclear power reactors, and DOE spent nuclear fuel from DOE and non-DOE production reactors, naval reactors, test and experimental reactors, and research reactors. Naval spent nuclear fuel shipments to the repository would be conducted under the authority of Presidential Executive Order 12344 and Public Law 106-65 and would be in compliance with applicable sections of the Code of Federal Regulations.

Most nuclear power reactors use solid uranium dioxide ceramic pellets of low-enriched uranium for fuel. The pellets are sealed in strong metal tubes, which are bundled together to form a nuclear fuel assembly. Depending on the type of reactor, typical fuel assemblies can be as long as 4.9 meters (16 feet) and weigh up to 540 kilograms (1,200 pounds). After a period in a reactor, the fuel is no longer efficient for the production of power and the assembly is removed from the reactor. After removal, the assembly (now called spent nuclear fuel) is highly radioactive and requires heavy shielding and remote handling to protect workers and the public.

High-level radioactive waste is the highly radioactive material that resulted from the reprocessing of spent nuclear fuel; it includes liquid waste that was produced directly in reprocessing and any solid material from such liquid waste that contains fission products in sufficient concentrations. High-level radioactive waste also includes other highly radioactive material that the U.S. Nuclear Regulatory Commission (NRC), consistent with existing law, has determined by rule to require permanent isolation. Immobilized surplus weapons-usable plutonium is part of the high-level radioactive waste inventory. All high-level radioactive waste would be in a solid form before DOE would ship it to Yucca Mountain.

H.2 Transportation Regulations

The shipment of spent nuclear fuel and high-level radioactive waste is highly regulated. For transportation of these materials to Yucca Mountain, DOE would meet or exceed U.S. Department of

Transportation and NRC rules. DOE would also work with states, local government officials, federally recognized American Indian tribes, utilities, the transportation industry, and other interested parties in a cooperative manner to develop the transportation system.

The *Hazardous Materials Transportation Act*, as amended (49 U.S.C. 1801 et seq.), directs the U.S. Department of Transportation to develop transportation safety standards for hazardous materials, including radioactive materials. Title 49 of the Code of Federal Regulations contains U.S. Department of Transportation standards and requirements for the packaging, transporting, and handling of radioactive materials for all modes of transportation. NRC sets additional design and performance standards for packages that carry materials with higher levels of radioactivity.

The *Nuclear Waste Policy Act*, as amended (42 U.S.C. 10101 et seq.; NWPA), requires that all shipments of spent nuclear fuel and high-level radioactive waste to Yucca Mountain be in NRC-certified casks and in accordance with NRC regulations related to advance notification of state and local governments. In addition, DOE has committed to notification of American Indian tribal governments for these shipments (DIRS 171934-DOE 2002, p. 23). NRC rules do not require notification of local authorities, which is the responsibility of the individual state governments. This section discusses the key regulations that govern the transportation of spent nuclear fuel and high-level radioactive waste.

H.2.1 PACKAGING

The primary means for the protection of people and the environment during radioactive materials shipment is the use of radioactive materials packages that meet U.S. Department of Transportation and NRC requirements. Packages are selected based on activity, type, and form of the material to be shipped. All spent nuclear fuel and high-level radioactive waste shipments to Yucca Mountain would be in Type B casks, which have the most stringent design standards to prevent release of radioactive materials under normal conditions of transport and during hypothetical accidents (Section H.4.10 discusses accident conditions). NRC regulates and certifies the design, manufacture, testing, and use of Type B packages under regulations in 10 CFR Part 71. All shippers must properly package radioactive materials so that external radiation levels do not exceed regulatory limits. The packaging protects handlers, transporters, and the public from exposure to dose rates in excess of recognized safe limits. Regulations in 10 CFR 71.47 and 49 CFR 173.441 prescribe the external radiation standards for all packages. For shipments to the repository, the limiting radiation dose limit would be 10 millirem per hour at any point 2 meters (6.6 feet) from the outer edge of the railcar or truck trailer.

H.2.2 MARKING, LABELING, AND PLACARDING

U.S. Department of Transportation regulations in 49 CFR require that shippers meet specific hazard communication requirements in marking and labeling packages that contain radioactive materials and other hazardous materials. Markings, labels, and placards identify the hazardous contents to emergency responders in the event of an incident.

Markings provide the proper shipping name, a four-digit hazardous materials number, the shipper's name and address, gross weight, and type of packaging; other important information labels on opposite sides of a package identify the contents and radioactivity level. Shippers of radioactive materials use one of three labels—Radioactive White I, Yellow II, or Yellow III—as shown in Figure H-1. The use of a particular label is based on the radiation level at the surface of the package and the transport index. The *transport* *index*, determined in accordance with 49 CFR 173.403, is a number on the label of a package that indicates the degree of control the carrier must exercise during shipment. Packaging that previously contained Class 7 (radioactive) materials and has been emptied of its contents as much as practicable is exempted from marking requirements. However, 49 CFR 173.428 requires the application of an Empty label (not shown) to the cask.

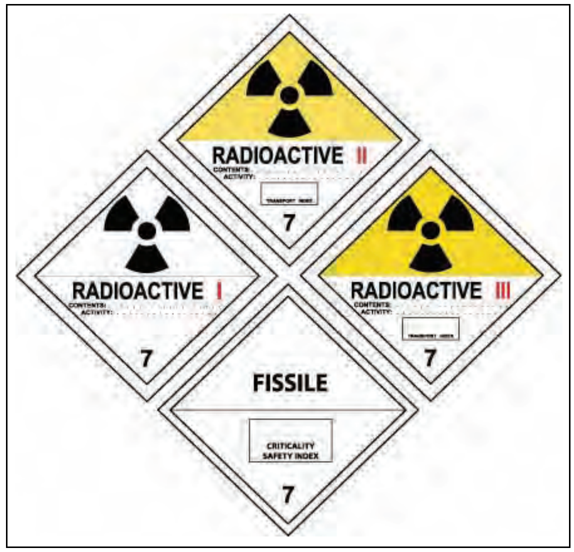

Figure H-1. Radioactive material shipment labels.

Figure H-1 also shows a Fissile label, which shippers must apply to each package with fissile material (a material that is capable of sustaining a chain reaction of nuclear fission). Such labels, where applicable, must be affixed adjacent to the labels for radioactive materials. The Fissile label includes the Criticality Safety Index, which indicates how many fissile packages can be grouped together on a conveyance.

Shipments of spent nuclear fuel and high-level radioactive waste are usually classified as Highway Route-Controlled Quantities of Radioactive Materials, and 49 CFR 172.403(c) requires Radioactive Yellow-III labels for them regardless of the radiation dose rate. For Radioactive Yellow III shipments, 49 CFR 172.504 requires radioactive hazard communication placards (Figure H-2) on each side and each end of a freight container, transport vehicle, or railcar. In addition, for Highway Route-Controlled Quantities of Radioactive Materials shipments the placard must be on a white square background with a black border (49 CFR 172.507 through 172.527). In addition to the placard, a vehicle might have a United Nations Identification Number near the placard. The United Nations assigns these four-digit numbers, which shippers commonly use throughout the world to aid in the quick identification of materials in bulk containers. The number appears on either an orange plane or on a plain white square-on-point configuration similar to a placard. The usual identification number for spent nuclear fuel is UN3328.

Figure H-2. Radioactive hazard communication placard.

H.2.3 SHIPPING PAPERS

The shipper prepares shipping papers and gives them to the carrier. These documents contain additional details about the cargo and include a signed certification that the material is properly classified and in proper condition for transport. Shipping papers also contain emergency information that includes contacts and telephone numbers. Highway carriers must keep shipping papers readily available during transport for inspection by appropriate officials such as state or federal inspectors.

H.2.4 ROUTING

U.S. Department of Transportation regulations classify spent nuclear fuel and high-level radioactive waste as Highway Route-Controlled Quantities of Radioactive Materials shipments. Carriers of these materials are required to use *preferred routes*, which include interstate highway systems or alternative routes selected by state or tribal routing authorities in accordance with U.S. Department of Transportation regulations. Preferred routes generally use beltways and bypasses around cities to avoid highly populated urban centers.

States and tribes can designate alternative preferred routes by following U.S. Department of Transportation regulations for designation and performing a comparative route analysis that adequately considers overall risk to the public. Factors for the analysis can include accident rates, traffic counts, distance, vehicle speeds, population density, land use, timeliness, and availability of emergency response capabilities. States must also document required consultation with affected neighboring jurisdictions. U.S. Department of Transportation highway routing regulations preempt any conflicting routing requirements that state, local, or tribal governments might issue, such as prohibitions on radioactive waste shipments through local nuclear-free zones.

No federal routing rules govern spent nuclear fuel and high-level radioactive waste shipments by rail. Because railroads are privately owned and operated, route selection would involve discussions between DOE and the chosen railroad companies and other stakeholders. Key factors for selection of rail routes include time and distance in transit, the track class and capacity, operational input from carriers, and infrastructure capabilities.

The U.S. Department of Homeland Security and U.S. Department of Transportation issued rulemaking proposals in relation to railroad routing for radioactive materials shipments for security purposes on December 21, 2006; Section H.2.9 discusses the proposals.

H.2.5 ADVANCE NOTIFICATION

DOE Manual 460.2-1, *Radioactive Material Transportation Practices* (DIRS 171934-DOE 2002, all), which implements DOE Order 460.1B, *Packaging and Transportation Safety* and NRC regulations (10 CFR 71.97 and 73.37), requires written notice to governors, or their designees, before shipment of spent nuclear fuel and high-level radioactive waste through their states. If sent by regular mail, the notice must be postmarked at least 7 days before the shipment; for messenger service, it must arrive 4 days before. The notification must contain the name, address, and telephone number of the shipper, the carrier, and the receiver; a description of the shipment; a list of the routes within the state; the estimated date and time of departure from the point of origin; the estimated date and time of entry into the state; and a statement on safeguarding schedule information. Federal regulations allow states to release certain advance information to local officials on a need-to-know basis. As required by Section 180 of the NWPA, all shipments to a repository would comply with NRC regulations on advance notification of state and local governments. In the event of a change in schedule that differs more than 6 hours from what was in the notification to the governor or their designee, DOE would provide the state with the new schedule by telephone. Although current regulations do not require notification of tribal authorities, DOE policy is to inform tribes of spent nuclear fuel and high-level radioactive waste shipments that would pass through their jurisdictions (DIRS 171934-DOE 2002, p. 23).

NRC issued an Advance Notice of Proposed Rulemaking (64 FR 71331) on December 21, 1999, to invite early input from affected parties and the public on advance notification to American Indian tribes of spent nuclear fuel and high-level waste shipments. Although the Commission approved a rulemaking plan, it put the rulemaking on hold pending review of Commission rules in response to the events of September 11, 2001. NRC is coordinating the schedule for this rulemaking with other security rulemaking activities. The current schedule would result in a proposed rule in about 2010.

H.2.6 RAILROAD SAFETY PROGRAM

The *Rail Safety Act of 1970* (Public Law 91-458) authorized states to work with the Federal Railroad Administration to enforce federal railroad safety regulations. States can enforce federal standards for track, signal and train control, motive power and equipment, and operating practices. In 1992, the State Safety Participation regulations (49 CFR Part 212) were revised to permit states to perform hazardous materials inspections of rail shipments. The Grade Crossing Signal System Safety regulations (49 CFR Part 234) were revised to authorize federal and state signal inspectors to ensure that railroad owners or operators were properly testing, inspecting, and maintaining automated warning devices at grade crossings. Before state participation can begin, each state agency must enter into a multiyear agreement with the Federal Railroad Administration for the exercise of specified authority. This agreement can delegate investigative and surveillance authority in relation to all or any part of federal railroad safety laws.

H.2.7 PERSONNEL TRAINING

U.S. Department of Transportation regulations require proper training for anyone involved in the preparation or transportation of hazardous materials, including radioactive materials. In accordance with 49 CFR Part 397, Subpart D, operators of vehicles that transport Highway Route-Controlled Quantities of Radioactive Materials receive special training that covers the properties and hazards of the materials, associated regulations, and applicable emergency procedures. In addition, DOE Orders require that driver or crew training covers operation of the specific package tie-down systems, cask recovery procedures, use of radiation detection instruments, use of satellite tracking systems and other communications equipment, adverse weather and safe parking procedures, public affairs awareness, first responder awareness (29 CFR 1910.120 [q]), and radiation worker “B” (or equivalent) training.

The U.S. Department of Transportation also requires training specific to the mode of transportation. Highway carriers are responsible for the development and maintenance of a qualification and training program that meets Department of Transportation requirements. Rail carriers must comply with Federal Railroad Administration regulations. Rail carriers are responsible for training and qualification of their crews, which includes application of 49 CFR Part 240 for locomotive engineer certification. If DOE decided to provide federal rail crews for waste shipments on the national rail system, the carriers would require a pilot, who would be an engineer familiar with the rail territory, unless the federal engineer was qualified on that route. The Federal Railroad Administration requires recurrent and function-specific training for personnel who perform specific work, such as train crews, dispatchers, and signal maintainers. In addition, the regulations require that each employee receives training that specifically addresses the job function.

H.2.8 OTHER REQUIREMENTS

Organizations that represent different transportation modes often establish mode-specific standards. For example, all North American shipments by rail that change carriers must meet Association of American

Railroads interchange rules. Equipment in interchanges must also meet the requirements of the *Association of American Railroads Field Manual of the A.A.R. Interchange Rules* (DIRS 175727-AAR 2005, all).

On May 1, 2003, the Association released Standard S-2043, *Performance Specification for Trains Used To Carry High-Level Radioactive Material* (DIRS 166338-AAR 2003, all) to establish performance guidelines and specifications for trains that carry spent nuclear fuel or high-level radioactive waste. These guidelines apply to the individual railcars within the train, and they promote communication between railroads, spent nuclear fuel and high-level radioactive waste shippers, and railcar suppliers. The objectives of this standard are (1) to provide a cask, railcar, and train system that ensures safe transportation of casks in the railroad operating environment and allows timetable speeds with limited restrictions and (2) to use the best available technology to minimize the chances of derailment in transportation. This standard reflects the current technical understanding of the railroad industry in relation to optimum vehicle performance through application of current and prospective new railcar technologies. On December 20, 2005, the Association adopted two appendixes to AAR S-2043: Appendix A, “Maintenance Standards and Recommended Practices for Trains Used To Carry High-Level Radioactive Material,” and Appendix B, “Operating Standard for Trains Used To Carry High-Level Radioactive Material” (DIRS 166338-AAR 2003, all). Changes and additions to this standard can be expected as specific vehicles are developed. All future changes will be based on the achievement of optimum performance within acceptable expectations for safe operations.

Association of American Railroads Circular No. OT-55-I, *Recommended Railroad Operating Practices for Transportation of Hazardous Materials* (DIRS 183011-AAR 2006, all), provides recommendations on operating practices that are adopted by Association of American Railroads and American Short Line and Regional Railroad Association members in the United States for these shipments. The current revision of the circular became effective July 17, 2006; its recommendations cover road operating practices, yard operating practices, storage and separation distances, transportation community awareness and emergency response program implementation, criteria for shipper notification, time-sensitive materials, and special provisions for spent nuclear fuel and high-level radioactive waste.

The Commercial Vehicle Safety Alliance has developed inspection procedures and out-of-service criteria for commercial highway vehicles that transport shipments of transuranic elements and Highway Route-Controlled Quantities of Radioactive Materials shipments (Section H.4.9). Under these procedures, each state through which a shipment passed would inspect each shipment to the repository, and a shipment would not begin or continue until inspectors determined that the vehicle and its cargo were free of defects.

Trucks that carry spent nuclear fuel or high-level radioactive waste and weigh over 36,300 kilograms (80,000 pounds) would exceed federal commercial vehicle weight limits for nondivisible loads (which cannot be separated into smaller loads). Most states require transportation companies to obtain permits when their vehicles exceed weight limits to control time and place of movement. Local jurisdictions also often require overweight permits. The criteria for the permitting process are not uniform among different jurisdictions. A number of factors affect issuance of these permits including traffic volumes and patterns, protection of state highways and structures such as bridges, zoning and general characteristics of the route, and safety of the motoring public.

