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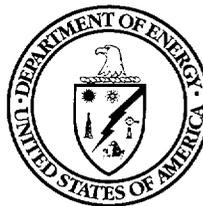
Environmental Impact Statement

for a

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



Volume II
Appendixes A through L



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

DOE/EIS-0250D

July 1999

9909150202 - Part 4

ACRONYMS AND ABBREVIATIONS

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

BWR	boiling-water reactor
CFR	Code of Federal Regulations
DOE	U.S. Department of Energy (also called <i>the Department</i>)
EIS	environmental impact statement
EPF	energy partition factor
FR	<i>Federal Register</i>
LCF	latent cancer fatality
MTHM	metric tons of heavy metal
NWPA	Nuclear Waste Policy Act, as amended
OCRWM	Office of Civilian Radioactive Waste Management
PM ₁₀	particulate matter with an aerodynamic diameter of 10 micrometers or less
PM _{2.5}	particulate matter with an aerodynamic diameter of 2.5 micrometers or less
PWR	pressurized-water reactor
UFSAR	Updated Final Safety Analysis Report
USC	United States Code

UNDERSTANDING SCIENTIFIC NOTATION

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

Positive Powers of 10

$$10^1 = 10 = 10$$

$$10^2 = 10 \times 10 = 100$$

and so on, therefore,

$$10^6 = 1,000,000 \text{ (or 1 million)}$$

Negative Powers of 10

$$10^{-1} = 1/10 = 0.1$$

$$10^{-2} = 1/100 = 0.01$$

and so on, therefore,

$$10^{-6} = 0.000001 \text{ (or 1 in 1 million)}$$

Probability is expressed as a number between 0 and 1 (0 to 100 percent likelihood of the occurrence of an event). The notation 3×10^{-6} can be read 0.000003, which means that there are three chances in 1,000,000 that the associated result (for example, a fatal cancer) will occur in the period covered by the analysis.

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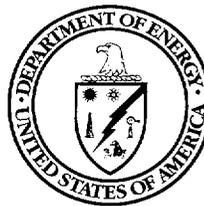
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Appendix A

Inventory and Characteristics of
Spent Nuclear Fuel, High-Level
Radioactive Waste, and Other
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APPENDIX A. INVENTORY AND CHARACTERISTICS OF SPENT NUCLEAR FUEL, HIGH-LEVEL RADIOACTIVE WASTE, AND OTHER MATERIALS

A.1 Introduction

This appendix describes the inventory and characteristics of the spent nuclear fuel and high-level radioactive waste that the U.S. Department of Energy (DOE) anticipates it would place in a monitored geologic repository at Yucca Mountain. It includes information about other highly radioactive material that DOE could dispose of in the proposed repository. It also provides information on the background and sources of the material, present storage conditions, the final disposal forms, and the amounts and characteristics of the material. The data provided in this appendix are the best available estimates of projected inventories.

The Proposed Action inventory evaluated in this environmental impact statement (EIS) consists of 70,000 metric tons of heavy metal (MTHM), comprised of 63,000 MTHM of commercial spent nuclear fuel and 7,000 MTHM of DOE materials. The DOE materials consist of 2,333 MTHM of spent nuclear fuel and 8,315 canisters (4,667 MTHM) of solidified high-level radioactive waste. The inventory includes approximately 50 metric tons (55 tons) of surplus weapons-usable plutonium as spent mixed-oxide fuel and immobilized plutonium.

The Nuclear Waste Policy Act, as amended (also called the NWPA), prohibits the U.S. Nuclear Regulatory Commission from approving the emplacement of more than 70,000 MTHM in the first repository until a second repository is in operation [Section 114(d)]. However, in addition to the Proposed Action, this EIS evaluates the cumulative impacts for two additional inventories (referred to as Inventory Modules 1 and 2):

- The Module 1 inventory consists of the Proposed Action inventory plus the remainder of the total projected inventory of commercial spent nuclear fuel, high-level radioactive waste, and DOE spent nuclear fuel. Emplacement of Inventory Module 1 wastes in the repository would raise the total amount emplaced above 70,000 MTHM. As mentioned above, emplacement of more than 70,000 MTHM of spent nuclear fuel and high-level radioactive waste would require legislative action by Congress unless a second licensed repository was in operation.
- Inventory Module 2 includes the Module 1 inventory plus the inventories of the candidate materials, commercial Greater-Than-Class-C low-level radioactive waste and DOE Special-Performance-Assessment-Required waste. There are several reasons to evaluate the potential for disposing of these candidate materials in a monitored geologic repository in the near future. Because both materials exceed Class C low-level radioactive limits for specific radionuclide concentrations as defined in 10 CFR Part 61, they are generally unsuitable for near-surface disposal. Also, the Nuclear Regulatory Commission specifies in 10 CFR 61.55(a)(2)(iv) the disposal of Greater-Than-Class-C waste in a repository unless the Commission approved disposal elsewhere. Further, during the scoping process for this EIS, several commenters requested that DOE evaluate the disposal of other radioactive waste types that might require isolation in a repository. Disposal of Greater-Than-Class-C and Special-Performance-Assessment-Required wastes at the proposed Yucca Mountain Repository could require a determination by the Nuclear Regulatory Commission that these wastes require permanent isolation. In addition, the present 70,000-MTHM limit on waste at the Yucca Mountain Repository could have to be addressed either by legislation or by opening a second licensed repository.

A.1.1 INVENTORY DATA SUMMARY

There are six general inventory categories, as follows:

- Commercial spent nuclear fuel
- DOE spent nuclear fuel
- High-level radioactive waste
- Surplus weapons-usable plutonium
- Commercial Greater-Than-Class-C waste
- DOE Special-Performance-Assessment-Required waste

This section summarizes the detailed inventory data in Section A.2. The data provide a basis for the impact analysis in this EIS. Data are provided for the candidate materials included in the initial 70,000 MTHM for the Proposed Action and other inventory that is not currently proposed but might be considered for repository disposal in the foreseeable future.

This summary provides general descriptive and historic information about each waste type, including the following:

- Primary purpose and use of the data
- General comparison of the data between waste types
- Potential for change in inventory data

Table A-1 lists the inventory data that DOE used in the EIS analyses and their descriptions throughout the document.

A.1.1.1 Sources

Figure A-1 shows the locations of generators or sources of spent nuclear fuel and high-level radioactive waste. Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation. The Proposed Action includes the disposal of 63,000 MTHM of commercial spent nuclear fuel in the repository. More than 99 percent of the commercial spent nuclear fuel would come from commercial nuclear reactor sites in 33 states (DOE 1995a, all). In addition, DOE manages an inventory of spent nuclear fuel. The Proposed Action includes 2,333 MTHM of spent nuclear fuel from four DOE locations: the Savannah River Site in South Carolina, the Hanford Site in Washington, the Idaho National Engineering and Environmental Laboratory, and Fort St. Vrain in Colorado.

High-level radioactive waste is the highly radioactive material resulting from the reprocessing or treatment of spent nuclear fuel. The Proposed Action includes disposing of 8,315 canisters of high-level radioactive waste in the repository. High-level radioactive waste is stored at the Savannah River Site, the Hanford Site, the Idaho National Engineering and Environmental Laboratory, and the West Valley Demonstration Project in New York.

The President has declared approximately 50 metric tons (55 tons) of plutonium to be surplus to national security needs (DOE 1998a, page 1-1). This surplus weapons-usable plutonium includes purified plutonium, nuclear weapons components, and plutonium residues. This inventory is included in the Proposed Action, and the Department would dispose of it as either spent mixed oxide fuel from a commercial nuclear reactor (that is, commercial spent nuclear fuel) or immobilized plutonium in a high-level radioactive waste canister (that is, as high-level radioactive waste), or a combination of these two inventory categories (DOE 1998a, page 1-3). Spent mixed-oxide fuel would come from one or more of

Table A-1. Use of Appendix A radioactivity inventory data in EIS chapters and appendixes (page 1 of 2).

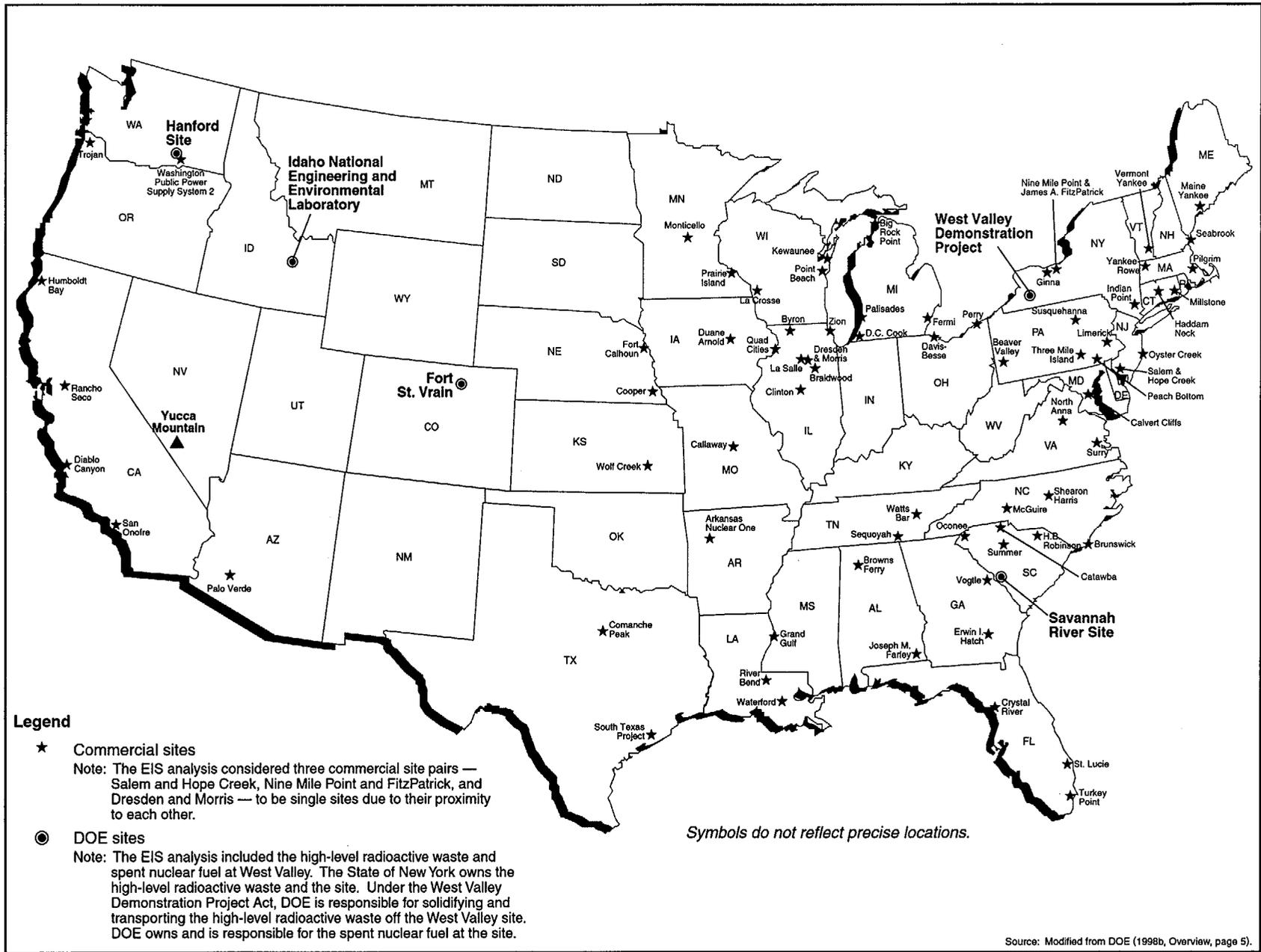
Item ^a	Appendix A	EIS section
Number of commercial nuclear sites	Table A-3	1.1, 2.2, 2.2.2, 2.4.1, 2.4.2.3, 2.4.2.4, 2.4.2.8, 2.4.3, 6.1, 7.0, 7.2.1, 7.3, J.1.3.1.1
Number of DOE sites	A.1.1	1.1, 2.2, 2.2.2, 2.4.1, 2.4.2.3, 2.4.2.4, 2.4.2.8, 2.4.3, 6.1, 7.0, 7.2.1, 7.3
Mapped location of sites	Figure A-1	Figure 1-1, Several Chapter 6, 7, App. J and K figures
Commercial SNF material	A.2.1.5.3	1.1.1
Commercial SNF dimensions	Table A-15	1.1.1, Figure 1-3, H.2.1.4
Commercial SNF cladding material	A.2.1.5.3	1.1.2.1.1, 5.2.2, K.2.1.4.1
Percentage of commercial SNF with stainless-steel cladding	A.2.1.5.3	1.1.2.1.1, 1.5.3, 5.2.2, 5.5.1, K.2.1.4.1
MOX SNF part of commercial SNF Proposed Action	A.2.4.5.1.1	1.1.2.1.1
Number of sites with existing or planned ISFSIs	Table A-4	1.1.2.1.1
Amount of commercial SNF projected for each site	Tables A-6 and A-7	1.1.2.1.1, 6.1.1, K.2.1.6
List of commercial SNF sites, state, operations period	Table A-3	Table 1-1
DOE SNF storage locations	Table A-17	1.1.2.1.2, K.2.1.6
HLW includes immobilized Pu	A.2.4.5.2.1	1.1.2.2
HLW generators	A.2.3.2	1.1.2.2, K.2.1.6
HLW vitrification status	A.2.3.4	1.1.2.2
Weapons-usable Pu declared surplus	A.2.4.1	1.1.2.3
Two forms: MOX and immobilized Pu	A.2.4.1	1.1.2.3
Proposed Action inventory	A.1	1.1.2.5, 1.3.2, 1.6.3.1, 2.1, Figure 2-3, 2.1.4, 2.2.2, 2.2.3, 5.1, 5.2.2, 5.6.3, 6.1.1.1, 7.0, 7.2, 8.1.2.1, J.1.3.1.1, J.1.3.1.2, K.2.1.6
Total projected inventory commercial SNF	Figure A-2	1.1.2.5, 1.6.3.1, 7.2, 7.3, 8.1.2.1, J.1.3.1.1, K.2.1.6
Total projected inventory DOE SNF	Figure A-2	1.1.2.5, 1.6.3.1, 6.1.1.1, 7.2, 7.3, 8.1.2.1, J.1.3.1.2, K.2.1.6
Total projected inventory HLW	Figure A-2	1.1.2.5, 1.6.3.1, 7.2, 7.3, 8.1.2.1, K.2.1.6
Total projected GTCC waste	Table A-51	1.6.3.1, 7.3, 8.1.2.1, I.3.1.2.4, J.1.3.1.3
Total projected SPAR waste	Table A-56	1.6.3.1, 7.3, 8.1.2.1, I.3.1.2.4, J.1.3.1.3
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Kr-85 (gas) is contained in fuel gap of commercial SNF	A.2.1.5.2	4.1, 4.1.2.3.2
Typical radionuclide inventory for commercial SNF	Tables A-8 and A-9	4.1.8.1, 6.1.3.2.1, H.2.1.4, Table H-4, I.3.1.1, I.3.1.2.1, J.1.5.2.1, K.2.1.6

Table A-1. Use of Appendix A radioactivity inventory data in EIS chapters and appendixes (page 2 of 2).

Item ^a	Appendix A	EIS section
Amount of chromium per SNF assembly	A.2.1.5.3	5.1.2
Commercial SNF comprises at least 92% of radioactivity in Proposed Action	A.1.1.4.2	5.2.2, 5.2.3.3
DOE SNF has a variety of cladding	A.2.2.5.3	5.2.2
Commercial SNF has higher radionuclide content than DOE SNF or HLW	Table A-2	6.1.2.1
Cs-137, actinide, and total curies contained in a rail shipping cask for commercial SNF, HLW, DOE SNF, and naval fuel	Derived from Tables A-8, A-27, and A-18	Table 6-2, Table J-17
Radiological inventory of GTCC and SPAR waste much less than commercial SNF or HLW	Derived from Tables A-8, A-27, A-18, A-54, and Section A.2.6.4	8.2.7, 8.2.8, 8.4.1.1, F.3
Average radionuclide inventory per package for SPAR and GTCC waste	Derived from Table A-54 and Section A.2.6.4	8.3.1.1, Table I-9
C-14 (gas) is contained in fuel gap of commercial SNF	Tables A-8 and A-9	5.5, 8.3.1.1, I.3.3, I.7
Typical PWR burnup, initial enrichment, and average cooling time	A.2.1.5	G.2.3.2, H.2.1.4, J.1.4.2.5
Typical BWR burnup, initial enrichment, and average cooling time	A.2.1.5	G.2.3.2, H.2.1.4
N-reactor radionuclide inventory per canister is larger than HLW radionuclide per canister.	Tables A-18 and A-27	H.2.1.1
21 PWR assemblies contain a higher radionuclide content than 44 BWR assemblies	Tables A-8 and A-9	H.2.1.1
DOE would emplace twice as many PWR assemblies as BWR	A.2.1.5.1	H.2.1.1
N-reactor fuel represents a large quantity of DOE SNF	Table A-17	H.2.1.1
Mass of N-reactor fuel per canister	Table A-17	H.2.1.1
Immobilized Pu disk dimensions	A.2.4.5.2.1	I.3
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DOE SNF radionuclide inventory	Table A-18	I.3.1.1, I.3.1.2.1
Assumed packaging method for GTCC and SPAR	A.2.5.4, A.2.6.4	I.3.1.2.4
Chemical makeup of waste inventory	Tables A-12, -13, -19, -29, -30, -31, -32, -33, and -34	Table I-10
MTU per assembly for PWR and BWR	Table A-15	J.1.4.1.1
Most HLW stored in underground vaults	A.2.3.3	K.2.1.5.2