H.2.9 PROPOSED RAIL REGULATIONS

The U.S. Department of Transportation's Pipeline and Hazardous Materials Safety Administration, in consultation with the Federal Railroad Administration, has proposed revision of the current requirements in the Hazardous Materials Regulations applicable to the safe and secure transportation of hazardous materials by rail at 49 CFR Parts 172 and 174 (71 FR 76834; December 21, 2006). The proposed rulemaking includes "Radioactive Materials" and "Class 7- Highway Route-Controlled Quantities of Radioactive Materials." The proposal would require rail carriers to compile annual data on specified shipments of hazardous materials, to use the data to analyze safety and security risks along rail transportation routes where those materials are transported, to assess alternative routes, and to make routing decisions based on those assessments. The Pipeline and Hazardous Materials Safety Administration has also proposed clarifications of the current security plan requirements to address en route storage, delays in transit, delivery notification, and additional security inspection requirements for hazardous materials shipments.

The Transportation Security Administration has proposed new security requirements for 49 CFR Parts 1520 and 1580 for freight railroad carriers; intercity, commuter, and short-haul passenger train service providers; rail transit systems; and rail operations at certain, fixed-site facilities that ship or receive specified hazardous materials by rail (71 FR 76852; December 21, 2006). The proposal would codify the scope of the existing inspection program and require regulated parties to allow Transportation Safety Administration and Department of Homeland Security officials to enter, inspect, and test property, facilities, and records relevant to rail security. This proposed rule would also require regulated parties to designate rail security coordinators and to report significant security concerns to the Department of Homeland Security.

In addition, the Transportation Security Administration has proposed that freight rail carriers and certain facilities that handle hazardous materials be able, on request, to report location and shipping information to the Administration and that they should implement chain-of-custody requirements to ensure a positive and secure exchange of specified hazardous materials (71 FR 76852, December 21, 2006). The proposal would clarify and extend the sensitive security information protections to cover certain information associated with rail transportation.

H.3 Transportation System Components

The DOE transportation system would consist of hardware (shipping containers, handling equipment, railcars, and truck trailers), a transportation operations center, a Cask Maintenance Facility, and the Nevada rail line.

H.3.1 SHIPPING CONTAINERS

As required by the NWPA, the designs of the shipping casks for transportation of the spent nuclear fuel and high-level radioactive waste would be NRC-certified. The casks would be sealed containers that could weigh up to 180 metric tons (200 tons). The casks would consist of layers of steel and lead or other materials, which would provide shielding against the radiation from the waste and prevent the materials from escaping to the environment in the event of an incident.

The open end of the cylindrical cask would be sealed with a heavy lid. Impact limiters on each end of the cask would absorb most of the impact force and provide protection of the container and its contents in the event of an incident. Figure H-3 illustrates generic rail and truck casks.

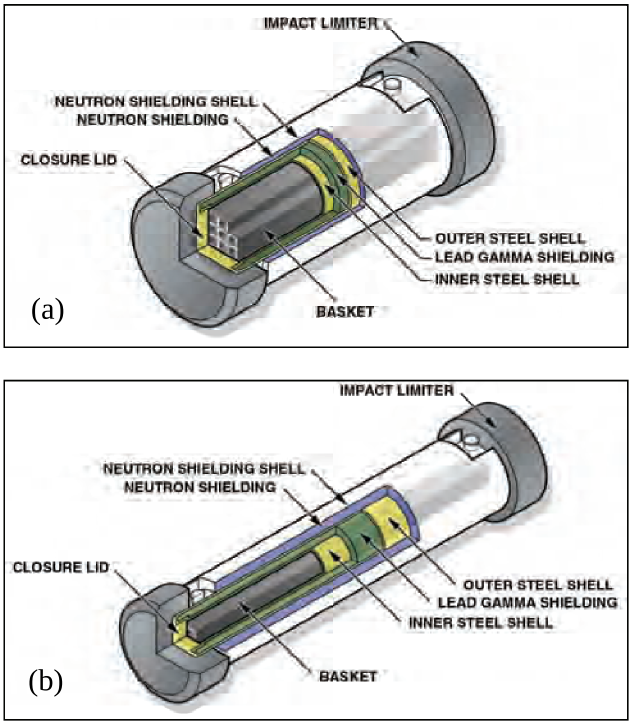

Figure H-3. Generic rail cask (a) and truck cask (b) for spent fuel.

DOE would procure NRC-certified casks from private industry. As required by Section 137 of the NWPA, DOE would use private industry to the fullest extent possible for each aspect of transportation. The Department has a preference for maximizing the use of existing cask designs rather than developing new ones. Existing cask designs would have to be modified to accommodate TAD canisters before NRC certification.

H.3.2 RAILCARS

The trains DOE would use to transport spent nuclear fuel and high-level radioactive waste to the repository would typically use locomotives, escort cars, one or more loaded cask railcars, and buffer railcars that would separate the cask railcars from occupied locomotives and escort railcars.

H.3.3 TRANSPORTATION OPERATIONS CENTER

The functions of a transportation operations center would include coordination between shipping sites and the repository, planning and scheduling of shipments, coordination with carriers, notifications to states and American Indian tribes, monitoring and tracking of shipments, en route communications, emergency management, and security coordination.

H.3.4 CASK MAINTENANCE FACILITY

Owners of rail and highway transportation casks and the associated equipment (for example, personnel barriers and impact limiters) must maintain them in proper condition to satisfy the requirements in their NRC certificates of compliance. The Cask Maintenance Facility would periodically remove casks from service and perform maintenance and inspection. The activities at the Cask Maintenance Facility would include but not be limited to testing, repair, minor decontamination, and approved modifications. The Cask Maintenance Facility would also serve as the primary recordkeeping facility for the cask fleet equipment.

H.3.5 TRANSPORT SERVICES

The U.S. freight railroad system consists of seven Class 1 railroads (mainline), 31 regional railroads, and over 500 local railroads (line-haul railroads smaller than regional railroads). Some origin sites of spent nuclear fuel and high-level radioactive waste have rail services, while others do not. DOE would use short-line or Class 1 railroads to transport casks from the origin sites. There are numerous short-line railroads that operate one or more relatively small sections of track that connect to the Class 1 rail network. Origin sites without rail service would require alternative intermodal delivery from the origin site to a nearby rail transfer facility, either by barge using a nearby dock or by heavy-haul truck using local highways.

At some sites with limited cask handling capability, DOE could use overweight trucks for smaller casks. After loading and preparation, DOE would pick up the cask and deliver it directly to the repository using the public highway network.

DOE would construct a branch rail line to transport casks from a Union Pacific mainline railroad in Nevada to the repository site, and the Department would contract the operation and maintenance of the branch rail line.

H.4 Operational Practices

DOE has adopted as policy the practices that were developed in consultation with stakeholders and are outlined in DOE Manual 460.2-1 (DIRS 171934-DOE 2002, all). The Manual establishes 14 standard transportation practices for Departmental programs to use in the planning and execution of shipments of radioactive materials including radioactive waste. It provides a standardized process and framework for planning and for interacting with state and tribal authorities and transportation contractors and carriers.

H.4.1 STAKEHOLDER INTERACTIONS

The *Strategic Plan for the Safe Transportation of Spent Nuclear Fuel and High-Level Radioactive Waste to Yucca Mountain: A Guide to Stakeholder Interactions* (DIRS 172433-DOE 2003, all) guides state and tribal government interactions, some of which are already under way. During planning and actual transportation operations, stakeholders are and would be involved in planning for route identification, funding approaches for emergency response planning and training, understanding safeguards and security requirements, operational practices, communications, and information access.

DOE is working collaboratively with states through State Regional Group committees, whose members are state officials responsible for transportation policy, law enforcement, emergency response, and oversight of hazardous materials shipments, and with American Indian tribal governments to assist them to prepare for the shipments.

In addition to coordination with State Regional Groups and tribal governments, a national cooperative effort is underway as part of the Transportation External Coordination Working Group, which involves a broad range of stakeholder organizations that routinely interact with DOE to provide input and recommendations on transportation planning and program information. DOE works with states, tribes, and industry to guide and focus emergency training, coordination with local officials, and other activities to prepare for shipments to the repository.

DOE is preparing a comprehensive national spent fuel transportation plan that accommodates stakeholder concerns to the extent practicable. The plan will outline the challenges and strategies for the development and implementation of the system required to transport the waste to Yucca Mountain.

H.4.2 ROUTE PLANNING PROCESS

An initial step in the planning process to ship spent nuclear fuel and high-level radioactive waste to Yucca Mountain is to identify a national suite of routes, both rail and highway. Stakeholder groups in the DOE program are participating in this process by examining potential routing criteria in the route identification process. State Regional Groups, tribal governments, transportation associations, industry, federal agencies, and local government organizations are some of the groups that work collaboratively with DOE in this process. DOE is performing and would perform the work through a Topic Group of the Transportation External Coordination Working Group, which would seek broader public input and collect comments on routing criteria and the process for development of a set of routes. The process includes consideration of industry practices, DOE requirements, and analysis of regional routes that states have previously evaluated in the process to identify a preliminary set of routes. Public involvement is an essential element of a safe, efficient, and flexible transportation system.

H.4.3 PLANNING AND MOBILIZATION

DOE would use the methods and requirements this section describes to establish the baseline operational organization and practices for route identification, fleet planning and acquisition, carrier interactions, and operations.

DOE would develop a Transportation Operations Plan to provide the basis for planning shipments. This plan would describe the operational strategy and delineate the steps to ensure compliance with applicable regulatory and DOE requirements. It would include information on organizational roles and responsibilities, shipment materials, projected shipping windows, estimated numbers of shipments, carriers, packages, sets of routes, prenotification procedures, safe parking arrangements, tracking systems, security arrangements, public information, and emergency preparedness, response, and recovery.

The Department would develop individual site plans to include the information necessary to ship from specific sites that included roles and responsibilities of the participants in the shipping campaign, shipment materials, schedules, number of shipments, types and number of casks and other equipment, carriers, routes, in-transit security arrangements, safe parking arrangements for rail and truck shipments, communications including prenotification, public information, tracking, contingency planning, and emergency preparedness, response, and recovery.

In addition, DOE would issue an Annual Shipment Projection at least 6 months to a year in advance of the beginning of a shipment year and would identify the sites from which it would ship spent nuclear fuel and high-level radioactive waste in a given calendar year, the expected characteristics and quantities of waste to be delivered by each site, types of casks, and anticipated numbers of casks and shipments. The Annual Shipment Projection would not define specific shipment schedules or routes, but DOE would use it for schedule and route planning.

H.4.4 DEDICATED TRAIN SERVICE POLICY

On July 18, 2005, in a policy statement (DIRS 182833-Golan 2005, all), DOE decided that dedicated train service would be the usual manner of rail shipment of commercial and most DOE spent nuclear fuel and high-level radioactive waste to Yucca Mountain. *Dedicated* indicates train service for one commodity (in this case, spent nuclear fuel and high-level radioactive waste). Past and current shipping campaigns have used dedicated train service to address issues of safety, security, cost, and operations. Analyses indicate that the primary benefit of dedicated train service would be significant cost savings over the lifetime of transportation operations. The added cost of dedicated train service would be offset by reductions in fleet size and its attendant operations and maintenance costs. In addition, the shorter times in transit and shorter layovers at switching yards would enhance safety and security. Use of dedicated train service would provide greater operational flexibility and efficiency because of the reduced transit time and greater predictability in routing and scheduling.

H.4.5 TRACKING AND COMMUNICATION

DOE would provide authorized state and tribal governments with the capability and training to monitor shipments to the repository through their jurisdictions using a satellite tracking system, such as the Transportation Tracking and Communication System, that would provide continuous, centralized monitoring and communications capability (DIRS 172433-DOE 2003, p. 5). Trained personnel could use such a system to monitor shipment progress and communicate with the dispatch center. A transportation operations center would be in contact with the carriers and the escorts throughout each shipment. In addition, all truck and rail escort cars would have communications equipment. The train control center would manage rail communications and signaling on the branch Nevada rail line.

DOE would develop detailed backup procedures to ensure safe operations in the event that the tracking system was temporarily unavailable. The procedures would be based on a telephone call-in system for operators to report shipment locations to DOE on a regular basis and before crossing state and tribal borders.

H.4.6 TRANSPORTATION OPERATIONAL CONTINGENCIES

DOE would obtain weather forecasts along routes as part of preshipment planning, notification, and dispatching. At the time of departure, current weather conditions, the weather forecast, and expected travel conditions would have to be acceptable for safe operations. If these conditions were not acceptable, DOE could delay the shipment until travel conditions became acceptable or reroute the shipment.

Shipments would not travel during severe weather or other adverse conditions that could make travel hazardous. DOE would obtain route conditions and construction information that could temporarily affect the planned route through consultation with the railroads and states along the planned route.

DOE would receive input from states and tribes on weather conditions through the satellite tracking system known as TRANSCOM, which they would also use to monitor shipments. Rail carriers use train control and monitoring systems to identify the locations of trains and to make informed decisions to avoid or minimize potentially adverse weather or track conditions. Truck dispatch centers and the transportation operations center would coordinate on weather conditions while shipments were en route.

Continuous communications with a transportation operations center would provide advance warning of potential adverse conditions along the route. If the shipment encountered unanticipated severe weather, the operators would contact this center to coordinate routing to a safe stopping area if it became necessary to delay the shipment until conditions improved.

H.4.7 CARRIER PERSONNEL QUALIFICATIONS

Carriers would develop and maintain qualification and training programs that met U.S. Department of Transportation requirements for drivers, operators, and security personnel. For truck drivers, qualifications include being at least 21 years of age, meeting physical standards, having a commercial driver's license, and successfully completing a road driving test in the shipment vehicle. In addition, drivers must have training on the properties and hazards of the shipment materials as well as the procedures to follow in the event of an emergency. Locomotive engineers must meet the Locomotive Engineer Certification requirements of 49 CFR Part 240, which include completion of an approved training program (Section H.2.7 addresses other training requirements).

H.4.8 NOTICE OF SHIPMENTS

The NRC requires advance notice, en route status, and other pertinent shipment information on DOE shipments (10 CFR Parts 71 and 73). Section H.2.5 addresses advance notification requirements. DOE and other stakeholders would use this information to support coordination of repository receipt operations, to support emergency response capabilities, to identify weather or road conditions that could affect shipments, to identify safe stopping locations, to schedule inspections, and to coordinate appropriate public information programs.

H.4.9 INSPECTIONS

To ensure safety, DOE would inspect shipments when they left their point of origin and when they arrived at the repository to verify vehicle safety and radiological safety of the shipping casks. These inspections would include radiological surveys of radioactive material packages to ensure that they met the radiation level limits of 49 CFR 173.441 and surface contamination limits of 49 CFR 173.443. DOE would inspect rail shipments in accordance with 49 CFR 174.9 and the Federal Railroad Administration High-Level Nuclear Waste Rail Transportation Inspection Policy in Appendix A of *Safety Compliance Oversight Plan for Rail Transportation of High-Level Radioactive Waste and Spent Nuclear Fuel* (DIRS 156703-FRA 1998, all), which includes motive power, signals, track conditions, manifests, and crew credentials. DOE would inspect highway shipments using the enhanced standards of the Commercial Vehicle Safety Alliance, which provide uniform inspection procedures for radiological requirements, drivers, shipping papers, vehicles, and casks (DIRS 175725-CVSA 2005, all).

Although DOE would minimize the number of stops to the extent practicable, under federal regulations states and tribes could order additional inspections when shipments entered their respective jurisdictions. DOE would attempt to coordinate those inspections with normal crew change locations whenever possible.