a. Abbreviations: SNF = spent nuclear fuel; MOX = mixed oxide; ISFSI = independent spent fuel storage installation; HLW = high-level radioactive waste; Pu = plutonium; GTCC = Greater-Than-Class-C; SPAR = Special-Performance-Assessment-Required; MTHM = metric tons of heavy metal; Kr = krypton; Cs = cesium; PWR = pressurized-water reactor; BWR = boiling-water reactor; MTU = metric tons of uranium.

the existing commercial reactor sites. Although the location of the plutonium immobilization facility has not been decided, DOE (1998a, page 1-9) has identified the Savannah River Site as the preferred alternative. For purposes of analysis, this EIS assumes that the high-level radioactive waste canisters, which would contain immobilized plutonium and borosilicate glass, would come from the Savannah River Site.



A-5

Figure A-1. Locations of commercial and DOE sites and Yucca Mountain.

Greater-Than-Class-C waste is waste with concentrations of certain radionuclides that exceed the Class C limits stated in 10 CFR Part 61, thereby making it unsuitable for near-surface disposal. Greater-Than-Class-C waste is generated by a number of sources including commercial nuclear utilities, sealed radioactive sources, and wastes from "other generators." These other generators include carbon-14 users, industrial research and development applications, fuel fabricators, university reactors, and others. These wastes are currently stored at the commercial and DOE sites and exist in most states. They are included in Inventory Module 2 of the EIS but are not part of the Proposed Action.

Special-Performance-Assessment-Required wastes are also Greater-Than-Class-C wastes managed by DOE and are stored primarily at the Hanford Site, Idaho National Engineering and Environmental Laboratory, West Valley Demonstration Project, and Oak Ridge National Laboratory in Tennessee. These wastes are included in Inventory Module 2 of the EIS but are not part of the Proposed Action.

A.1.1.2 Present Storage and Generation Status

Commercial spent nuclear fuel is stored at reactor sites in either a spent fuel pool or in a dry storage configuration generally referred to as an independent spent fuel storage installation. Through 1995, approximately 32,000 MTHM of commercial spent nuclear fuel has been discharged from reactors (Heath 1998, Appendix C). DOE spent nuclear fuel is also stored either underwater in basins or in a dry storage configuration similar to that used for commercial spent nuclear fuel.

As discussed in the next section, DOE would receive high-level radioactive waste at the repository in a solidified form in stainless-steel canisters. Until shipment to the repository, the canisters would be stored at the commercial and DOE sites. With the exception of the West Valley Demonstration Project, the filled canisters would be stored in below-grade facilities. The West Valley canisters would be stored in an above-ground shielded facility.

A.1.1.3 Final Waste Form

Other than drying or potential repackaging, processing is not necessary for commercial spent nuclear fuel. Therefore, the final form would be spent nuclear fuel either as bare intact assemblies or in sealed canisters. Bare intact fuel assemblies are those that do not have any disruption of their cladding and could be shipped to the repository in an approved shipping container for repackaging in a waste package in the Waste Handling Building. Other assemblies would be shipped to the repository in canisters that were either intended or not intended for disposal. Canisters not intended for disposal would be opened and repackaged in waste packages in the Waste Handling Building.

For most of the DOE spent nuclear fuel categories, the fuel would be shipped in disposable canisters (canisters that can be shipped and are suitable for direct insertion into waste packages without being opened) in casks licensed by the Nuclear Regulatory Commission. Uranium oxide fuels with intact zirconium alloy cladding are similar to commercial spent nuclear fuel and could be shipped either in DOE standard canisters or as bare intact assemblies. Uranium metal fuels from Hanford and aluminum-based fuels from the Savannah River Site could require additional treatment or conditioning before shipment to the repository. If treatment was required, these fuels would be packaged in DOE disposable canisters. Category 14 sodium-bonded fuels are also expected to require treatment before disposal.

High-level radioactive waste shipped to the repository would be in stainless-steel canisters. The waste would have undergone a solidification process that yielded a leach-resistant material, typically a glass form called borosilicate glass. In this process, the high-level radioactive waste is mixed with glass-forming materials, heated and converted to a durable glass waste form, and poured into stainless-steel canisters (Picha 1997, Attachment 4, page 2). Depending on future decisions stemming from other EISs, ceramic and metal waste matrices could be sent to the repository from Argonne National Laboratory-West

in Idaho. The ceramic and metal matrices would be different solidified mixtures that also would be in stainless-steel canisters. These wastes would be the result of the proposed electrometallurgical treatment of sodium bonded fuels.

As briefly described in Section A.1.1.1, the surplus weapon-usable plutonium would probably be sent to the repository in two different waste forms—spent mixed-oxide fuel assemblies or an immobilized plutonium ceramic form in a high-level radioactive waste canister and surrounded by high-level radioactive waste. The spent mixed-oxide fuel assemblies would be very similar to conventional low-enriched uranium assemblies and DOE would treat them as such. The immobilized plutonium would be placed in small cans, inserted in the high-level radioactive waste canisters, and covered with molten borosilicate glass (can-in-canister technique). The canisters containing immobilized plutonium and high-level radioactive waste would be externally identical to the normal high-level radioactive waste canisters.

A.1.1.4 Waste Characteristics

A.1.1.4.1 Mass and Volume

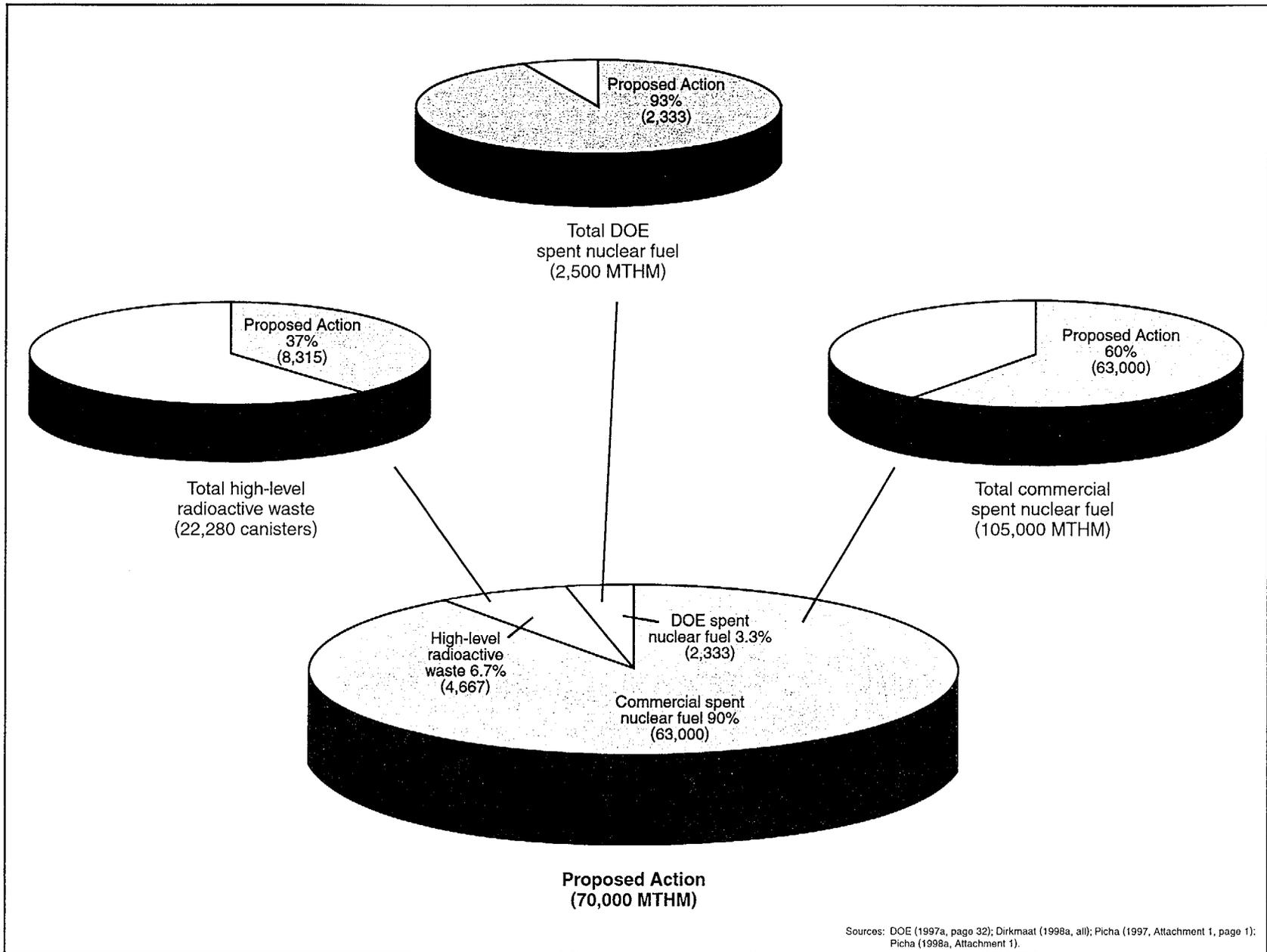
As discussed in Section A.1, the Proposed Action includes 70,000 MTHM in the forms of commercial spent nuclear fuel, DOE spent nuclear fuel, high-level radioactive waste, and surplus weapons-usable plutonium. Figure A-2 shows percentages of MTHM included in the Proposed Action and the relative amounts of the totals of the individual waste types included in the Proposed Action. As stated above, the remaining portion of the wastes is included in Inventory Module 1. Because Greater-Than-Class-C and Special-Performance-Assessment-Required wastes are measured in terms of volume, Figure A-3 shows the relative volume of the wastes in Inventory Module 2 compared to the inventory in Module 1.

The No-Action Alternative (see Chapter 7 and Appendix K) used this information to estimate the mass and volume of the spent nuclear fuel and high-level radioactive waste at commercial and DOE sites in five regions of the contiguous United States.

The mass and volume data for commercial spent nuclear fuel is the result of several years of annual tracking and projections by DOE, which anticipates few changes in the overall mass and volume projections for this waste type. The data projections for DOE spent nuclear fuel are fairly stable because most of the projected inventory already exists, as opposed to having a large amount projected for future generation. Mass and volume data for high-level radioactive waste estimates are not as reliable. Most high-level radioactive waste currently exists as a form other than solidified borosilicate glass. The solidification processes at the Savannah River Site and West Valley Demonstration Project are under way; therefore, the resulting mass and volume are known. However, the processes at the Idaho National Engineering and Environmental Laboratory and the Hanford Site have not started. Therefore, there is some uncertainty about the mass and volume that would result from those processing operations. For this analysis, DOE assumed that the high-level radioactive waste from the Hanford Site and the Idaho National Engineering and Environmental Laboratory would represent 65 and 6 percent of the total high-level radioactive waste inventory, respectively, in terms of the number of canisters.

A.1.1.4.2 Amount and Nature of Radioactivity

The primary purpose of presenting these data is to quantify the isotopic inventory expected in the projected waste types. These data were used for accident scenario analyses associated with transportation, handling, and repository operations. The data were also used to develop the source term associated with accident scenarios and long-term effects for the Proposed Action and the No-Action Alternative.



Sources: DOE (1997a, page 32); Dirkmaat (1998a, all); Picha (1997, Attachment 1, page 1); Picha (1998a, Attachment 1).

Figure A-2. Proposed Action spent nuclear fuel and high-level radioactive waste inventory.

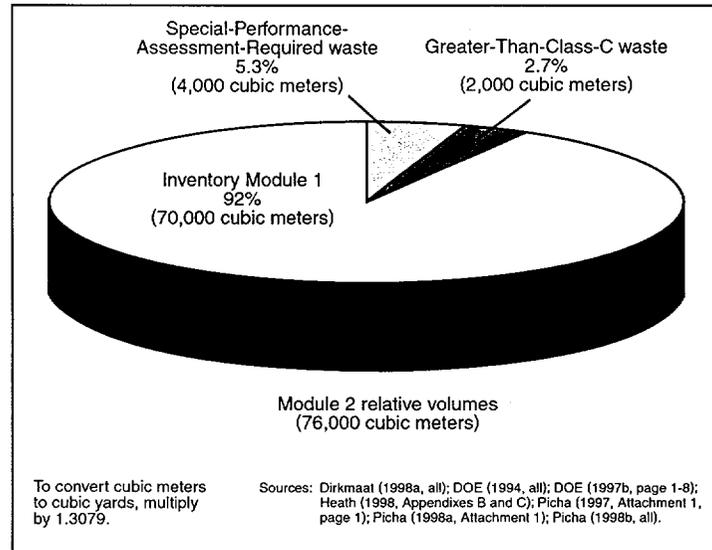


Figure A-3. Inventory Module 2 volume.

In a comparison of the relative amounts of radioactivity in a particular waste type, radionuclides of concern depend on the analysis being performed. For example, cesium-137 is the primary radionuclide of concern when reviewing preclosure impacts and shielding requirements. For postclosure impacts, the repository performance assessment evaluated nine radionuclides (see Appendix I) and identified technetium-99 and neptunium-237 as the nuclides that provide the greatest impacts. Plutonium-238 and -239 are shown in Chapter 7 to contribute the most to doses for the No-Action Alternative. Table A-2 presents the inventory of each of these radionuclides included in the Proposed Action. Figure A-4 shows that at least 92 percent of the total inventory of each of these radionuclides is in commercial spent nuclear fuel.

Table A-2. Selected nuclide inventory for the Proposed Action (curies).

	Commercial spent nuclear fuel	DOE spent nuclear fuel	High-level radioactive waste	Surplus plutonium	Totals
Cesium-137	4.0×10^9	1.7×10^8	1.7×10^8	NA ^a	4.3×10^9
Technetium-99	9.2×10^5	2.9×10^4	2.1×10^4	NA	9.7×10^5
Neptunium-237	2.8×10^4	1.1×10^3	4.5×10^2	NA	3.0×10^4
Plutonium-238	2.1×10^8	5.6×10^6	3.0×10^6	7.6×10^4	2.2×10^8
Plutonium -239	2.3×10^7	3.8×10^5	4.4×10^4	1.0×10^6	2.5×10^7

a. NA = not applicable.

A.1.1.4.3 Chemical Composition

The appendix presents data for the chemical composition of the primary waste types. For commercial spent nuclear fuel, the elemental composition of typical pressurized-water and boiling-water reactor fuel is provided on a per-assembly basis. Data are also provided on the number of stainless-steel clad assemblies in the current inventory.

For DOE spent nuclear fuel and high-level radioactive waste, this appendix contains tables that describe the composition of the total inventory of the spent nuclear fuel (by representative category) or high-level radioactive waste (by site).

The chemical composition data were used primarily in the repository performance assessment (see Chapter 5 and Appendix I) to evaluate the relative amounts of materials that would need further study.