H.4.10 PROCEDURES FOR OFF-NORMAL CONDITIONS

Off-normal conditions are potentially adverse conditions that do not relate to accidents, incidents, or emergencies. They include but are not limited to mechanical breakdowns, fuel problems, tracking system failure, and illness, injury, or other incapacity of a member of the truck, train, or escort crew. DOE would require carriers to provide operators with specific written procedures that define detailed actions for off-normal events. Procedures would address notifications, deployment of appropriate hazard warnings, security, medical assistance, operator or escort replacement, and maintenance, repair, replacement, or recovery of equipment, as appropriate. Procedures would also cover selection of alternative routes and safe parking areas.

H.4.11 POSTSHIPMENT RADIOLOGICAL SURVEYS

DOE would visually inspect and radiologically survey the external surfaces of a cask after shipment in accordance with U.S. Department of Transportation, DOE, and NRC regulations. Receiving facility operators would survey each cask and transporter on arrival (before unloading) and determine if there was radiological contamination in excess of the applicable limits. The inspections would include the cask, tie-downs, and associated hardware to determine if physical damage occurred during transit.

H.4.12 SHIPMENT OF EMPTY TRANSPORT CASKS

Except before their first use, shipments of all empty transportation casks would comply with the requirements of the NRC certificate of compliance or 49 CFR 173.428, which addresses empty radioactive materials packages, whichever was applicable. DOE would ship casks that did not meet the criteria for “empty” in accordance with the applicable U.S. Department of Transportation hazardous materials regulations. Advance shipment notifications and en route inspections would not apply to the shipment of empty transportation casks; however, DOE would use dedicated train service to realize the cost benefits of a decreased fleet requirement.

H.5 Cask Safety

The purpose of the NRC regulations for transportation of spent nuclear fuel and high-level radioactive waste (10 CFR Part 71) is to protect the public health and safety from normal and off-normal conditions of transport and to safeguard and secure shipments of these materials. Over the years, NRC has amended its regulations to be compatible with the latest editions of the International Atomic Energy Agency and other standards (69 FR 3698, January 26, 2004).

In addition to the standard testing discussed below, NRC has committed to a package performance study for the full-scale testing of a spent nuclear fuel package of the kind DOE would likely use. The Commission approved the proposed test in June 2005 (DIRS 182896-Vietti-Cook 2005, all; DIRS 182897-Reyes 2005, all). According to the proposal, the package would contain surrogate fuel elements and be mounted on a railcar placed at 90 degrees to a simulated rail crossing. The rail package would be subjected to a collision with a locomotive and several freight cars at 96 kilometers (60 miles) per hour. NRC is formulating the study to give the public greater confidence in the movement of spent nuclear fuel, to provide information on the methods and processes of transportation system qualification, and to validate the applicability of NRC regulations.

Regulations in 10 CFR Part 71 require that casks for shipping spent nuclear fuel and high-level radioactive waste must be able to meet specified radiological performance criteria for normal transport and for transport under severe accident conditions. Meeting these requirements is an integral part of the safety assurance process for transportation casks. The ability of a design to withstand these conditions can be demonstrated by comparing designs to similar casks, engineering analyses (such as computer-simulated tests), or by scale-model or full-scale testing. As shown in Figure H-4, these hypothetical accident conditions include, in sequence, a 9-meter (30-foot) drop onto an unyielding flat surface, a 1-meter (40-inch) drop onto a vertical steel bar, exposure of the entire package to fire for 30 minutes, and

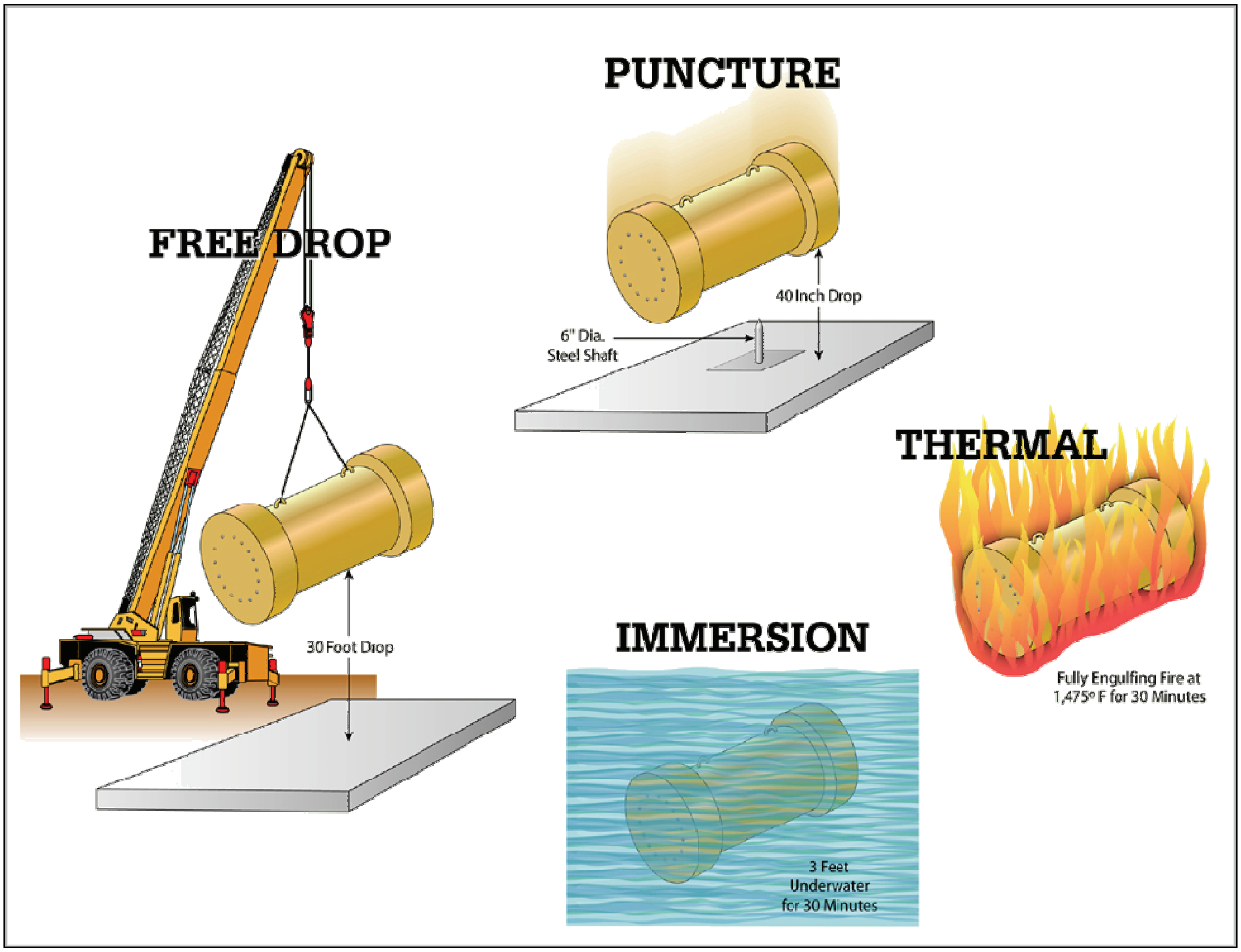

Figure H-4. Hypothetical accident conditions.

immersion in 0.9 meter (3 feet) of water. In addition, an undamaged cask must be able to survive submersion in the equivalent pressure of 15 and 200 meters (50 and 650 feet) of water.

For most accidents more severe than those the hypothetical accident conditions simulate, NRC studies (DIRS 152476-Sprung et al. 2000, all; DIRS 181841-Adkins et al. 2007, all; DIRS 182014-Adkins et al. 2006, all) show that the radiological criteria for containment, shielding, and subcriticality would still be satisfied. The studies also show that for the few severe incidents in which these criteria could be exceeded, only containment and shielding would be affected, and the regulatory criteria could be exceeded only slightly. Based on the analyses of the *Final Environmental Impact Statement for a Geological Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada* (DIRS 155970-DOE 2002, all), casks would continue to contain spent nuclear fuel and high-level radioactive waste fully in more than 99.99 percent of all incidents (of the thousands of shipments over the last 30 years, none has resulted in an injury due to the release of radioactive materials). The following sections discuss each of these packaging performance criteria.

H.5.1 NINE-METER DROP ONTO AN UNYIELDING SURFACE

The first set of accident conditions in the sequence simulates impact and evaluation of a 9-meter (30-foot) free fall onto an unyielding surface with the cask striking the target in the most damaging orientation.

The free fall results in a final velocity of 48 kilometers (30 miles) per hour. Although this velocity is less than the expected speed of interstate highway traffic, it is severe because the target surface is unyielding. This results in the cask absorbing all the energy of the drop, which is approximately equivalent to a 96-kilometer (60-mile)-per-hour impact with a medium hardness surface (such as shale or other relatively soft rock) and a 150-kilometer (90-mile)-per-hour impact with a soft surface (such as tillable soil).

H.5.2 ONE-METER DROP ONTO A STEEL BAR

The second set of accident conditions simulates a cask hitting a rod or bar-like object that could be present in an accident. This requires evaluation for a 1-meter (40-inch) drop onto a 15-centimeter (6-inch)-diameter rod on an unyielding surface. The cask must be in the orientation in which maximum damage would be likely. In addition, the bar must be long enough to cause maximum damage to the cask. This evaluates several impacts in which different parts of a cask strike the bar either by simulation or physical testing.

H.5.3 FIRE

The third set of accident conditions simulates a fire that occurs after the two impacts. This involves a hydrocarbon fire with an average flame temperature of 800°C (1,475°F) and requires the cask to be fully engulfed in the flame for 30 minutes.

H.5.4 WATER IMMERSION

The final set of accident conditions in the sequence is shallow immersion. The cask must be immersed in 0.9 meter (3 feet) of water. The purpose of this test is to ensure that water cannot leak into the cask after having passed through the challenges.

An undamaged version of the cask must also be able to survive immersion in the equivalent of 15 meters (50 feet) of water at a pressure of about 1,500 grams per square centimeter (22 pounds per square inch) to test for leakage. Furthermore, shipping casks for more than 1 million curies of radioactivity must be able to survive water pressure of about 20,000 grams per square centimeter (290 pounds per square inch) for 1 hour without collapsing, buckling, or leaking. That pressure is equivalent to a depth of about 200 meters (650 feet).

H.5.5 ACCEPTANCE CRITERIA

To be judged successful in meeting all but the 200-meter (650-foot) submersion requirement, a cask must not release more than limited amounts of radioactive material in 1 week. These release limits are set for each radionuclide based on dispersivity and toxicity. In addition, the cask must not emit radiation at a dose rate of greater than 1 rem per hour at a distance of 1 meter (3.3 feet) from the cask surface. Last, the contents of the cask must not be capable of undergoing a nuclear chain reaction, or criticality, as a result of the hypothetical accident conditions.

H.5.6 USE OF MODELS

Manufacturers can demonstrate the ability of a cask to survive these hypothetical accident conditions in several ways. They can subject a full-size model of the cask to the sequences, use smaller models of the casks (typically half- or quarter-scale), compare the cask design to previously licensed designs, or analyze the hypothetical accident scenarios with computer models. NRC approves what level of physical testing or analysis is necessary for each cask design. Because NRC generally accepts the results of scale-model testing, more expensive full-scale testing rarely occurs, although NRC sometimes requires such tests for specific cask components. For example, NRC could accept quarter-scale drop tests for a particular cask design but full-scale tests of the cask's impact limiters. Computer analysis could be sufficient for meeting the hypothetical fire and criticality control criteria.

H.6 Emergency Response

H.6.1 ROLES AND RESPONSIBILITIES

States and tribes along shipping routes have the primary responsibility for the protection of the public and environment in their jurisdictions. If an emergency that involved a DOE radioactive materials shipment occurred, incident command would be established based on the procedures and policies of the state, tribe, or local jurisdiction. When requested by civil authorities, DOE would provide technical advice and assistance including access to teams of experts in radiological monitoring and related technical areas. DOE staffs eight Regional Coordinating Offices 24 hours a day, 365 days a year with teams of nuclear engineers, health physicists, industrial hygienists, public affairs specialists, and other professionals (Section H.6.2 contains further detail on the DOE role). Under NWPA Section 180(c), DOE must provide technical assistance and funds to states for training for public safety officials of appropriate units of local government and American Indian tribes through whose jurisdiction DOE plans to transport spent nuclear fuel or high-level radioactive waste. Training must cover procedures for safe routine transportation of these materials as well as for emergency response situations.

DOE would require selected carriers to provide drivers and train crews with specific written procedures that defined detailed actions for an emergency or incident that involved property damage, injury, or the release or potential release of radioactive materials. Procedures would comply with U.S. Department of Transportation guidelines for emergency response in the 2004 *Emergency Response Guidebook* (DIRS 175728-DOT 2004, all) and would address emergency assistance to injured crew or others who were involved in identification and assessment of the situation, notification and communication requirements, securing of the site and controlling access, and technical help to first responders.

H.6.2 FEDERAL COORDINATION

The Department of Homeland Security coordinates the overall Federal Government response to radiological Incidents of National Significance in accordance with Homeland Security Presidential Directive 5 (DIRS 182271-DHS 2003, all) and the *National Response Plan* (DIRS 175729-DHS 2004, all). Based on Directive 5 criteria, an Incident of National Significance is an actual or potential high-impact event that requires a coordinated and effective response by, and appropriate combination of, federal, state, local, tribal, nongovernmental, or private-sector entities to save lives and minimize damage, and to provide the basis for long-term community recovery and mitigation activities.

In Directive 5, the President designates the Secretary of Homeland Security as the Principal Federal Official for domestic incident management and empowers the Secretary to coordinate federal resources used in response to terrorist attacks, major disasters, or other emergencies in specific cases (DIRS 182271-DHS 2003, all). The Directive establishes a single, comprehensive National Incident Management System that unifies federal, state, territorial, tribal, and local lines of government into one coordinated effort. This system encompasses much more than the Incident Command System, which is nonetheless a critical component of the National Incident Management System. That system also provides a common foundation for training and other preparedness efforts, communicating and sharing information with other responders and with the public, ordering resources to assist with a response effort, and integrating new technologies and standards to support incident management. The Incident Command System uses as its base the local first responder protocols; that use does not eliminate the required agreements and coordination among all levels of government.

In Directive 5 (DIRS 182271-DHS 2003, all), the President directed the development of the new *National Response Plan* (DIRS 175729-DHS 2004, all) to align federal coordination structures, capabilities, and resources into a unified approach to domestic incident management. The Plan is built on the template of the National Incident Management System. The Plan provides a comprehensive, all-hazards approach to domestic incident management. All federal departments and agencies must adopt the National Incident Management System and use it in their individual domestic incident management and emergency prevention, preparedness, response, recovery, and mitigation activities, as well as in support of all actions taken to assist state or local entities.

DOE supports the Department of Homeland Security as the coordinating agency for incidents that involve the transportation of radioactive materials by or for DOE. DOE is otherwise responsible for the radioactive material, facility, or activity in the incident. DOE is part of the Unified Command, which is an application of the Incident Command System for when there is more than one agency with incident jurisdiction or when incidents cross political jurisdictions. DOE coordinates the federal radiological response activities as appropriate. Agencies work together through the designated members of the Unified Command, often the senior person from agencies or disciplines that participate in the Unified Command, to establish a common set of objectives and strategies.

DOE, as the transporter of radiological material, would notify state and tribal authorities and the Homeland Security Operations Center. The Department of Homeland Security and DOE coordinate federal response and recovery activities for the radiological aspects of an incident. DOE reports information and intelligence in relation to situational awareness and incident management to the Homeland Security Operations Center.

The Department of Homeland Security and DOE are responsible for coordination of security activities for federal response operations. While spent nuclear fuel and high-level radioactive waste shipments are in transit, state, local, and tribal governments could provide security for a radiological transportation incident that occurred on public lands. The Department of Homeland Security, with DOE as the coordinating agency, approves issuance of all technical data to state, local, and tribal governments.