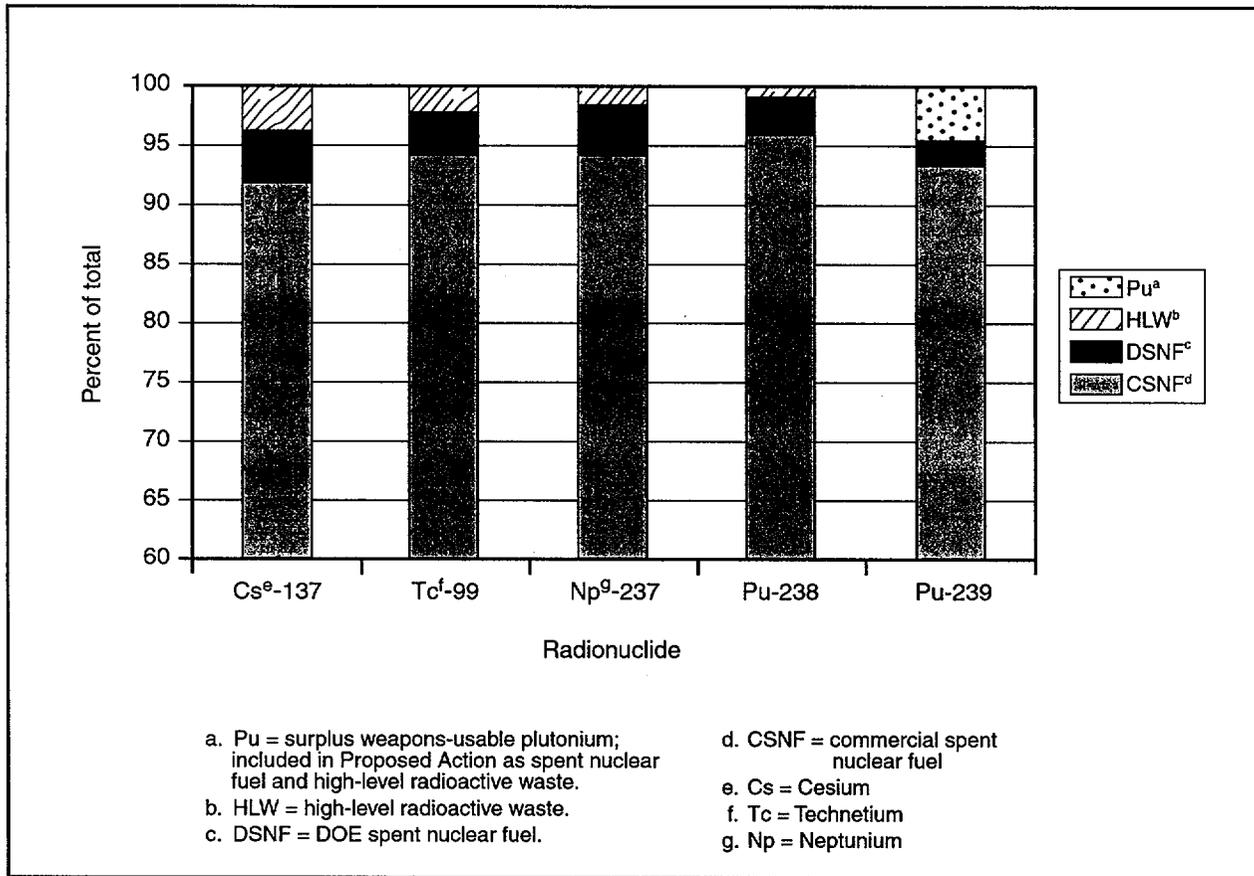


Figure A-4. Proposed Action radionuclide distribution by material type.

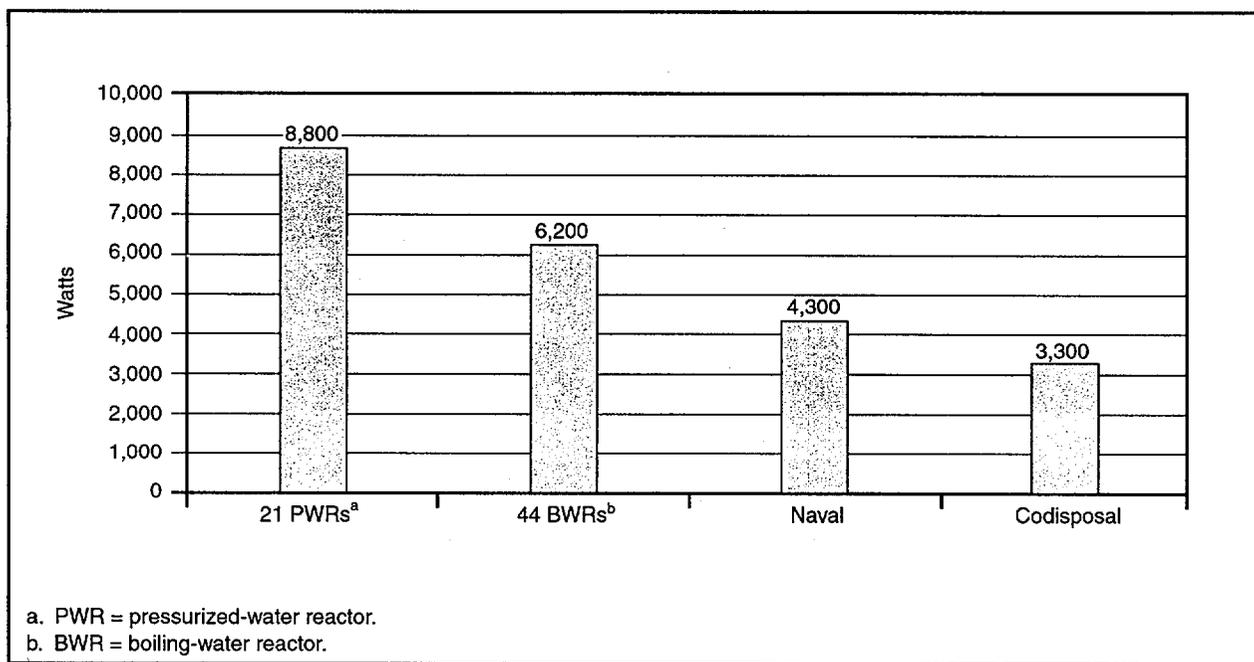


Figure A-5. Thermal generation (watts per waste package).

As a result of an initial screening, the repository performance assessment evaluated the long-term impacts of molybdenum, uranium, and chromium in the repository.

A.1.1.4.4 Thermal Output

Thermal generation data associated with each material type are provided in this appendix. These data were used to develop the thermal loads associated with the repository design. Chapter 2 describes the thermal load scenarios. The thermal data demonstrate that the EIS analysis can make simplifying assumptions that the thermal output of the commercial spent nuclear fuel waste packages, particularly the pressurized-water reactor assemblies, would bound the thermal output of all other waste packages (see Figure A-5).

The data presented in the thermal output sections of this appendix for each waste type are presented as watts per assembly or MTHM for commercial spent nuclear fuel, and watts per canister for DOE spent nuclear fuel or high-level radioactive waste. Figure A-5 normalizes these data into a common, watts-per-waste-package comparison. The following waste packages are compared: one containing 21 typical pressurized-water reactor assemblies, one containing 44 typical boiling-water reactor assemblies, a co-disposal waste package containing five high-level radioactive waste canisters and one DOE spent nuclear fuel canister, and a waste package containing one dual-purpose canister of naval spent nuclear fuel (also a DOE spent fuel). Another potential waste package containing four multi-canister overpacks of DOE uranium metal fuels is not included in Figure A-5 because its estimated maximum thermal generation is only 72 watts per waste package.

Figure A-5 uses conservative assumptions to illustrate the bounding nature of the thermal data for commercial spent nuclear fuel. The commercial spent nuclear fuel data represent typical assemblies that are assumed to have cooled for nearly 30 years. The naval spent nuclear fuel data are a best estimate of the thermal generation of 5-year old spent nuclear fuel. The thermal data selected for the high-level radioactive waste are conservatively represented by the canisters from the Savannah River Site and are combined with the highest values of thermal output from all projected DOE spent nuclear fuel categories.

A.1.1.4.5 Canister Data

Typically, DOE spent nuclear fuel and high-level radioactive waste would be sent to the repository in disposable canisters. The design specifications for DOE spent nuclear fuel canisters are in DOE (1998c, all). These canisters are generally of two diameters—46 and 61 centimeters (18 and 24 inches). They also would be designed for two different lengths, nominally 3 and 4.6 meters (10 and 15 feet), to enable co-disposal with high-level radioactive waste canisters. Certain DOE spent nuclear fuel categories require specific disposal canister designs. Naval fuels would be sent to the repository in Navy dual-purpose canisters, which are described in Dirkmaat (1997a, Attachment, pages 86 to 88) and USN (1996, pages 3-1 to 3-11). N-Reactor fuels from the Hanford Site would be sent to the repository in multicanister overpacks 64 centimeters (25.3 inches) in diameter, which are described in Parsons (1999, all).