The Interagency Modeling and Atmospheric Assessment Center, is responsible for production, coordination, and dissemination of consequence predictions for an airborne hazardous material release. The Center generates the single federal prediction of atmospheric dispersions and their consequences using the best available resources.

Federal monitoring and assessment activities are coordinated with state, local, and tribal governments. Federal agency plans and procedures for implementation of this activity are designed to be compatible with the radiological emergency planning requirements for state and local governments, specific facilities, and existing memoranda of understanding and interagency agreements.

DOE maintains national and regional coordination offices at points of access to federal radiological emergency assistance. Requests for Radiological Assessment Program teams go directly to the DOE Emergency Operations Center in Washington, D.C. If the situation requires more assistance than a team can provide, DOE alerts or activates additional resources. DOE can respond with additional resources including the Aerial Measurement System to provide wide-area radiation monitoring and Radiation Emergency Assistance Center/Training Site medical advisory teams. Some participating federal agencies have radiological planning and emergency responsibilities as part of their statutory authority, as well as established working relationships with state counterparts. The monitoring and assessment activity, which DOE coordinates, does not alter these responsibilities but complements them by providing coordination of the initial federal radiological monitoring and assessment response activities.

The U.S. Department of Homeland Security and DOE, as the coordinating agency, oversee the development of Federal Protective Action Recommendations. In this capacity, they provide advice and assistance to state, tribal, and local governments, which can include advice and assistance on measures to avoid or reduce exposure of the public to radiation from a release of radioactive material and advice on emergency actions such as sheltering and evacuation.

State, local, and tribal governments are encouraged to follow closely the *National Response Plan* (DIRS 175729-DHS 2004, all), the Nuclear/Radiological Incident Annex, and the National Incident Management System protocols and procedures. As established, all federal, state, local and tribal responders agree to and follow the Incident Command System.

H.7 Technical Assistance and Funding for Training of State and American Indian Public Safety Officials

The NWPA requires DOE to provide technical assistance and funds to states and American Indian tribes for training public safety officials of appropriate units of local governments through whose jurisdictions the Department plans to transport spent nuclear fuel or high-level radioactive waste. Section 180(c) further provides that training must cover procedures for safe routing and emergency response situations. Section 180(c) encompasses all modes of transportation, and funding would come from the Nuclear Waste Fund. Once implemented, this program would provide funding and technical assistance to train firefighters, law enforcement officers, and other public safety officials in preparation for repository shipments through their jurisdictions.

To implement this requirement in the 1990s, DOE published four Federal Register notices to solicit public comment on its approach to implementing Section 180(c). DOE responded to the comments in subsequent notices through April 1998. In 2004, the changes in homeland security and DOE transportation practices made it timely for DOE to renew efforts to develop Section 180(c) policy and implementation procedures. DOE evaluated changes in emergency preparedness and funding for responders since 1998 as well as emergency preparedness grant programs that began after September 11, 2001. The evaluation considered programs the Department of Homeland Security and the Federal Emergency Management Agency developed and relevant DOE funding and emergency response training efforts such as the Waste Isolation Pilot Plant and Foreign Research Reactor transportation programs.

The revisitation of Section 180(c) implementation began with the formation of a Transportation External Coordination Working Group Topic Group in April 2004. DOE also worked with State Regional Groups and the Tribal Issues Topic Group of the Transportation External Coordination Working Group to solicit stakeholder input on the policy. Topic Group members wrote issue papers on specific Section 180(c) topics such as allowable activities, funding allocation method, timing and eligibility, and definitions. From these materials, DOE developed a draft policy that it issued in a Federal Register notice on July 23, 2007 (72 FR 40139) to request additional comments from stakeholders and the public. DOE plans to conduct a pilot test of the program and then issue the final Section 180(c) policy.

Under the proposed policy, DOE would make two grants available to eligible state and tribal governments. An initial assessment and planning grant would be available about 4 years before shipments through a jurisdiction began. Once the state or tribe completed the assessment and planning grant activities, they would be eligible for the training grant every year that shipments traveled through their jurisdiction.

H.8 Transportation Security

Transportation safeguards and security are among the highest DOE priorities as it plans for shipments of spent nuclear fuel and high-level radioactive waste to Yucca Mountain. DOE would build the security program for the shipments on the successful security program it developed and has successfully used in past decades for shipments of spent nuclear fuel to DOE facilities from foreign and domestic reactors.

An effective security program must protect members of the public near transportation routes as well as minimize potential threats to workers, and it must include security elements appropriate to each phase of transportation. DOE would continually test security procedures to identify improvements in the security system throughout transportation operations. The key elements of a secure transportation program include physical security systems, information security, materials control and accounting, personnel security, security program management, and emergency response capabilities.

DOE is working closely with other federal agencies including NRC and the Department of Homeland Security to understand and mitigate potential threats to shipments. In addition to domestic efforts, the Department is a member of the International Working Group on Sabotage for Transport and Storage Casks, which investigates the consequences of a potential act of sabotage and explores opportunities to enhance the physical protection of casks. As a result of these efforts, DOE would modify its methods and systems as appropriate between now and the time of shipments.

In coordination with other federal agencies, DOE is working with other stakeholders including state, local, and tribal governments; industry associations such as the Association of American Railroads, and technical advisory and oversight organizations such as the National Academies of Science and the Nuclear Waste Technical Review Board. This enables DOE to take advantage of the experience and practical recommendations of experts on a broad range of security-related technical, procedural, and operational matters.

H.9 Liability

The *Price-Anderson Act* provides indemnification for liability for nuclear incidents that apply to the proposed Yucca Mountain repository. The following sections address specific details or provisions of the Act.

H.9.1 THE PRICE-ANDERSON ACT

In 1957, Congress enacted the *Price-Anderson Act* as an amendment to the *Atomic Energy Act* to encourage the development of a commercial nuclear industry and to ensure prompt and equitable compensation in the event of a nuclear incident. The *Price-Anderson Act* establishes a system of financial protection for persons who could be liable for and persons who could be injured by a nuclear incident. The purposes of the Act are (1) to encourage growth and development of the nuclear industry through the increased participation of private industry and (2) to protect the public by ensuring that funds are available to compensate victims for damages and injuries sustained in the event of a nuclear incident. Congress renewed and amended the indemnification provisions in 1966, 1969, 1975, and 1988. The 1988 *Price-Anderson Amendments Act* extended the Act for 14 years until August 1, 2002 (Public Law 100-408, 102 Stat. 1066). Since then, Congress has extended the Act until December 31, 2025, and increased liability to $10.26 billion for an extraordinary nuclear occurrence (that is, any nuclear incident that causes substantial damage), subject to increase for inflation.

H.9.2 INDEMNIFICATION UNDER THE PRICE-ANDERSON ACT

For each shipper, DOE must include an agreement of indemnification in each contract that involves the risk of a nuclear incident. This indemnification (1) provides omnibus coverage of all persons who could be legally liable, (2) fully indemnifies all legal liability up to the statutory limit on such liability (currently $10.26 billion for a nuclear incident in the United States), (3) covers all DOE contractual activity that could result in a nuclear incident in the United States, (4) is not subject to the usual limitation on the availability of appropriated funds, and (5) is mandatory and exclusive.

H.9.3 COVERED AND EXCLUDED INDEMNIFICATION

The *Price-Anderson Act* indemnifies liability arising out of, or resulting from, a nuclear incident or precautionary evacuation, including all reasonable additional costs incurred by a state or a political subdivision of a state, in the course of responding to a nuclear incident or a precautionary evacuation. It excludes (1) claims under state or federal worker compensation acts of indemnified employees or persons who are at the site of, and in connection with, the activity where the nuclear incident occurs, (2) claims that arise out of an act of war, and (3) claims that involve certain property on the site.

H.9.4 PRICE-ANDERSON ACT DEFINITION OF A NUCLEAR INCIDENT

A nuclear incident is any occurrence, including an extraordinary nuclear occurrence, that causes bodily injury, sickness, disease, death, loss of or damage to property, or loss of use of property, that arises out of or results from the radioactive, toxic, explosive, or other hazardous properties of source, special nuclear, or byproduct material (42 U.S.C. 2014).

H.9.5 PROVISIONS FOR PRECAUTIONARY EVACUATION

A precautionary evacuation is an evacuation of the public within a specified area near a nuclear facility or the transportation route in the case of an incident that involves transportation of source material, special nuclear material, byproduct material, spent nuclear fuel, high-level radioactive waste, or transuranic waste. It must be the result of an event that is not classified as a nuclear incident but poses an imminent danger of injury or damage from the radiological properties of such nuclear materials and causes an evacuation. The evacuation must be initiated by an official of a state or a political subdivision of a state who is authorized by state law to initiate such an evacuation and who reasonably determined that such an evacuation was necessary to protect the public health and safety.

H.9.6 AMOUNT OF INDEMNIFICATION

The *Price-Anderson Act* establishes a system of private insurance and federal indemnification to ensure compensation for damage or injuries suffered by the public in a nuclear incident. The current amount of $10.26 billion reflects a threshold level beyond which Congress would review the need for additional payment of claims in the case of a nuclear incident with catastrophic damage. The limit for incidents that occur outside the United States is $500 million, and the nuclear material must be owned by, and used by or under contract with, the United States.

H.9.7 INDEMNIFICATION OF TRANSPORTATION ACTIVITIES

DOE indemnifies any nuclear incident that arises in the course of any transportation activities in connection with a DOE contractual activity, including transportation of nuclear materials to and from DOE facilities.

H.9.8 COVERED NUCLEAR WASTE ACTIVITIES

The indemnification specifically includes nuclear waste activities that DOE undertakes in relation to the storage, handling, transportation, treatment, disposal of, or research and development on spent nuclear fuel, high-level radioactive waste, or transuranic waste. It would cover liability for incidents that could occur while wastes were in transit from nuclear power plants, at a storage facility, or at Yucca Mountain. If a DOE contractor or other indemnified person was liable for the nuclear incident or a precautionary evacuation that resulted from its contractual activities, that person would be indemnified for that liability. While DOE tort liability would be determined under the *Federal Tort Claims Act* (28 U.S.C. 1346(b), 1402(b), 2401(b), and 2671 through 2680), the Department would use contractors to transport spent nuclear fuel and high-level radioactive waste and to construct and operate a repository. Moreover, if public liability arose out of activities that the Nuclear Waste Fund supported, the Fund would pay compensation up to the maximum amount of protection. The NWPA established the fund to support federal activities for the disposal of spent nuclear fuel and high-level radioactive waste.

H.9.9 INDEMNIFICATION FOR STATE, AMERICAN INDIAN, AND LOCAL GOVERNMENTS

State, American Indian, and local governments are persons in the sense that they might be indemnified if they incur legal liability. The *Price-Anderson Act* defines a person as including "(1) any individual, corporation, partnership, firm, association, trust, estate, public or private institution, group, government agency other than [DOE or the Nuclear Regulatory] Commission, any state or any political subdivision of, or any political entity within a state, any foreign government or nation or any political subdivision of any such government or nation, or other entity; and (2) any legal successor, representative, agent, or agency of the foregoing" (42 U.S.C. 2214). A state or a political subdivision of a state could be entitled to indemnification for legal liability, which would include all reasonable additional costs of responding to a nuclear incident or an authorized precautionary evacuation. In addition, indemnified persons could include contractors, subcontractors, suppliers, shippers, transporters, emergency response workers, health professional personnel, workers, and victims.

H.9.10 PROCEDURES FOR CLAIMS AND LITIGATION

Numerous provisions ensure the prompt availability and equitable distribution of compensation, which would include emergency assistance payments, consolidation and prioritization of claims in one federal court, channeling of liability to one source of funds, and waiver of certain defenses in the event of a large incident. The *Price-Anderson Act* authorizes payments for immediate assistance after a nuclear incident. In addition, it provides for the establishment of coordinated procedures for the prompt handling, investigation, and settlement of claims that result from a nuclear incident.

H.9.11 FEDERAL JURISDICTION OVER CLAIMS

The U.S. District Court for the district in which a nuclear incident occurred would have original jurisdiction "with respect to any [suit asserting] public liability...without regard to the citizenship of any party or the amount in controversy" [42 U.S.C. 2210(n)]. If a case was brought in another court, it would be removed to the U.S. District Court with jurisdiction upon motion of a defendant, NRC, or DOE.

H.9.12 CHANNELING LIABILITY TO ONE SOURCE OF FUNDS

The *Price-Anderson Act* channels the indemnification (that is, the payment of claims that arise from the legal liability of any person for a nuclear incident) to one source of funds. This *economic channeling* eliminates the need to sue all potential defendants or to allocate legal liability among multiple potential defendants. Economic channeling results from the broad definition of indemnified persons to include any person who could be legally liable for a nuclear incident. Therefore, regardless of individual legal liability for a nuclear incident that resulted from a DOE contractual activity or NRC-licensed activity, the indemnity will pay the claim.

In the hearings on the original Act, "the question of protecting the public was raised where some unusual incident, such as negligence in maintaining an airplane motor, should cause an airplane to crash into a reactor and thereby cause damage to the public. Under this bill, the public is protected and the airplane company can also take advantage of the indemnification and other proceedings" (DIRS 155789-DOE 1999, p. 12).

H.9.13 LEGAL LIABILITY UNDER STATE TORT LAW

The *Price-Anderson Act* does not define legal liability, but the legislative history clearly indicates that state tort law determines the covered legal liabilities (DIRS 155789-DOE 1999, p. A-6). In 1988, *public liability action* was defined to state explicitly that "the substantive rules for decision in such action shall be derived from the law of the state in which the nuclear incident involved occurs, unless such law is inconsistent with the provisions of [Section 2210 of Title 42]" (42 U.S.C. 2014).

H.9.14 PROVISIONS WHERE STATE TORT LAW MAY BE WAIVED

The *Price-Anderson Act* includes provisions to minimize protracted litigation and to eliminate the need to prove the fault of or to allocate legal liability among various potential defendants. Certain provisions of state law may be superseded by uniform rules that the Act prescribes, such as a limitation on punitive damages. In the case of an extraordinary nuclear occurrence, the Act imposes strict liability by requiring the waiver of any defenses in relation to conduct of the claimant or fault of any indemnified person. Such waivers would result, in effect, in strict liability, the elimination of charitable and governmental immunities, and the substitution of a 3-year discovery rule in place of statutes of limitations that would normally bar all suits after a specified number of years.

H.9.15 COVERAGE AVAILABLE FOR INCIDENTS IF THE PRICE-ANDERSON ACT DOES NOT APPLY

If an incident does not involve the actual release of radioactive materials or a precautionary evacuation is not authorized, *Price-Anderson Act* indemnification does not apply. If the indemnification does not apply, liability is determined under state law, as it would be for any other type of transportation incident. Private insurance could apply. As noted above, however, the Act would cover all DOE contracts for transportation of spent nuclear fuel and high-level radioactive waste to a repository for nuclear incidents and precautionary evacuations. Indemnified persons under that DOE contractual activity would include the contractors, subcontractors, suppliers, state, American Indian, and local governments, shippers and transporters, emergency response workers, and all other workers and victims.

Carriers would have private insurance to cover liability from a nonnuclear incident and for environmental restoration for such incidents. The *Motor Carrier Act* (42 U.S.C. 10927) and its implementing regulations (49 CFR Part 387) require all motor vehicles that carry spent nuclear fuel or high-level radioactive waste to maintain financial responsibility of at least $5 million. Federal law does not require rail, barge, or air carriers of radioactive materials to maintain liability coverage, but these carriers often voluntarily cover such insurance. Private insurance policies often exclude coverage of nuclear incidents. Therefore, private insurance policies generally apply only to the extent that the *Price-Anderson Act* is not applicable.

H.10 National Academy of Sciences Findings and Recommendations

In 2006, the National Academy of Sciences Committee on Transportation of Radioactive Waste issued *Going the Distance? The Safe Transport of Spent Nuclear and High-Level Radioactive Waste in the United States* (DIRS 182032-National Research Council 2006, all). The following sections provide the findings and recommendations from this report that are relevant to this Repository SEIS along with a discussion of the DOE position on or approach to the aspects of the findings and recommendations.