High-level radioactive waste would be sent to the repository in stainless-steel canisters, 61 centimeters (25 inches) in diameter and either 3 or 4.6 meters (10 or 15 feet) in length, depending on the DOE site. The canister design specifications are contained in Marra, Harbour, and Plodinec (1995, all) and WVNS (1996, WQR-2.2, all) for the operating vitrification processes at Savannah River Site and West Valley Demonstration Project, respectively. The other sites would use canister designs similar to those currently in use (Picha 1997, all).

These data were for analysis of the No-Action Alternative (see Chapter 7 and Appendix K) to determine the time required to breach the canisters after they are exposed to weather elements.

A.2 Materials

This section describes the characteristics of the materials DOE has considered for disposal in the proposed Yucca Mountain Repository. All candidate materials would have to meet approved acceptance criteria.

A.2.1 COMMERCIAL SPENT NUCLEAR FUEL

A.2.1.1 Background

Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation. Spent nuclear fuel from light-water reactors (pressurized-water and boiling-water reactors) would be the primary source of radioactivity and thermal load in the proposed monitored geologic repository. Spent nuclear fuels from civilian research reactors (General Atomics, Aerotest, etc.) account for less than 0.001 percent of the projected total in the Proposed Action (DOE 1995a, all). The fuels addressed in this section are those discharged from commercial light-water reactors.

Section A.2.2 discusses the spent nuclear fuel from the Fort St. Vrain reactor in Colorado as part of DOE spent nuclear fuels, as are the fuels from Shippingport, Three Mile Island-2, and other fuels from commercial facilities that DOE is managing at its facilities.

A.2.1.2 Sources

The sources of commercial spent nuclear fuel are the commercial nuclear powerplants throughout the country. Table A-3 lists the individual reactors, reactor type, state, and actual or projected years of operation. The operation period is subject to change if a utility pursues extension of the operating license or shuts down early.

A.2.1.3 Present Status

Nuclear power reactors store spent nuclear fuel in spent fuel pools under U.S. Nuclear Regulatory Commission licenses, but they can use a combination of storage options: (1) in-pool storage and (2) above-grade dry storage in an independent spent fuel storage installation. When a reactor is refueled, spent fuel is transferred to the spent fuel pool, where it typically remains until the available pool capacity is reached. When in-pool storage capacity has been fully used, utilities have turned to dry cask storage in an independent spent fuel storage installation to expand their onsite spent fuel storage capacities. In 1990, the Nuclear Regulatory Commission amended its regulations to authorize licensees to store spent nuclear fuel at reactor sites in approved storage casks (Raddatz and Waters 1996, all).

Commercial nuclear utilities currently use three Nuclear Regulatory Commission-approved general dry storage system design types—metal storage casks and metal canisters housed in concrete casks and concrete vaults—for use in licensed independent spent fuel storage installations. Raddatz and Waters (1996, all) contains detailed information on models currently approved by the Commission. Table A-4 lists existing and planned independent spent fuel storage installations in the United States.

A.2.1.4 Final Spent Nuclear Fuel Form

The final form of commercial spent nuclear fuel to be disposed of in the proposed repository would be the current reactor fuel assemblies. The repository would receive bare spent nuclear fuel assemblies, spent nuclear fuel packaged in canisters not intended for disposal, and spent nuclear fuel packaged in canisters intended for disposal.

Table A-3. Commercial nuclear power reactors in the United States and their projected years of operation.^a

Unit name	Reactor type ^b	State	Operations period ^c	Unit name	Reactor type ^b	State	Operations period ^c
Arkansas Nuclear One 1	PWR	AR	1974-2014	Millstone 3	PWR	CT	1986-2025
Arkansas Nuclear One 2	PWR	AR	1978-2018	Monticello	BWR	MN	1971-2010
Beaver Valley 1	PWR	PA	1976-2016	Nine Mile Point 1	BWR	NY	1969-2009
Beaver Valley 2	PWR	PA	1978-2018	Nine Mile Point 2	BWR	NY	1987-2026
Big Rock Point	BWR	MI	1963-1997	North Anna 1	PWR	VA	1978-2018
Braidwood 1	PWR	IL	1987-2026	North Anna 2	PWR	VA	1980-2020
Braidwood 2	PWR	IL	1988-2027	Oconee 1	PWR	SC	1973-2013
Browns Ferry 1	BWR	AL	1973-2013	Oconee 2	PWR	SC	1973-2013
Browns Ferry 2	BWR	AL	1974-2014	Oconee 3	PWR	SC	1974-2014
Browns Ferry 3	BWR	AL	1976-2016	Oyster Creek	BWR	NJ	1969-2009
Brunswick 1	BWR	NC	1976-2016	Palisades	PWR	MI	1972-2007
Brunswick 2	BWR	NC	1974-2014	Palo Verde 1	PWR	AZ	1985-2024
Byron 1	PWR	IL	1985-2024	Palo Verde 2	PWR	AZ	1986-2025
Byron 2	PWR	IL	1987-2026	Palo Verde 3	PWR	AZ	1987-2027
Callaway	PWR	MO	1984-2024	Peach Bottom 2	BWR	PA	1973-2013
Calvert Cliffs 1	PWR	MD	1974-2014	Peach Bottom 3	BWR	PA	1974-2014
Calvert Cliffs 2	PWR	MD	1976-2016	Perry 1	BWR	OH	1986-2026
Catawba 1	PWR	SC	1985-2024	Pilgrim 1	BWR	MA	1972-2012
Catawba 2	PWR	SC	1986-2026	Point Beach 1	PWR	WI	1970-2010
Clinton	BWR	IL	1987-2026	Point Beach 2	PWR	WI	1973-2013
Comanche Peak 1	PWR	TX	1990-2030	Prairie Island 1	PWR	MN	1974-2013
Comanche Peak 2	PWR	TX	1993-2033	Prairie Island 2	PWR	MN	1974-2014
Cooper Station	BWR	NE	1974-2014	Quad Cities 1	BWR	IL	1972-2012
Crystal River 3	PWR	FL	1977-2016	Quad Cities 2	BWR	IL	1972-2012
D. C. Cook 1	PWR	MI	1974-2014	Rancho Seco	PWR	CA	1974-1989
D. C. Cook 2	PWR	MI	1977-2017	River Bend 1	BWR	LA	1985-2025
Davis-Besse	PWR	OH	1977-2017	Salem 1	PWR	NJ	1976-2016
Diablo Canyon 1	PWR	CA	1984-2021	Salem 2	PWR	NJ	1981-2020
Diablo Canyon 2	PWR	CA	1985-2025	San Onofre 1	PWR	CA	1967-1992
Dresden 1	BWR	IL	1959-1978	San Onofre 2	PWR	CA	1982-2013
Dresden 2	BWR	IL	1969-2006	San Onofre 3	PWR	CA	1983-2013
Dresden 3	BWR	IL	1971-2011	Seabrook 1	PWR	NH	1990-2026
Duane Arnold 1	BWR	IA	1974-2014	Sequoyah 1	PWR	TN	1980-2020
Edwin I. Hatch 1	BWR	GA	1974-2014	Sequoyah 2	PWR	TN	1981-2021
Edwin I. Hatch 2	BWR	GA	1978-2018	Shoreham Harris	BWR	NC	1987-2026
Fermi 2	BWR	MI	1985-2025	Shoreham	BWR	NY	1989 ^d
Fort Calhoun 1	PWR	NE	1973-2013	South Texas Project 1	PWR	TX	1988-2016
Ginna	PWR	NY	1969-2009	South Texas Project 2	PWR	TX	1989-2023
Grand Gulf 1	BWR	MS	1984-2022	St. Lucie 1	PWR	FL	1976-2016
Haddam Neck	PWR	CT	1968-1996	St. Lucie 2	PWR	FL	1983-2023
Hope Creek	BWR	NJ	1986-2026	Summer 1	PWR	SC	1982-2022
Humboldt Bay	BWR	CA	1962-1976	Surry 1	PWR	VA	1972-2012
H.B. Robinson 2	PWR	SC	1970-2010	Surry 2	PWR	VA	1973-2013
Indian Point 1	PWR	NY	1962-1974	Susquehanna 1	BWR	PA	1982-2022
Indian Point 2	PWR	NY	1973-2013	Susquehanna 2	BWR	PA	1984-2024
Indian Point 3	PWR	NY	1976-2015	Three Mile Island 1	PWR	PA	1974-2014
James A. FitzPatrick/ Nine Mile Point	BWR	NY	1974-2014	Trojan	PWR	OR	1975-1992
Joseph M. Farley 1	PWR	AL	1977-2017	Turkey Point 3	PWR	FL	1972-2012
Joseph M. Farley 2	PWR	AL	1981-2021	Turkey Point 4	PWR	FL	1973-2013
Kewaunee	PWR	WI	1973-2013	Vermont Yankee	BWR	VT	1973-2012
LaCrosse	BWR	WI	1967-1987	Vogtle 1	PWR	GA	1987-2027
LaSalle 1	BWR	IL	1970-2022	Vogtle 2	PWR	GA	1989-2029
LaSalle 2	BWR	IL	1970-2023	Washington Public Power Supply System 2	BWR	WA	1984-2023
Limerick 1	BWR	PA	1985-2024	Waterford 3	PWR	LA	1985-2024
Limerick 2	BWR	PA	1989-2029	Watts Bar 1	PWR	TN	1996-2035
Maine Yankee	PWR	ME	1972-1996	Wolf Creek	PWR	KS	1985-2025
McGuire 1	PWR	NC	1981-2021	Yankee-Rowe	PWR	MA	1963-1991
McGuire 2	PWR	NC	1983-2023	Zion 1	PWR	IL	1973-1997
Millstone 1	BWR	CT	1970-2010	Zion 2	PWR	IL	1974-1996
Millstone 2	PWR	CT	1975-2015				

a. Source: DOE (1997a, Appendix C).