H.10.1 TRANSPORTATION SAFETY AND SECURITY

Principal Academy Finding on Transportation Safety

> The committee could identify no fundamental technical barriers to the safe transport of spent nuclear fuel and high-level radioactive waste in the United States. Transport by highway (for small-quantity shipments) and by rail (for large-quantity shipments) is, from a technical viewpoint, a low-radiological-risk activity with manageable safety, health, and environmental consequences when conducted with strict adherence to existing regulations. However, there are a number of social and institutional challenges to the successful initial implementation of large-quantity shipping programs that will require

> expeditious resolution as described in this report. Moreover, the challenges of sustained implementation should not be underestimated.

DOE agrees that the transportation of spent nuclear fuel and high-level radioactive waste has a low radiological risk with manageable safety. DOE also agrees that there are social and institutional challenges, but the Department believes it would meet these challenges successfully through a process that has transportation safety as its priority.

Principal Academy Finding on Transportation Security

> Malevolent acts against spent fuel and high-level waste shipments are a major technical and societal concern, especially following the September 11, 2001, terrorist attacks on the United States. The committee judges that some of its recommendations for improving transportation safety might also enhance transportation security. The Nuclear Regulatory Commission is undertaking a series of security studies, but the committee was unable to perform an in-depth technical examination of transportation security because of information constraints.

Academy Recommendation

> An independent examination of the security of spent fuel and high-level waste transportation should be carried out prior to the commencement of large-quantity shipments to a federal repository or to interim storage. This examination should provide an integrated evaluation of the threat environment, the response of packages to credible malevolent acts, and operational security requirements for protecting spent fuel and high-level waste while in transport. This examination should be carried out by a technically knowledgeable group that is independent of the government and free from institutional and financial conflicts of interest. This group should be given full access to the necessary classified documents and Safeguards Information to carry out this task. The findings and recommendations from this examination should be made available to the public to the fullest extent possible.

Transportation safeguards and security are among DOE's highest priorities as it plans for shipments of spent nuclear fuel and high-level radioactive waste to the proposed repository. The Department would build the security program for the repository shipments on the security program that it has developed and successfully used in past decades for shipments of spent nuclear fuel to DOE facilities from foreign and domestic reactors.

An effective security program must protect members of the public near transportation routes as well as potential threats to workers, and it must include security elements appropriate to each phase of transportation. Continual testing of security procedures would result in improvements in the security system through completion of transportation operations for Yucca Mountain. The most important elements of a secure transportation program include physical security systems, information security, materials control and accounting, personnel security, security program management, and emergency response capabilities.

DOE is working closely with other Federal agencies including the NRC, and the U.S. Department of Homeland Security, and the Transportation Security Agency to understand and eliminate potential threats to repository shipments. In addition to its domestic efforts, the Department is a member of the

International Working Group on Sabotage for Transport and Storage Casks, which is investigating the consequences of a potential act of sabotage and is exploring opportunities to enhance the physical protection of casks. As a result of these efforts, DOE would modify its methods and systems as appropriate between now and the time of shipments.

In coordination with other Federal agencies, DOE is working with other stakeholders including state, tribal, and local governments; industry associations such as the Association of American Railroads and technical advisory and oversight organizations such as the National Academy of Sciences and the Nuclear Waste Technical Review Board. This allows DOE to take advantage of the experience and practical recommendations of experts on a broad range of security-related technical, procedural, and operational matters.

H.10.2 TRANSPORTATION RISK

Academy Finding

There are two types of transportation risk: health and safety risks and social risks. The health and safety risks arise from the potential exposure of transportation workers as well as other people who travel, work, or live near transportation routes to radiation that may be emitted or released from these loaded packages. Social risks arise from social processes and human perceptions and can have both direct socioeconomic impacts and perception-based impacts.

There are two potential sources of radiological exposures from transporting spent fuel and high-level waste: (1) radiation shine from spent fuel and high-level waste transport packages under normal transport conditions; and (2) potential increases in radiation shine and release of radioactive materials from transport packages under accident conditions that are severe enough to compromise fuel element and package integrity. The radiological risks associated with the transportation of spent fuel and high-level waste are well understood and are generally low, with the possible exception of risks from releases in extreme accidents involving very long duration, fully engulfing fires. While the likelihood of such extreme accidents appears to be very small, their occurrence cannot be ruled out based on historical accident data for other types of hazardous material shipments. However, the likelihood of occurrence and consequences can be reduced further through relatively simple operational controls and restrictions and route-specific analyses to identify and mitigate hazards that could lead to such accidents.

Academy Recommendation

To address radiological risk, the NAS stated there were clear transportation operations and safety advantages to be gained from shipping older (i.e. radiologically and thermally cooler) spent fuel first.

Transportation planners and managers should undertake detailed surveys of transportation routes to identify potential hazards that could lead to or exacerbate extreme accidents involving very long duration, fully engulfing fires. Planners and managers should also take steps to avoid or mitigate such hazards before the commencement of shipments or shipping campaigns.

This Repository SEIS evaluated the radiological risks of transportation accidents and found these risks to be very low, as did the Yucca Mountain FEIS. In addition, NRC has evaluated the response of spent nuclear fuel casks to the environments that existed during the Baltimore tunnel fire and the Caldecott tunnel fire, which would be representative of long duration, fully engulfing fires. These evaluations show that releases of radioactive material during these types of events, if they occurred at all, would be very small. Based on recommendations from the NRC, the Association of American Railroads has modified its operating standards to prohibit trains that carry flammable materials from being in a tunnel at the same time as a train that carries spent fuel. This administrative adjustment addresses some of the concerns of the Academy.

An initial step in the planning process to ship spent nuclear fuel and high-level radioactive waste to the Yucca Mountain repository would be to identify a national suite of rail and highway routes. Stakeholder groups in the DOE transportation program are participating in this process by examining routing criteria that DOE could use in the route identification process. State Regional Groups, American Indian tribes, transportation associations, industry, Federal agencies, and local government organizations are some of the groups that work collaboratively with DOE in this process.

Academy Finding

> The social risks for spent fuel and high-level waste transportation pose important challenges to the successful implementation of programs for transporting spent fuel and high-level waste in the United States. Such risks have received substantially less attention than health and safety risks, and some are difficult to characterize. Current research and practice suggest that transportation planners and managers can take early proactive steps to characterize, communicate, and manage the social risks that arise from their operations. Such steps may have additional benefits: they may increase the openness and transparency of transportation planning and programs; build community capacity to mitigate these risks; and possibly increase trust and confidence in transportation programs.

Academy Recommendation

> Transportation implementers should take early and proactive steps to establish formal mechanisms for gathering high-quality and diverse advice about social risks and their management on an ongoing basis. The committee makes two recommendations for the establishment of such mechanisms for the Department of Energy's program to transport spent fuel and high-level waste to a federal repository at Yucca Mountain: (1) expand the membership and scope of an existing advisory group (Transportation External Coordination Working Group; see Chapter 5) to obtain outside advice on social risk, including impacts and management; and (2) establish a transportation risk advisory group that is explicitly designed to provide advice on characterizing, communicating, and mitigating the social, security, and health and safety risks that arise from the transportation of spent fuel and high-level waste to a federal repository or interim storage. This group should be comprised of risk experts and practitioners drawn from the relevant technical and social science disciplines and should be convened under the Federal Advisory Committee Act or a similar arrangement to enhance the openness of its operations. Its members should receive security clearances to facilitate access to appropriate transportation security information. The existing federal Nuclear Waste Technical Review Board, which will cease operations no later than one year after the

Department of Energy begins disposal of spent fuel or high-level waste in a repository, could be broadened to serve this function.

DOE has reviewed the Academy recommendation on involving social scientists in the Transportation External Coordination Working Group and on expert panels, and the Department has contacted some panel members to explore opportunities for future studies. DOE has sponsored studies by social scientists in the past on risk perception about transportation of radioactive materials and adjusted its programs to focus on local officials and support for emergency planning and training as a result. The Department needs to update this study and is in the process of reviewing literature to understand gaps in research to address some of the most pressing transportation issues. In addition, DOE has proposed a topic group within the Transportation External Coordination Working Group to address social risks. The Working Group membership has not yet indicated if that is an area they want to focus on at this time.

H.10.3 CURRENT CONCERNS ABOUT TRANSPORTATION OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE

H.10.3.1 *Package Performance*

Academy Finding

Transportation packages play a crucial role in the safety of spent fuel and high-level radioactive waste shipments by providing a robust barrier to the release of radiation and radioactive material under both normal transport and accident conditions. International Atomic Energy Agency package performance standards and associated Nuclear Regulatory Commission regulations are adequate to ensure package containment effectiveness over a wide range of transport conditions, including most credible accident conditions. However, recently published work suggests that extreme accident scenarios involving very long duration, fully engulfing fires might produce thermal loading conditions sufficient to compromise containment effectiveness. The consequences of such thermal loading conditions for containment effectiveness are the subject of ongoing investigations by the Nuclear Regulatory Commission and other parties, and this work is improving the understanding of package performance. Nonetheless, additional analyses and experimentation are needed to demonstrate a bounding-level understanding of package performance in response to very long duration, fully engulfing fires for a representative set of package designs.

Academy Recommendation

The Nuclear Regulatory Commission should build on recent progress in understanding package performance in very long duration fires. To this end, the agency should undertake additional analyses of very long duration fire scenarios that bound expected real world accident conditions for a representative set of package designs that are likely to be used in future large-quantity shipping programs. The objectives of these analyses should be to:

- Understand the performance of package barriers (spent fuel cladding and package seals);

- Estimate the potential quantities and consequences of any releases of radioactive material; and

- Examine the need for regulatory changes (e.g., package testing requirements) or operational changes (e.g., restrictions on trains carrying spent fuel) either to help prevent accidents that could lead to such fire conditions or to mitigate their consequences.

> Strong consideration should also be given to performing well-instrumented tests for improving and validating the computer models used for carrying out these analyses, perhaps as part of the full-scale test planned by the Nuclear Regulatory Commission for its package performance study. Based on the results of these investigations, the Commission should implement operational controls and restrictions on spent fuel and high-level radioactive waste shipments as necessary to reduce the chances that such fire conditions might be encountered in service. Such effective steps might include, for example, additional operational restrictions on trains carrying spent fuel and high-level radioactive waste to prevent co-location with trains carrying flammable materials in tunnels, in rail yards, and on sidings.

As Section H.10.2 notes, NRC has addressed operating restrictions for tunnels by working with the Association of American Railroads to adjust rail operating practices. In addition, DOE has committed to supporting the NRC Package Performance Study to better understand severe accidents.

Academy Finding

> The committee strongly endorses the use of full-scale testing to determine how packages will perform under both regulatory and credible extra-regulatory conditions. Package testing in the United States and many other countries is carried out using good engineering practices that combine state-of-the-art structural analyses and physical tests to demonstrate containment effectiveness. Full-scale testing is a very effective tool both for guiding and validating analytical engineering models of package performance and for demonstrating the compliance of package designs with performance requirements. However, deliberate full-scale testing of packages to destruction through the application of forces that substantially exceed credible accident conditions would be marginally informative and is not justified given the considerable costs for package acquisitions that such testing would require.

Academy Recommendation

> Full-scale package testing should continue to be used as part of integrated analytical, computer simulation, scale-model, and testing programs to validate package performance. Deliberate full-scale testing of packages to destruction should not be required as part of this integrated analysis or for compliance demonstrations.

DOE would use NRC-certified casks for transportation of spent nuclear fuel and high-level radioactive waste to the proposed repository. Cask vendors would supply these NRC-certified casks to DOE under contractual requirements. To obtain the certificate, the vendors would conduct testing as NRC specifies.

H.10.3.2 *Route Selection for Research Reactor Spent Fuel Transport*

Academy Finding

> The Department of Energy's procedures for selecting routes within the United States for shipments of foreign research reactor spent fuel appear on the whole to be adequate and reasonable. These procedures are risk informed; they make use of standard risk assessment methodologies in identifying a suite of potential routes and then make final route selections by taking into account security, state and tribal preferences, and information from states and tribes on local transport conditions. The Department of Energy's procedures reflect the agency's position (which is consistent with Department of Transportation regulations) that the states are competent and responsible for selecting highway routes. For rail route selection, the Department of Energy's practice of negotiating routes with carriers in consultation with states is analogous to its interaction with states on highway routing.

Academy Recommendation

> The Department of Energy should continue to ensure the systematic, effective involvement of states and tribal governments in its decisions involving routing and scheduling of foreign and DOE research reactor spent fuel shipments.

For shipments to the repository, DOE would use its *Strategic Plan for the Safe Transportation of Spent Nuclear Fuel and High-Level Radioactive Waste to Yucca Mountain: A Guide to Stakeholder Interactions* (DIRS 172433-DOE 2003, all) to guide interactions with state and tribal governments. During planning and actual transportation operations, DOE would involve these stakeholders in route identification, funding approaches for emergency response planning and training, understanding safeguards and security requirements, operational practices, and communications and information access.

DOE is working collaboratively with states through State Regional Group committees (whose members are state officials responsible for transportation policy, law enforcement, emergency response, and oversight of hazardous materials shipments) and with American Indian tribal governments to assist them to prepare for the shipments.

In addition to State Regional Group and tribal coordination, a national cooperative effort is underway as part of the Transportation External Coordination Working Group and its various Topic Groups, which involves a broad range of stakeholder organizations that routinely interact with DOE to provide input and recommendations on transportation planning and program information. States, tribes, and industry are working with DOE to guide and focus emergency training, coordination with local officials, and other transportation activities to prepare for shipments to the repository.

Academy Finding

> Highway routes for shipment of spent nuclear fuel are dictated by DOT regulations (49 CFR Part 397). The regulations specify that shipments normally must travel by the fastest route using highways designated by the states or the federal government. They do not require the carrier or shipper to evaluate risks of portions of routes that meet this criterion. These regulations are a satisfactory means of ensuring safe transportation, provided that the shipper actively and systematically consults with the states and tribes

along potential routes and that states follow the route designation procedures prescribed by the DOT.

Academy Recommendation

DOT should ensure that states that designate routes for shipment of spent nuclear fuel rigorously comply with its regulatory requirement that such designations be supported by sound risk assessments. DOT and DOE should ensure that all potentially affected states are aware of and prepared to fulfill their responsibilities regarding highway route designations.

DOE is working collaboratively with states through State Regional Group committees (whose members are state officials responsible for transportation policy, law enforcement, emergency response, and oversight of hazardous materials shipments) and with American Indian tribal governments to assist them to prepare for the shipments.

As part of the routing discussions, DOE has provided training to officials of these stakeholders on its routing model (TRAGIS; DIRS 181276-Johnson and Michelhaugh 2003, all) and the risk model (RADTRAN 5; DIRS 150898-Neuhauser and Kanipe 2000, all). If states or tribes choose to designate alternative highway routes, technical assistance is available from the experts at the national laboratories who manage these two models. In addition, State Regional Group staff support their states with routing assistance as part of the cooperative efforts DOE supports.

H.10.4 FUTURE CONCERNS FOR TRANSPORTATION OF SPENT NUCLEAR FUEL AND HIGH-LEVEL RADIOACTIVE WASTE

H.10.4.1 *Mode for Transporting Spent Nuclear Fuel and High-Level Radioactive Waste to a Federal Repository*

Academy Finding

Transport of spent fuel and high-level waste by rail has clear safety, operational, and policy advantages over highway transport for large-quantity shipping programs. The committee strongly endorses DOE’s selection of the “mostly rail” option for the Yucca Mountain transportation program for the following reasons:

- It reduces the total number of shipments to the federal repository by roughly a factor of five, which reduces the potential for routine radiological exposures, conventional traffic accidents, and severe accidents.
- Rail shipments have a greater physical separation from other vehicular traffic and reduced interactions with people along transportation routes, which also contributes to safety.
- Operational logistics are simpler and more efficient.
- There is a clear public preference for this option.