b. PWR = pressurized-water reactor; BWR = boiling-water reactor.

c. As defined by current shutdown or full operation through license period (as of 1997).

d. Shoreham is no longer a licensed plant and has transferred all fuel to Limerick.

Table A-4. Sites with existing or planned independent spent fuel storage installations.^a

Reactor	Status	Reactor	Status
Prairie Island	Existing	Rancho Seco	Planned
Point Beach	Existing	Trojan	Planned
Palisades	Existing	Washington Public Power Supply System	Planned
Surry	Existing	Big Rock Point	Planned
Calvert Cliffs	Existing	Oyster Creek	Planned
Arkansas Nuclear	Existing	Duane Arnold	Planned
H. B. Robinson	Existing	McGuire	Planned
Oconee	Existing	Yankee Rowe	Planned
Davis-Besse	Existing	Maine Yankee	Planned
North Anna	Planned	Peach Bottom	Planned
James A. FitzPatrick/Nine Mile Point	Planned	Palo Verde	Planned
Dresden	Planned	Humboldt Bay	Planned
Susquehanna	Planned		

a. Sources: Raddatz and Waters (1996, all); Cole (1998a, all).

A.2.1.5 Spent Nuclear Fuel Characteristics

There are 22 classes of nuclear fuel assemblies, with 127 individual fuel types in those classes. Seventeen of the classes are for pressurized-water reactor fuels and 5 are for boiling-water reactors (DOE 1992, Appendix 2A). For this EIS, the typical assemblies chosen for analysis represent an assembly type being used in the more recently built reactors. This results in physical characteristics that might be slightly higher than average (size, uranium per assembly, etc.), but that, however, provide a realistic estimate for EIS analyses. Specifically chosen to represent the typical fuel types were the Westinghouse 17 x 17 LOPAR fuel assembly for the pressurized-water reactor and the General Electric BWR/4-6, 8 x 8 fuel assembly for the boiling-water reactor. Table A-5 lists the fissile content and performance parameters selected to define the radiological characteristics of these typical fuel assemblies.

Table A-5. Typical spent nuclear fuel parameters.^a

Fuel type ^b	Burnup (MWd/MTHM) ^c	Initial enrichment (percent of U-235 by weight)	Age (years)
Typical PWR	39,560	3.69	25.9
Typical BWR	32,240	3.00	27.2

a. Source: TRW (1998, page 3-15).

b. PWR = pressurized-water reactor; BWR = boiling-water reactor.

c. MWd/MTHM = megawatt-days per metric ton of heavy metal; to convert metric tons to tons, multiply by 1.1023.

A.2.1.5.1 Mass and Volume

As discussed in Section A.1, the Proposed Action includes 63,000 MTHM of commercial spent nuclear fuel. For the No-Action Alternative (continued storage) analysis, Table A-6 lists the distribution of this expected inventory by reactor site. The historic and projected spent nuclear fuel discharge and storage information in Table A-6 is consistent with the annual projections provided by the Energy Information Administration (DOE 1997a, page 32). The "1995 Actual" data presented in Table A-6 represents the amount of spent nuclear fuel stored at a particular site regardless of the reactor from which it was discharged. For analysis purposes, the table lists spent nuclear fuel currently stored at the General Electric Morris, Illinois, facility to be at Dresden, because these facilities are located near each other.

For analyses associated with the Proposed Action, the projected spent nuclear fuel from pressurized-water reactors comprises 65 percent of the 63,000 metric tons of heavy metal (TRW 1997, page A-2). The

Table A-6. Proposed Action spent nuclear fuel inventory (MTHM).^a

Site	Fuel type ^b	1995 actual	1996-2011 ^c	Total ^d	Equivalent assemblies	Site	Fuel type ^b	1995 actual	1996-2011 ^c	Total ^d	Equivalent assemblies
Arkansas Nuclear One	PWR	643	466	1,109	2,526	Monticello	BWR	147	280	426	2,324
Beaver Valley	PWR	437	581	1,018	2,206	North Anna	PWR		613	1,184	2,571
								570			
Big Rock Point	BWR	44	14	58	439	Oconee	PWR	1,098	767	1,865	4,028
Braidwood	PWR	318	711	1,029	2,424	Oyster Creek	BWR	374	325	699	3,824
Browns Ferry	BWR	840	1,092	1,932	10,402	Palisades	PWR	338	247	585	1,473
Brunswick	Both	448	448	896	4,410	Palo Verde	PWR	556	1,118	1,674	4,082
Byron	PWR	404	664	1,068	2,515	Peach Bottom	BWR	908	645	1,554	8,413
Callaway	PWR	280	422	702	1,609	Perry	BWR	178	274	452	2,470
Calvert Cliffs	PWR	641	501	1,142	2,982	Pilgrim	BWR	326	201	527	2,853
Catawba	PWR	465	683	1,148	2,677	Point Beach	PWR	529	347	876	2,270
Clinton	BWR	174	303	477	2,588	Prairie Island	PWR	518	348	866	2,315
Comanche Peak	PWR	176	821	998	2,202	Quad Cities	BWR	813	464	1,277	6,953
Cooper	BWR	175	277	452	2,435	Rancho Seco	PWR	228	-- ^e	228	493
Crystal River	PWR	280	232	512	1,102	River Bend	BWR	176	356	531	2,889
D. C. Cook	PWR	777	656	1,433	3,253	Salem/Hope Creek	Both	793	866	1,659	7,154
Davis-Besse	PWR	243	262	505	1,076	San Onofre	PWR	722	701	1,423	3,582
Diablo Canyon	PWR	463	664	1,126	2,512	Seabrook	PWR	133	292	425	918
Dresden	BWR	1,557	590	2,146	11,602	Sequoyah	PWR	452	570	1,023	2,218
Duane Arnold	BWR	258	208	467	2,545	Shearon Harris	Both	498	252	750	2,499
Edwin I. Hatch	BWR	755	692	1,446	7,862	South Texas Project	PWR	290	722	1,012	1,871
Fermi	BWR	155	368	523	2,898	St. Lucie	PWR	601	419	1,020	2,701
Fort Calhoun	PWR	222	157	379	1,054	Summer	PWR	225	301	526	1,177
Ginna	PWR	282	180	463	1,234	Surry	PWR	660	534	1,194	2,604
Grand Gulf	BWR	349	506	856	4,771	Susquehanna	BWR	628	648	1,276	7,172
H. B. Robinson	PWR	145	239	384	903	Three Mile Island	PWR	311	236	548	1,180
Haddam Neck	PWR	355	65	420	1,017	Trojan	PWR	359	--	359	780
Humboldt Bay	BWR	29	--	29	390	Turkey Point	PWR	616	458	1,074	2,355
Indian Point	PWR	678	486	1,164	2,649	Vermont Yankee	BWR	387	222	609	3,299
James A. FitzPatrick/ Nine Mile Point	BWR	882	930	1,812	9,830	Vogtle	PWR	335	745	1,080	2,364
Joseph M. Farley	PWR	644	530	1,174	2,555	Washington Public Power Supply System 2	BWR	243	338	581	3,223
Kewaunee	PWR	282	169	451	1,172	Waterford	PWR	253	247	500	1,217
La Crosse	BWR	38	--	38	333	Watts Bar	PWR	--	251	251	544
La Salle	BWR	465	487	952	5,189	Wolf Creek	PWR	226	404	630	1,360
Limerick	BWR	432	711	1,143	6,203	Yankee-Rowe	PWR	127	--	127	533
Maine Yankee	PWR	454	82	536	1,421	Zion	PWR	841	211	1,052	2,302
McGuire	PWR	714	725	1,439	3,257						
Millstone	Both	959	749	1,709	6,447	Totals		31,926	31,074	63,000	218,700

- a. Source: Heath (1998, Appendixes B and C).
- b. PWR = pressurized-water reactor; BWR = boiling-water reactor.
- c. Projected.
- d. To convert metric tons to tons, multiply by 1.1023.
- e. -- = no spent nuclear fuel production.

balance consists of spent nuclear fuel from boiling-water reactors. Using the nominal volume for the spent nuclear fuel assemblies described in Section A.2.1.5.5, the estimated volume of spent nuclear fuel in the Proposed Action, exclusive of packaging, is 29,000 cubic meters.