The committee does not endorse the development of an extended truck transportation program to ship spent fuel cross-country or within Nevada should DOE fail to complete construction of the Nevada rail spur or procure the necessary rail equipment by the time the federal repository is opened.

Academy Recommendation

DOE should fully implement its mostly rail decision by completing construction of the Nevada rail spur, obtaining the needed rail packages and conveyances, and working with commercial spent fuel owners to ensure that facilities are available at plants to support this option. These steps should be completed before DOE commences the large-quantity shipment of spent fuel and high-level waste to a federal repository to avoid the need to procure infrastructure and construct facilities to support an extended truck transportation program. DOE should also examine the feasibility of further reducing its needs for cross-country truck shipments of spent fuel through the expanded use of intermodal transportation (i.e., combining heavy-haul truck, legal-weight truck, and barge) to allow the shipment of rail packages from plants that do not have direct rail access.

In this Repository SEIS, DOE analyzed the intermodal transfer of rail casks for generator sites that do not have direct rail access. The SEIS analysis identified nine such sites from which DOE would ship spent nuclear fuel or high-level radioactive waste using 2,650 truck shipments. In addition, DOE's transportation operational planning recognizes the value of barge and some heavy-haul truck shipments to maximize rail use to ship to the repository. DOE would address all modes of transportation in future transportation campaign plans.

H.10.4.2 *Route Selection for Transportation to a Federal Repository*

Academy Finding

DOE has not made public a specific plan for selecting rail and highway routes for transporting spent fuel and high-level waste to a federal repository. DOE also has not determined the role of its program management contractors in selecting routes or specific plans for collaborating with affected states, tribes, and other parties.

Academy Recommendation

DOE should identify and make public its suite of preferred highway and rail routes for transporting spent fuel and high-level waste to a federal repository as soon as practicable to support state, tribal, and local planning, especially for emergency responder preparedness. DOE should follow the practices of its foreign research reactor spent fuel transport program of involving states and tribes in these route selections to obtain access to their familiarity with accident rates, traffic and road conditions, and emergency responder preparedness within their jurisdictions. Involvement by states and tribes may improve the public acceptability of route selections and may reduce conflicts that can lead to program delays.

An initial step in the DOE planning process to ship spent nuclear fuel and high-level radioactive waste to the proposed Yucca Mountain repository would be to identify a national suite of routes, both rail and highway, that DOE could use. Stakeholder groups are participating with DOE in this process by examining routing criteria the Department could use in the route identification process. State Regional Groups, American Indian tribes, transportation associations, industry, federal agencies, and local government organizations are some of the groups that work collaboratively with DOE in this process.

The work would be conducted through a topic group of the Transportation External Coordination Working Group. Broader public input would also be sought to collect comments on routing criteria and the process for developing a set of routes. Industry practices, DOE requirements, and analyses of regional routes that were evaluated by state organizations would be included in the process to identify a preliminary set of routes. Public involvement is central to contributing to a safe, efficient, and flexible transportation system.

H.10.4.3 *Use of Dedicated Trains for Transport to a Federal Repository*

Academy Finding

Studies carried out to date on transporting spent fuel by dedicated versus general trains have failed to show a clear radiological risk based advantage for either option. However, the committee finds that there are clear operational, safety, security, communications, planning, programmatic, and public preference advantages that favor dedicated trains. The committee strongly endorses DOE's decision to transport spent fuel and most high-level waste to a federal repository using dedicated trains.

Academy Recommendation

DOE should fully implement its dedicated train decision before commencing the large-quantity shipment of spent fuel and high-level waste to a federal repository to avoid the need for a stop gap shipping program using general trains.

DOE made a decision to use dedicated trains for its usual mode of shipment, which offers benefits that include efficient use of casks and rail cars, lower dwell time in rail yards and, in combination with other service features, direct service from origin to destination. DOE agrees with the Academy's recommendation.

H.10.4.4 *Acceptance Order for Commercial Spent Nuclear Fuel Transport to a Federal Repository*

Academy Finding

The order for accepting commercial spent fuel that is mandated by the Nuclear Waste Policy Act (NWPA) was not designed with the transportation program in mind. In fact, the acceptance order prescribed by the NWPA could require DOE to initiate its transportation program with long cross-country movements of younger (i.e., radiologically and thermally hotter) spent fuel from multiple commercial sites. There are clear transportation operations and safety advantages to be gained from shipping older (i.e., radiologically and thermally cooler) spent fuel first and for initiating the transportation program with relatively short, logistically simple movements to gain experience and build operator and public confidence.

Academy Recommendation

DOE should negotiate with commercial spent fuel owners to ship older fuel first to a federal repository or federal interim storage, except in cases (if any) where spent fuel storage risks at specific plants dictate the need for more immediate shipments of younger fuel. Should these negotiations prove to be ineffective, Congress should consider legislative remedies. Within the context of its current contracts with commercial spent fuel owners, DOE should initiate transport through a pilot program involving relatively short, logistically simple movements of older fuel from closed reactors to demonstrate the ability to carry out its responsibilities in a safe and operationally effective manner. DOE should use the lessons learned from this pilot activity to initiate its full-scale transportation program from operating reactors.

The terms of the "Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste" (10 CFR Part 961) require DOE to assign priority to those generator sites whose fuel was discharged earliest. This is usually called the "Oldest Fuel First" priority. DOE must pick up fuel from sites that were designated by those generators as those with the oldest fuel regardless of the location. At sites that were designated by the generators who own the oldest spent nuclear fuel, DOE must pick up fuel the generators have selected and that has cooled for at least 5 years.

Regardless of which fuel DOE would ship first, it would conduct the shipments safely in NRC-certified casks for that type of fuel.

H.10.4.5 *Emergency Response Planning and Training*

Academy Finding

Emergency responder preparedness is an essential element of safe and effective programs for transporting spent fuel and high-level waste. Emergency responder preparedness has so far received limited attention from DOE, states, and tribes for the planned transportation program to the federal repository. DOE has the opportunity to be innovative in carrying out its responsibilities for emergency responder preparedness. Emergency responders are among the most trusted members of their communities. Well-trained responders can become important emissaries for DOE's transportation program in local communities and can enhance community preparedness to respond to other kinds of emergencies.

Academy Recommendation

DOE should begin immediately to execute its emergency responder preparedness responsibilities defined in Section 180(c) of the Nuclear Waste Policy Act. In carrying out these responsibilities, DOE should proceed to (1) establish a cadre of professionals from the emergency responder community who have training and comprehension of emergency response to spent fuel and high-level waste transportation accidents and incidents; (2) work with the Department of Homeland Security to provide consolidated "all-hazards" training materials and programs for first responders that build on the existing national emergency response platform; (3) include trained emergency responders on the escort teams that accompany spent fuel and high-level waste shipments; and (4) use emergency responder preparedness programs as an outreach mechanism to communicate broadly about plans and programs for transporting spent fuel and high-level waste to a federal repository with communities along planned shipping routes.

The NWPA requires DOE to provide technical assistance and funds to states and American Indian tribes for training public safety officials of appropriate units of local governments through whose jurisdictions the Department plans to transport spent nuclear fuel or high-level radioactive waste. Section 180(c) further provides that training cover procedures required for safe routing transportation of these materials, as well as procedures for dealing with emergency response situations. Section 180(c) indicates that funding for work under this subsection would come from the Nuclear Waste Fund. Once implemented, this program would provide the increment of funding and technical assistance necessary to train fire fighters, law enforcement officers, and other public safety officials in preparation for repository shipments through their jurisdictions.

To implement this requirement in the 1990s, DOE published four Federal Register notices soliciting public comments on its approach to implementing Section 180(c). Comments received in response to these notices were addressed in each subsequent Federal Register notice with the last notice issued in April 1998. In 2004, the changes in homeland security and DOE's transportation practices made it timely for DOE to renew efforts to develop Section 180(c) policy and implementation procedures. Changes in emergency preparedness and funding for responders since 1998 were reviewed and evaluated as well as emergency preparedness grant programs initiated after September 11, 2001. Programs developed by Department of Homeland Security and the Federal Emergency Management Agency were considered. Relevant DOE funding and emergency response training efforts such as the Waste Isolation Pilot Plant and Foreign Research Reactor transportation programs were also evaluated.

DOE's revisiting of Section 180(c) implementation began with the formation of the Transportation External Coordination Working Group 180(c) Topic Group in April 2005. DOE also worked with the state regional groups and the Tribal Issues Topic Group of the Transportation External Coordination Working Group to solicit stakeholder input on the policy. Topic Group members wrote issue papers on specific Section 180(c) topics such as allowable activities, funding allocation method, timing and eligibility, and definitions. From these materials, DOE developed a draft policy which it issued in a Federal Register Notice on July 23, requesting additional comments from stakeholders and the public. DOE plans to conduct a pilot to test the program, and then issue the final Section 180(c) Policy.

Under the proposed policy, two grants would be made available to eligible state and tribal governments. An initial assessment and planning grant would be available about four years prior to shipments commencing through a jurisdiction. Once the state or tribe completes the assessment and planning grant activities, they would be eligible for the training grant every year that shipments travel through their jurisdiction.

H.10.4.6 *Information Sharing and Openness*

Academy Finding

> There is a conflict between the open sharing of information on spent fuel and high-level waste shipments and the security of transportation programs. This conflict is impeding effective risk communication and may reduce public acceptance and confidence. Post–September 11, 2001, efforts by transportation planners, managers, and regulators to further restrict information about spent fuel shipments make it difficult for the public to assess the safety and security of transportation operations.

Academy Recommendation

> The Department of Energy, Department of Homeland Security, Department of Transportation, and Nuclear Regulatory Commission should promptly complete the job of developing, applying, and disclosing consistent, reasonable, and understandable criteria for protecting sensitive information about spent fuel and high-level waste transportation. They should also commit to the open sharing of information that does not

require such protection and should facilitate timely access to such information: for example, by posting it on readily accessible Web sites.

Interactions with state and tribal governments would be guided by the *Office of Civilian* Radioactive *Waste Management Strategic Plan for the Safe Transportation of Spent Nuclear Fuel and High-Level Radioactive Waste to Yucca Mountain: A Guide to Stakeholder Interactions* (DIRS 172433-DOE 2003, all). During planning and actual transportation operations, states, tribes, industry, and other key stakeholders would be involved in route identification, funding approaches for emergency response planning and training, understanding safeguards and security requirements, operational practices, and communications and information access.

In addition to key stakeholder organizations and groups, the public has access to transportation information through the DOE web page and through the Transportation External Coordination Working Group web page. These two mechanisms allow program information that should be shared reach a broad audience.

H.10.4.7 *Organizational Structure of the Federal Transportation Program*

Academy Finding

Successful execution of DOE's program to transport spent fuel and high-level waste to a federal repository will be difficult given the organizational structure in which it is embedded, despite the high quality of many current program staff. As currently structured, the program has limited flexibility over commercial spent fuel acceptance order (Section 5.2.4); it also has limited control over its budget and is subject to the annual federal appropriations process, both of which affect the program's ability to plan for, procure, and construct the needed transportation infrastructure. Moreover, the current program may have difficulty supporting what appears to be an expanding future mission to transport commercial spent nuclear fuel for interim storage or reprocessing. In the committee's judgment, changing the organizational structure of this program will improve its chances for success.

Academy Recommendation

The Secretary of Energy and the U.S. Congress should examine options for changing the organizational structure of the Department of Energy's program for transporting spent fuel and high-level waste to a federal repository. The following three alternative organizational structures, which are representative of progressively greater organizational change, should be specifically examined: (1) a quasi-independent DOE office reporting directly to upper-level DOE management; (2) a quasi-government corporation; or (3) a fully private organization operated by the commercial nuclear industry. The latter two options would require changes to the Nuclear Waste Policy Act. The primary objectives in modifying the structure should be to give the transportation program greater planning authority; greater budgetary flexibility to make the multiyear commitments necessary to plan for, procure, and construct the necessary transportation infrastructure; and greater flexibility to support an expanding future mission to transport spent fuel and high-level waste for interim storage or reprocessing. Whatever structure is selected, the organization should place a strong emphasis on operational safety and reliability and should be responsive to social concerns.

The NWPA defines the Federal Government's responsibilities for disposal of spent nuclear fuel and high-level radioactive waste. The NWPA created the Office of Civilian Radioactive Waste Management within DOE to carry out these responsibilities, which include the development of a transportation system. The Act requires the Office to maximize use of the private sector to implement its transportation responsibilities. That collaborative development effort is underway, and would continue until the law changed.

REFERENCES

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166338	AAR 2003	AAR (Association of American Railroads) 2003. *Performance Specification for Trains Used to Carry High-Level Radioactive Material.* AAR S-2043. Washington, D.C.: Association of American Railroads. TIC: 257585.
175727	AAR 2005	AAR (Association of American Railroads) 2005. *Field Manual of the Interchange Rules.* Change No. 05-1. Washington, D.C.: Association of American Railroads. TIC: 257725; 257789.
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181841	Adkins et al. 2007	Adkins, H.E., Jr.; Koeppel, B.J.; Cuta, J.M.; Guzman, A.D.; and Bajwa, C.S. 2007. *Spent Fuel Transportation Package Response to the Caldecott Tunnel Fire Scenario.* NUREG/CR-6894, Rev. 1. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20070724.0370.
175725	CVSA 2005	CVSA (Commercial Vehicle Safety Alliance) 2005. *North American Standard Out-of-Service Criteria.* Handbook and Out-of-Service Pictorial Edition. Washington, D.C.: Commercial Vehicle Safety Alliance. TIC: 257711.
182271	DHS 2003	DHS (U.S. Department of Homeland Security) 2003. "Management of Domestic Incidents." Homeland Security Presidential Directive/HSPD-5. Washington, D.C.: U.S. Department of Homeland Security. Accessed August 2, 2007. ACC: MOL.20070809.0003. URL: http://www.dhs.gov/xnews/releases/press_release_0105.shtm

175729	DHS 2004	DHS (U.S. Department of Homeland Security) 2004. *National Response Plan.* Washington, D.C.: U.S. Department of Homeland Security. ACC: MOL.20051020.0132.
155789	DOE 1999	DOE (U.S. Department of Energy) 1999. *Report to Congress on the Price-Anderson Act*. Washington, D.C.: U.S. Department of Energy. ACC: MOL.20011009.0012.
155970	DOE 2002	DOE (U.S. Department of Energy) 2002. *Final Environmental Impact Statement for a Geological Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada*. DOE/EIS- 0250. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20020524.0314; MOL.20020524.0315; MOL.20020524.0316; MOL.20020524.0317; MOL.20020524.0318; MOL.20020524.0319; MOL.20020524.0320.
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172433	DOE 2003	DOE (U.S. Department of Energy) 2003. *Strategic Plan for the Safe Transportation of Spent Nuclear Fuel and High-Level Radioactive Waste to Yucca Mountain: A Guide to Stakeholder Interactions*. Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20041206.0113.
175728	DOT 2004	DOT (U.S. Department of Transportation) 2004. *2004 Emergency Response Guidebook.* Washington, D.C.: U.S. Department of Transportation. ACC: MOL.20051020.0131.
156703	FRA 1998	FRA (Federal Railroad Administration) 1998. *Safety Compliance Oversight Plan for Rail Transportation of High-Level Radioactive Waste and Spent Nuclear Fuel*. Washington, D.C.: U.S. Department of Transportation. ACC: MOL.20011212.0115.
182833	Golan 2005	Golan, P. 2005. "Letter Informing Stakeholders of Policy Adopted by the Department to Use Dedicated Trains for its Shipment of Spent Nuclear Fuel and High-Level Radioactive Waste to the Yucca Mountain Repository." Letter from P. Golan to Stakeholder, July 18, 2005, with attachment. ACC: HQO.20051003.0001.
181276	Johnson and Michelhaugh 2003	Johnson, P.E. and Michelhaugh, R.D. 2003. *Transportation Routing Analysis Geographic Information System (TRAGIS) User's Manual.* ORNL/NTRC-006. Oak Ridge, Tennessee: Oak Ridge National Laboratory.

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182897	Reyes 2005	Reyes, L.A. 2005. "Details and Projected Cost of a Demonstration Test of a Full-Scale Spent Nuclear Fuel Rail Transportation Cask Under the Package Performance Study." Policy Issue Notation from L.A. Reyes to the Commissioners. Marcy 28, 2005.
152476	Sprung et al. 2000	Sprung, J.L.; Ammerman, D.J.; Breivik, N.L.; Dukart, R.J.; Kanipe, F.L.; Koski, J.A.; Mills, G.S.; Neuhauser, K.S.; Radloff, H.D.; Weiner, R.F.; and Yoshimura, H.R. 2000. *Reexamination of Spent Fuel Shipment Risk Estimates*. NUREG/CR-6672. Two volumes. Washington, D.C.: U.S. Nuclear Regulatory Commission. ACC: MOL.20001010.0217.
182896	Vietti-Cook 2005	Vietti-Cook, A.L. 2005. "Details and Projected Cost of a Demonstration Test of a Full-Scale Spent Nuclear Fuel Rail Transportation Cask Under the Package Performance Study." Commission Voting Record from A. L. Vietti-Cook to Distribution, June 9, 2005, with enclosures.

Appendix I

Federal Register Notices

APPENDIX I. FEDERAL REGISTER NOTICES

The following table lists Federal Register Notices used in this *Draft 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-S1D). Notices can be found on the U.S. Government Printing Office GPO Access website at http://origin.www.gpoaccess.gov/fr/.

Volume and Page	Publication Date	Title
60 FR 28680	June 1, 1995	Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs.
66 FR 14194	March 9, 2001	Notice of Realty Action: Public Law 106-113, as Amended, Non-Competitive Sale of Public Lands and the Conveyance of Public Lands for Recreation and Public Purposes.
67 FR 39737	June 10, 2002	Nye County Habitat Conservation Plan for Lands Conveyed at Lathrop Wells, NV.
67 FR 53359	August 15, 2002	Public Land Order No. 7534; Extension of Public Land Order No. 6802; Nevada.
67 FR 63167	October 10, 2002	In the Matter of All Power Reactor Licensees, Research and Test Reactor Licensees, and Special Nuclear Material Licensees Who Possess and Ship Spent Nuclear Fuel; Order Modifying License. (Effective Immediately)
67 FR 65539	October 25, 2002	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.
67 FR 65564	October 25, 2002	Environmental Impact Statements; Notice of Availability.
67 FR 79906	December 31, 2002	Record of Decision for the Final Environmental Impact Statement for the Relocation of Technical Area 18 Capabilities and Materials at the Los Alamos National Laboratory.
68 FR 58815	October 10, 2003	Electronic Maintenance and Submission of Information; Final Rule. (Part 63—Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada.)
68 FR 74951	December 29, 2003	Notice of Preferred Nevada Rail Corridor.
68 FR 74965	December 29, 2003	Notice of Proposed Withdrawal and Opportunity for Public Meeting; Nevada.
69 FR 2280	January 14, 2004	Changes to Adjudicatory Process 10 CFR Parts 1, 2, 50, 51, 52, 54, 60, 63, 70, 72, 73, 75, 76, and 110. (Part 63-- Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada.)

Volume and Page	Publication Date	Title
69 FR 3698	January 26, 2004	Compatibility With IAEA Transportation Safety Standards (TS–R–1) and Other Transportation Safety Amendments.
69FR 18557	April 8, 2004	Record of Decision on Mode of Transportation and Nevada Rail Corridor for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, NV.
69 FR 18565	April 8, 2004	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 22496	April 26, 2004	Comment Period Extension and Additional Public Scoping Meetings for 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 52040	August 24, 2004	Policy Statement on the Treatment of Environmental Justice Matters in NRC Regulatory and Licensing Actions.
69 FR 58841	October 1, 2004	Hazardous Materials Regulations; Compatibility With the Regulations of the International Atomic Energy Agency; Correction; Final Rule.
70 FR 35073	June 16, 2005	West Valley Demonstration Project Waste Management Activities.
70 FR 49014	August 22,2005	Public Health and Environmental Radiation Protection Standards for Yucca Mountain, Nevada; Proposed Rule.
70 FR 56647	September 28,2005	Notice of Intent To Prepare a Programmatic Environmental Impact Statement, Amend Relevant Agency Land Use Plans, Conduct Public Scoping Meetings, and Notice of Floodplain and Wetlands Involvement.
70 FR 75165	December 19, 2005	Office of Environmental Management; Record of Decision for the Idaho High-Level Waste and Facilities Disposition Final Environmental Impact Statement.
70 FR 76854	December 28, 2005	Public Land Order No. 7653; Withdrawal of Public Lands for the Department of Energy To Protect the Caliente Rail Corridor; Nevada.
71 FR 10068	February 28, 2006	Notice of Issuance of Materials License Snm–2513 for the Private Fuel Storage Facility.
71 FR 60484	October 13, 2006	Amended Notice of Intent To Expand the Scope of the Environmental Impact Statement for the Alignment, Construction, and Operation of a Rail Line to a Geologic Repository at Yucca Mountain, Nye County, NV.

Volume and Page	Publication Date	Title
71 FR 60490	October 13, 2006	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 61731	October 19, 2006	Notice of Intent To Prepare a Supplement to the Stockpile Stewardship and Management Programmatic Environmental Impact Statement—Complex 2030.
71 FR 65785	November 9, 2007	Extension of Public Comment Period and Additional Public Meeting for the Supplemental Yucca Mountain Rail Corridor and Rail Alignment Environmental Impact Statement.
71 FR 65786	November 9, 2006	Extension of Public Comment Period and Additional Public Meeting for the 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.
72 FR 331	January 4, 2007	Notice of Intent To Prepare a Programmatic Environmental Impact Statement for the Global Nuclear Energy Partnership.
72 FR 1235	January 10, 2007	Notice of Proposed Withdrawal and Opportunity for Public Meeting; Nevada.
72 FR 14543	March 28, 2007	Notice of Intent To Prepare a Supplemental Environmental Impact Statement for Surplus Plutonium Disposition at the Savannah River Site.
72 FR 40135	July 23, 2007	Notice of Intent To Prepare an Environmental Impact Statement for the Disposal of Greater-Than-Class-C Low-Level Radioactive Waste.

Appendix J

Distribution List

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J. DISTRIBUTION LIST

J.1 United States Congress

The U.S. Department of Energy is providing copies of this *Draft 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-S1D) to federal, state, and local elected and appointed officials and agencies of government; American Indian groups; national, state, and local environmental and public interest groups; and other organizations and individuals listed below. Copies will be provided to other interested parties upon request.

J.1.1 UNITED STATES SENATORS FROM NEVADA

The Honorable John E. Ensign
U.S. Senator
United States Senate

The Honorable Harry Reid
Senate Majority Leader
United States Senate

J.1.2 UNITED STATES REPRESENTATIVES FROM NEVADA

The Honorable Shelley Berkley
1st District Representative
U.S. House of Representatives

The Honorable Jon C. Porter, Sr.
3rd District Representative
U.S. House of Representatives

The Honorable Dean A. Heller
2nd District Representative
U.S. House of Representatives

J.1.3 UNITED STATES SENATE COMMITTEES

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Chairman
Senate Committee on Energy & Natural Resources

The Honorable Thad Cochran
Ranking Member
Senate Committee on Appropriations

The Honorable James Inhofe
Ranking Member
Senate Committee on Environment & Public Works

The Honorable Carl Levin
Chairman
Senate Committee on Armed Services

The Honorable John S. McCain
Vice Chairman
Senate Armed Services Committee

The Honorable Bernard Sanders
Senate Committee on Environment & Public Works

The Honorable John Warner
Senate Committee on Armed Services
Senate Committee on Environment & Public Works

The Honorable Robert C. Byrd
Chairman
Senate Committee on Appropriations

The Honorable Pete V. Domenici
Ranking Member
Senate Committee on Energy & Natural Resources

The Honorable Daniel K. Inouye
Chairman
Senate Committee on Commerce, Science & Transportation
Subcommittee on Surface Transportation & Merchant Marine

The Honorable Trent Lott
Senate Committee on Commerce, Science & Transportation
Subcommittee on Surface Transportation & Merchant Marine Infrastructure, Safety & Security

The Honorable Ted Stevens
Vice Chairman
Senate Committee on Commerce, Science & Transportation

J.1.4 UNITED STATES HOUSE OF REPRESENTATIVES COMMITTEES

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Ranking Minority Member
House Committee on Energy & Commerce

The Honorable John D. Dingell
Chairman
House Committee on Energy & Commerce

The Honorable David Hobson
Ranking Member
House Committee on Appropriations
Subcommittee on Energy & Water Development

The Honorable David Obey
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House Committee on Appropriations

The Honorable Peter J. Visclosky
House Committee on Appropriations
Subcommittee on Energy & Water Development

The Honorable Rick Boucher
House Committee on Energy & Commerce
Subcommittee on Energy & Air Quality

The Honorable Ralph M. Hall
House Committee on Energy & Commerce
Subcommittee on Energy & Air Quality

The Honorable Duncan Hunter
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House Committee on Armed Services

The Honorable Jerry Lewis
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J.2 Federal Agencies

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Ms. Barbara Durham
Tribal Representative
Timbisha Shoshone Tribe

Mr. Atef Elzeftawy
Tribal Representative
Las Vegas Paiute Tribe

Mr. Ron Escobar
Tribal Representative
Chemehuevi Tribe

The Honorable John Feliz, Jr.
Chairwoman
Coyote Valley Band of Pomo Indians of California

Mr. Maurice Frank-Churchill
Tribal Representative
Duckwater Shoshone Tribe

The Honorable Elizabeth Hansen
Chairwoman
Redwood Valley Rancheria of Pomo Indians of California

Mr. Bill Helmer
Tribal Historic Preservation Officer
Big Pine Pauite Tribe of the Owens Valley

The Honorable Mike Jackson, Sr.
President
Quechan Tribe of the Fort Yuma Indian Reservation, California & Arizona

Ms. Clara Belle Jim
Tribal Representative
Pahrump Paiute Tribe

Mr. Gerald Kane
Tribal Representative
Bishop Paiute Indian Tribe

The Honorable Roland E. Johnson
Governor
Pueblo of Laguna, New Mexico

The Honorable Jason Johnson
Governor
Pueblo of Acoma, New Mexico

Ms. Lawanda Laffoon
Tribal Representative
Colorado River Indian Tribes

Mr. A. David Lester
Executive Director
Council of Energy Resource Tribes

Ms. Cynthia V. Lynch
Tribal Representative
Pahrump Paiute Tribe

Ms. Dorena Martineau
Tribal Representative
Paiute Indian Tribes of Utah

The Honorable Arlan D. Melendez
Tribal Chair
Reno-Sparks Indian Tribe

Mr. Calvin Meyers
Tribal Representative
Moapa Band of Paiutes

Ms. Lalovi Miller
Tribal Representative
Moapa Band of Paiutes

The Honorable Antone Minthorn
Chairman, Board of Trustees
Confederated Tribes of the Umatilla Reservation, Oregon

The Honorable Alfreda L. Mitre
Chairwoman
Las Vegas Paiute Tribe

The Honorable Virgil Moose
Chairman
Big Pine Paiute Tribe of the Owens Valley

The Honorable Larry Nuckolls
Governor
Absentee Shawnee Tribe of Indians of Oklahoma

The Honorable Kay Rhoads
Principal Chief
Sac & Fox Nation, Oklahoma

The Honorable Tony Salazar
Chairman
Kickapoo Tribe of Oklahoma

The Honorable Joseph C. Saulque
Chairman
Benton Paiute Indian Tribe

The Honorable George Scott
Town King
Thlopthlocco Tribal Town, Oklahoma

The Honorable Arturo Senclair
Governor
Ysleta Del Sur Pueblo of Texas

Mr. Herman Shorty
Navajo Nation

The Honorable Barry E. Snyder, Sr.
President
Seneca Nation of New York

The Honorable Ronald Suppah
Chairman
Confederated Tribes of the Warm Springs Reservation

Ms. Eleanor Tom
Tribal Representative
Paiute Indian Tribes of Utah

The Honorable Lora E. Tom
Chairwoman
Paiute Indian Tribe of Utah

Ms. Rebecca Van Lieshout
Coordinator, Tribal EPA
Forest County Potawatomi Community of Wisconsin

Mr. Ken Watteron
Timbisha Shoshone Tribe

Mr. Richard Wilder
Tribal Representative
Fort Independence

The Honorable Charles Wood
Chairman
Chemehuevi Indian Tribe

Genia Williams
Tribal Chairperson
Walker River Paiute Tribe

J.6 Environmental and Public Interest Groups

Ms. Beth Gallegos
Citizens Against Contamination

Mr. Robert Holden
Director
National Congress of American Indians (NCAI)
Emergency Management & Radioactive Programs

Mr. Toney Johnson
Citizens Against Nuclear Trash

Mr. David Albright
President
Institute for Science and International Security

Ms. Kathryn Landreth
State Director
Nevada Field Office
The Nature Conservancy

Mr. Thomas A. Schatz
President
Council for Citizens Against Government Waste

Mr. David Beckman
Natural Resources Defense Council
Los Angeles Office

Ms. Joan B. Claybrook
President
Public Citizen

Mr. John Flicker
Chief Executive Officer
National Audubon Society

Dave Lochbaum, Ph.D.
Sr. Nuclear Safety Engineer
Union of Concerned Scientists

Mr. Paul R. Portney
President & Senior Fellow
Resources for the Future

Mr. John Tanner
Coalition 21

Mr. John M. Bailey
Institute for Local Self-Reliance

Ms. Mavis Belisle
Director
Peace Farm

Mr. David Bradley
Executive Director
National Community Action Foundation

Mr. Jim C. Bridgman
Program Director
Alliance for Nuclear Accountability

Mr. David Brunner
Chief Operating Officer
National Fish & Wildlife Foundation

Ms. Jodi Dart
Alliance for Nuclear Accountability

Ms. Anna M. Frazier
Coordinator
Dine CARE

Mr. Tom Goldtooth
Executive Director
Indigenous Environmental Network

Ms. Lisa Gover
Program Director
National Tribal Environmental Council (NTEC)

Ms. Thea Harvey
Executive Director
Economists for Peace & Security at the Levy Institute

Mr. Daniel Hirsch
President
Committee to Bridge the Gap

Ms. Traci Laird
Regional Coordinator
Sierra Club
Southern Plains Regional Field Office

Reverend Mac Legerton
Center for Community Action

Mr. Jim Lyon
Senior Director for Congressional & Federal Affairs
National Wildlife Foundation

Mr. Michael McCally
Physicians for Social Responsibility

Ms. Elizabeth Merritt
Deputy General Counsel
National Trust for Historic Preservation

Dr. LeRoy Moore
Consultant on Radiation Health Issues
Rocky Mountain Peace & Justice Center

Mr. Steve Moyer
Vice President for Governmental Affairs
Trout Unlimited

Ms. Martez Norris
Administrator
Nuclear Waste Strategy Coalition

Mr. Daniel R. Patterson
Desert Ecologist
Center for Biological Diversity

Mr. Roger Rivera
President
National Hispanic Environmental Council

Nick Roth
Nuclear Age Peace Foundation

Mr. Paul Schwartz
National Campaign Director
Clean Water Action

Ms. Alice Slater
President
Global Resource Action Center for the Environment

Mr. Gerald Pollet
Executive Director
Heart of America Northwest

Mr. Richard J. Sawicki
The Wilderness Society
Ecology & Economics Research Department

Ms. Kassie Siegel
Air Climate and Energy Director
Center for Biological Diversity

Jennifer Olaranna Viereck
Healing Ourselves & Mother Earth
Mr. Jim Riccio
Greenpeace International

Ms. Meg Power
Senior Advisor
National Community Action Foundation

Mr. Fred Krupp
President
National Headquarters
Environmental Defense

Mr. Albert (Brandt) Petrasek
State and Tribal Government Working Group

Ms. Carolyn Hanson
Environmental Council of the States

Mr. Mark Wenzler
Sierra Club
Legislative Office

Mr. John H. Adams
President
Natural Resources Defense Council

Ms. Carol Browner
Chairman of the Board
National Audubon Society

Mr. Larry Fahn
President, Board of Directors
Sierra Club

Mr. Pete Litster
Executive Director
Shundahai Network

Mr. Steven J. McCormick
President & Chief Executive Officer
The Nature Conservancy

Ms. Sara Szynelski
Program Manager
Energy Communities Alliance

Ms. Joni Arends
Executive Director
Concerned Citizens for Nuclear Safety (CCNS)

Mr. Chuck Broscious
Executive Director
Environmental Defense Institute

Dr. Robert D. Bullard
Director
Clark Atlanta University
Environmental Justice Resource Center

Ms. Kateri Callahan
President
Alliance to Save Energy

Ms. Laura Carlsen
Director, Americas Program
International Relations Center

Mrs. Nina Carter
Executive Director, Washington State Office
National Audubon Society

Mr. Thomas Cassidy
Director of Federal Programs
The Nature Conservancy

Ms. Christine Chandler
Responsible Environment Action League

Ms. Janet Feldman
Sierra Club

Mr. Bob Fulkerson
Progressive Leadership Alliance of Nevada

Ms. Susan Gordon
Director
Alliance for Nuclear Accountability

Ms. Nicole Graysmith
Legal Aid of North Carolina
Environmental Poverty Law Project

Ms. Janet Greenwald
Citizens for Alternatives to Radioactive
Dumping (CARD)

Mr. Don Hancock
Director, Nuclear Waste Safety Program &
SRIC Administrator
Southwest Research & Information Center

Ms. Amelia Hellman
State Director
The Nature Conservancy
Nevada Field Office

Ms. Marylia Kelley
Tri-Valley CAREs

Ms. Tonya Kleuskens
President
STAND, Inc.

Mr. Ronald Lamb
Coalition for Health Concern

Mr. Lloyd Leonard
League of Women Voters

Mr. Kevin Martin
Executive Director
Peace Action Education Fund

Dr. Mildred McClain
Executive Director
Harambee House, Inc.
Projects: ACA-Net & Citizens for
Environmental Justice

Mr. Allen Metscher
Citizens Advisory Council

Mr. Richard Moore
Executive Director
Southwest Network for Environmental &
Economic Justice

Robert K. Musil, Ph.D.
Executive Director and CEO
Physicians for Social Responsibility

Mr. Ehrich Pica
Friends of the Earth

Ms. Laura Raicovich
Dia Art Foundation

Ms. Susan Shaer
Executive Director
Women's Action for New Directions

Ms. Gail Small
Executive Director
Native Action

Mr. Derek Stack
Executive Director
Great Lakes United

Ms. Peggy Maze Johnson, Executive Director
Citizens Alert

J.7 Other Groups and Individuals

Mr. Ralph Anderson
Project Manager, Plant Support
Nuclear Energy Institute

Mr. & Mrs. Dirk and Marta Agee
Commissioner
Tempiute Grazing Association

Mr. Steven Kerekes
Director, Media Relations
Nuclear Energy Institute

Mr. William Ramsey
Executive Director
Troutman Sanders

Dr. Budhi Sagar
Technical Director
Southwest Research Institute
Center for Nuclear Waste Regulatory Analyses

Dr. Stephen Wells
President
Desert Research Institute

Ms. Barbara Bauman Tyran
Director, Washington Relations
Electric Power Research Institute

Ms. Anna Aurilio
Director, Washington, DC Office
U.S. Public Interest Research Group

Thomas B. Cochran, Ph.D.
Director, Nuclear Programs
Natural Resources Defense Council, Inc.
Washington Office

Mr. David Hawkins
Director, Climate Center
Natural Resources Defense Council
Washington Office

Mr. Ace Robison
Consultant

Dr. David Shafer
Executive Director
CERM, Desert Research Institute
Center for Environmental Remediation & Monitoring

Mr. Robert A. Beck
Executive Vice President
National Coal Council

Mr. W. Scott Field
Policy Analyst
Western Interstate Energy Board (WIEB)

Ms. Angelina S. Howard
Executive Vice President
Nuclear Energy Institute

Mr. Steven P. Kraft
Director, Spent Nuclear Fuel Management
Nuclear Energy Institute

Mr. Richard M. Loughery
Director, Environmental Activities
Edison Electric Institute

Mr. Robert Robinson
Senior Economist
Center for Applied Research

Ms. Kara Colton
National Governors' Association (NGA)
Environment, Energy & Natural Resources Division

Ms. Shauna Larsen
Government Relations Representative
American Public Power Assoc.

Mr. Edward W. Lent, III
Region III President
International Association of Emergency Managers
c/o General Physics Corporation

Mr. Robert E. Marvin
Director
FRA/State Rail Safety Participation Program
c/o Transportation Division
Ohio Public Utilities Commission

Mr. William T. Pound
Executive Director
National Conference of State Legislatures

Mr. Jim B. Reed
Transportation Program Director
National Conference of State Legislators (NCSL)

Ms. Lisa R. Sattler
Senior Policy Analyst
Council of State Governments-Midwest Office (CSG/MW)

Ms. Diane Shea
Director, Natural Resources Committee
National Governors' Association

Mr. Joe Strolin
Western Interstate Energy Board
Mr. Robert Thompson
Energy Communities Alliance

Mr. Dave Trebisacci, CSP
Senior Fire Service Specialist
National Fire Protection Association (NFPA)

Mr. Chris Turner
Training Coordinator
International Association of Fire Fighters (IAFF)
Hazardous Materials Training Department

Ms. Julie Ufner
Associate Legislative Director
National Association of Counties (NACo)

Donald H. Baepler, Ph.D.
Executive Director
University of Nevada, Las Vegas
Harry Reid Center

Ms. Jill Kennay
Assistant Director
Natural Land Institute

Ms. Abigail C. Johnson
Consultant
Abigail C. Johnson Consulting

Ms. Mary Olson
Radioactive Waste Project & NIX MOX Campaign
Nuclear Information & Resource Service

Dr. Klaus Stetzenbach
Director
University of Nevada, Las Vegas
Harry Reid Center for Environmental Studies

ENV1 - Environmental Not VIP Jackie Cabasso
Executive Director
Western States Legal Foundation

Ms. Amy Greer
Natural Resources Defense Council
Public Education

Admiral, Retired Frank L. "Skip" Bowman
President & Chief Executive Officer
Nuclear Energy Institute

Mr. Tom Barry
Senior Policy Analyst & Co-founder
International Relations Center

Mr. Kevin Kamps
Nuclear Information & Resource Service

Mr. Paul Leventhal
Founding President
Nuclear Control Institute

Arjun Makhijani, Ph.D.
President
Institute for Energy & Environmental Research (IEER)

Mr. Rod McCullum
Nuclear Energy Institute

Mr. Robert J. Moran
Washington Representative
American Petroleum Institute

Mr. Vincent Scoccia
Nye Regional Medical Center

Mr. Richard Bryan, Chairman
Nevada Commission on Nuclear Projects

Ms. Paula Cotter, Esq.
Senior Environmental Counsel
National Association of Attorneys General (NAAG)

Mr. Patrick Cummins
Air Quality Project Manager
Western Regional Air Partnership

Mr. Robert E. Fronczak
Assistant Vice President, Environment & Hazardous Materials
Association of American Railroads (AAR)
Safety & Operations Dept.

Mr. Walter Isaacson
President & CEO
The Aspen Institute
Program on Energy, the Environment & the Economy

Mr. Douglas Larson
Executive Director
Western Interstate Energy Board (WIEB)

Mr. William Mackie
Program Manager
Western Governors' Association

Mr. Brian J. O'Connell
Director, Nuclear Waste Program
National Assoc. of Regulatory Utility Commissioners (NARUC)

Mr. Duane Parde
Executive Director
American Legislative Exchange Council (ALEC)

Mr. Randy Rawson
President
American Boiler Manufacturers Association

Ms. Eileen Supko
Senior Consultant
Energy Resources International, Inc.

Mr. James S. Tulenko
President
American Nuclear Society

Mr. George D. Turner
President & Chief Executive Officer
American Nuclear Insurers

Mr. William Vascone

Arden L. Bement, Ph.D.
Director
National Science Foundation

Mr. Anthony DeSouza
Director, Board on Earth Studies & Natural Sciences
National Academy of Sciences

J.8 Reading Rooms and Libraries

Esmeralda County Yucca Mountain Oversight Office
Goldfield, NV

Nye County Department of Natural Resources and Federal Facilities
Pahrump, NV
U.S. Department of Energy Headquarters Office Public Reading Room
Washington, D.C.

Ms. Susan Beard
Librarian
Northern Arizona University
Cline Library

Ms. Michelle Born
Librarian
Clark County Library
Main Branch
Reference Department

Mr. Bert Chapman
Purdue University
Hesse Library - Documents Department

Ms. Pauline Conner
Administrator
Savannah River Operations
University of South Carolina-Aiken, Gregg-Graniteville Library
Public Reading Room

Ms. Kathy Edwards
Nevada State Library & Archives

Ms. Amy Sue Goodin
Associate Director of Research
University of New Mexico
Institute for Public Policy

Ms. Sandra Groleau
Bates College Library

Ms. Paige Harper
University of Florida
Documents Department

Ms. Deanna Harvey
Strategic Petroleum Reserve Project Management Office
SPRPMO/Reading Room

Mr. John Horst
National Renewable Energy Lab
Public Reading Room

Librarian, Government Documents
University of New Hampshire
Dimond Library

Librarian
San Jose State University
Martin Luther King, Jr. Library
Government Publications Department

Librarian
University of Colorado, Boulder
Library - Government Publications

Librarian
University Library, California State University
Government Documents

Librarian
Susanville District Library

Librarian
Sparks Branch Library

Librarian
Fishlake Branch Library

Librarian
Carson City Library
Reference Department

Librarian
Laughlin Branch Library

Librarian
Needles Library

Librarian
Round Mountain Public Library

Librarian
University of Wisconsin, Madison
Wendt Library - Technical Reports Center

Librarian
University of Texas, San Antonio
Library - Government Documents Department

Mrs. Pat Loper
Great Basin College Library

Mr. Joseph Milazzo
Southern Methodist University
Fondren Library East
Government Information

Ms. Janice Parthree
U.S. Department of Energy
Richland Operations Center
Public Reading Room

Mr. Tim D. Petrosky
Public Affairs Director
Consumer Energy
Big Rock Point

Mr. William H. Schalk
Communications Manager
American Electric Power Company
Cook Visitor Information Center

Ms. Susie Skarl
Head, Government Publications
University of Nevada, Las Vegas
Lied Library

Ms. Barbara Sullivan
Center for Health, Environment and Justice
CHEJ Library

Ms. T. Twyford
Librarian
Tecopa Library

Mr. John Walters
Utah State University
Merrill Library

Ms. Judy Weiss
U.S. Department of Energy
Oakland Operations Office

Mr. Joel Yudken
Sectoral Economist, Department of Public Policy
AFL-CIO

Mr. Robert A. Apel
Newcrest Resources, Inc.

W. A. Armstrong
Preservation Capital Corp

Mr. Gerald W. Baughman
Nevada Eagle Resources, LLC
Lincoln County Nuclear Waste Project Office
Caliente, NV

Pahrump Yucca Mountain Information Center
Pahrump, NV

The University of Nevada Libraries
Business and Government Information Center
Reno, NV

Mr. Dan Barkley
University of New Mexico
Zimmerman Library

Ms. Jan Bennett
Public Affairs Representative
Entergy Nuclear Vermont Yankee

Ms. Carol Brown
Office of Intergovernmental Affairs
City of Chicago

Mr. Hui Hua Chua
Michigan State University

Ms. Kay Collins
University of California, Irvine
Langson Library

Ms. Sherry DeDecker
University of California, Santa Barbara
Davidson Library - Government Information Center

Ms. Heather Elliott
Clearinghouse Coordinator, Nevada State Clearinghouse
Department of Administration
State of Nevada
Public Reading Room

Ms. Verna Graybill
Librarian
Northwestern Oklahoma State University
J.W. Martin Library Depository

Mr. Michele Hayslett
North Carolina State University Libraries
Government Information Services – RISD

Mr. Mark Holt
Library of Congress

Ms. Alisa Huckle
Mr. Brent Jacobson
Idaho Operations Office
INEEL Technical Library
U.S. DOE Public Reading Room

Mr. Walter Jones
University of Utah
Marriott Library, Special Collections

Librarian
University of Notre Dame
Hesburgh Library - Government Documents Center

Librarian
University of California, Riverside
Rivera Library - Government Publications Department

Librarian
U.S. Geological Survey
Serial Records Unit
National Center

Librarian
Spring Valley Library

Librarian
Billinghurst Middle School Library

Librarian
Inyo County Free Library

Librarian
New Mexico State University
Branson Library - Documents

Librarian
North Las Vegas Public Library
Librarian
Brainerd Memorial Library

Librarian
University of Rhode Island
Library - Government Publications Office

Librarian
University of Maryland, Baltimore County
Albin O. Kuhn Library and Gallery -
Government Documents

Mr. David Mienke
Alliant Utilities, Inc.
Duane Arnold Energy Center

Mr. Gary Morell
College Hill Library
Rocky Flats Field Office

Mr. Walter Perry
U.S. Department of Energy Information Center
Oak Ridge Operations Office

Ms. Angela Riley
Community Relations Representative
TXU Electric & Gas
Comanche Peak Visitors Center

Mr. John Shuler
Richard J. Daley Library
Documents, Maps and Microforms Department
University of Illinois at Chicago

Mr. Bobby St. John
Media Coordinator
Washington TRU Solutions

Mr. Tim Sutherland
Indiana University
Northwest Library, Documents Department

Mr. Carl E. Unger
Senior Communications Representative
Public Information Center
Arizona Public Service Company
Palo Verde Nuclear Generating Station

Ms. Elaine Watson
Boise State University
Library - Government Documents

Ms. Beryl A. Zundel
Sunrise Library

Mr. Bill Zimmerman
U.S. Department of Energy
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Mr. David Blee
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Mr. Dennis P. Bryan
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Mr. Jonathan M. Brown
Nevada Mining Association

Mr. Paul J. Burkett
Nevada Gold & Casinos, Inc.

Ms. Anne deLain Clark
Coordinator
Radioactive Waste Consultation Task Force
New Mexico Energy Minerals & Natural
Resources Dept.

Mr. John M. Engler
President
National Association of Manufacturers

Mr. Michael Govan
Dia Art Foundation

Ms. Arlene Grider
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Independence Chamber of Commerce

Ms. Sharon Kerrick
Manager
American Nuclear Society (ANS)
Outreach & Volunteer Dept.

Mr. Scott Palmer
Brotherhood of Locomotive Engineers &
Trainment

Mr. Cort Richardson
Project Director
Council of State Governments - Eastern
Regional Conference (CSG/ERC)

Mr. Steven Specker
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Mr. John J. Castellani
President
Business Roundtable

Mr. Thomas Donohue
President and CEO
U.S. Chamber of Commerce

Mr. Glenn L. English, Jr.
Chief Executive Officer
National Rural Electric Cooperative Assoc.

Mr. Irwin Forman
Formarts Corp.

Mr. Robert Alan Kemp
NV Central, Great Basin, NV Northern Railroads

Mr. Thomas R. Kuhn
President
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Mr. Michael McKloskey
International Affairs Specialist
Transport Logistics International

Mr. Alan H. Richardson
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Ms. Jennifer A. D. Smith
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Mr. Gary Vine
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