Section A.1 also discusses the additional inventory modules evaluated in this EIS. Inventory Modules 1 and 2 both include the maximum expected discharge inventory of commercial spent nuclear fuel. Table A-7 lists historic and projected amounts of spent nuclear fuel discharged from commercial reactors through 2046. The estimated unpackaged volume of spent nuclear fuel for these modules is approximately 47,000 cubic meters. For conservatism, these data were derived from the Energy Information Administration "high case" assumptions. The high case assumes that all currently operating nuclear units would renew their operating licenses for an additional 10 years (DOE 1997a, page 32).

Table A-7. Inventory Modules 1 and 2 spent nuclear fuel inventory (MTHM).^a

Site	Fuel type ^b	1995 actual	1996-2046 ^c	Total ^d	Equivalent assemblies	Site	Fuel type ^b	1995 actual	1996-2046 ^c	Total ^d	Equivalent assemblies
Arkansas Nuclear One	PWR	643	1,007	1,650	3,757	Monticello	BWR	147	390	537	2,924
Beaver Valley	PWR	437	1,395	1,832	3,970	North Anna	PWR	570	1,384	1,955	4,246
Big Rock Point	BWR	44	14	58	439	Oconee	PWR	1,098	1,576	2,674	5,774
Braidwood	PWR	318	1,969	2,287	5,385	Oyster Creek	BWR	374	470	844	4,619
Browns Ferry	BWR	840	2,508	3,348	18,024	Palisades	PWR	338	395	733	1,845
Brunswick	Both	448	992	1,440	7,355	Palo Verde	PWR	556	3,017	3,573	8,712
Byron	PWR	404	1,777	2,181	5,139	Peach Bottom	BWR	908	1,404	2,312	12,523
Callaway	PWR	280	1,008	1,288	2,953	Perry	BWR	178	732	910	4,974
Calvert Cliffs	PWR	641	1,069	1,710	4,466	Pilgrim	BWR	326	444	770	4,170
Catawba	PWR	465	1,752	2,217	5,168	Point Beach	PWR	529	614	1,143	2,961
Clinton	BWR	174	910	1,084	5,876	Prairie Island	PWR	518	692	1,210	3,234
Comanche Peak	PWR	176	2,459	2,635	5,816	Quad Cities	BWR	813	1,020	1,834	9,982
Cook	PWR	777	1,379	2,155	4,892	Rancho Seco	PWR	228	-- ^e	228	493
Cooper	BWR	175	587	762	4,106	River Bend	BWR	176	956	1,132	6,153
Crystal River	PWR	280	525	805	1,734	Salem/Hope Creek	Both	793	2,452	3,245	11,584
Davis-Besse	PWR	243	582	825	1,757	San Onofre	PWR	722	1,321	2,043	5,144
Diablo Canyon	PWR	463	1,725	2,187	4,878	Seabrook	PWR	133	831	964	2,083
Dresden	BWR	1,557	984	2,541	13,740	Sequoyah	PWR	452	1,393	1,845	4,001
Duane Arnold	BWR	258	434	692	3,776	Shearon Harris	Both	498	707	1,205	3,535
Fermi	BWR	155	1,005	1,160	6,429	South Texas Project	PWR	290	2,029	2,319	4,286
Fort Calhoun	PWR	222	312	534	1,485	St. Lucie	PWR	601	1,010	1,611	4,265
Ginna	PWR	282	283	565	1,507	Summer	PWR	225	732	958	2,141
Grand Gulf	BWR	349	1,261	1,610	8,976	Surry	PWR	660	1,029	1,689	3,682
H. B. Robinson	PWR	145	364	509	1,197	Susquehanna	BWR	628	1,745	2,373	13,338
Haddam Neck	PWR	355	65	420	1,017	Three Mile Island	PWR	311	513	825	1,777
Hatch	BWR	755	1,517	2,272	12,347	Trojan	PWR	359	--	359	780
Humboldt Bay	BWR	29	--	29	390	Turkey Point	PWR	616	905	1,520	3,334
Indian Point	PWR	678	1,005	1,683	3,787	Vermont Yankee	BWR	387	434	822	4,451
James A. FitzPatrick/ Nine Mile Point	BWR	882	2,018	2,900	15,732	Vogtle	PWR	335	2,122	2,458	5,378
Joseph M. Farley	PWR	644	1,225	1,869	4,070	Washington Public Power Supply System 2	BWR	243	924	1,167	6,476
Kewaunee	PWR	282	330	612	1,591	Waterford	PWR	253	685	938	2,282
La Crosse	BWR	38	--	38	333	Watts Bar	PWR	--	893	893	1,937
La Salle	BWR	465	1,398	1,863	10,152	Wolf Creek	PWR	226	1,052	1,278	2,759
Limerick	BWR	432	1,958	2,390	12,967	Yankee-Rowe	PWR	127	--	127	533
Maine Yankee	PWR	454	82	536	1,421	Zion	PWR	841	211	1,052	2,302
McGuire	PWR	714	1,813	2,527	5,720	Totals		31,926	73,488	105,414	359,963
Millstone	Both	959	1,695	2,655	8,930						

- a. Source: Heath (1998, Appendixes B and C).
- b. PWR = pressurized-water reactor; BWR = boiling-water reactor.
- c. Projected.
- d. To convert metric tons to tons, multiply by 1.1023.
- e. -- = no spent nuclear fuel production.

A.2.1.5.2 Amount and Nature of Radioactivity

DOE derived radionuclide inventories for the typical pressurized-water reactor and boiling-water reactor fuel assemblies from the Light-Water Reactor Radiological Database (DOE 1992, page 1.1-1). The inventories are presented at the average decay years for each of the typical assemblies. Tables A-8 and A-9 list the inventories of the nuclides of interest for the typical assemblies for both reactor types.

Table A-10 combines the typical inventories (curies per MTHM) with the projected totals (63,000 MTHM and 105,000 MTHM) to provide a total projected radionuclide inventory for the Proposed Action and additional modules.

A.2.1.5.3 Chemical Composition

Commercial spent nuclear fuel consists of the uranium oxide fuel itself (including actinides, fission products, etc.), the cladding, and the assembly hardware.

Table A-8. Radionuclide activity for typical pressurized-water reactor fuel assemblies.^{a,b}

Isotope	Curies per assembly	Isotope	Curies per assembly	Isotope	Curies per assembly
Hydrogen-3	9.8×10 ¹	Cesium-134	1.6×10 ¹	Neptunium-237	2.3×10 ⁻¹
Carbon-14	6.4×10 ⁻¹	Cesium-135	2.5×10 ⁻¹	Plutonium-238	1.7×10 ³
Chlorine-36	5.4×10 ⁻³	Cesium-137	3.1×10 ⁴	Plutonium-239	1.8×10 ²
Cobalt-60	1.5×10 ²	Samarium-151	1.9×10 ²	Plutonium-240	2.7×10 ²
Nickel-59	1.3	Lead-210	2.2×10 ⁻⁷	Plutonium-241	2.0×10 ⁴
Nickel-63	1.8×10 ²	Radium-226	9.3×10 ⁻⁷	Plutonium-242	9.9×10 ⁻¹
Selenium-79	2.3×10 ⁻¹	Radium-228	1.3×10 ⁻¹⁰	Americium-241	1.7×10 ³
Krypton-85	9.3×10 ²	Actinium-227	7.8×10 ⁻⁶	Americium-242/242m	1.1×10 ¹
Strontium-90	2.1×10 ⁴	Thorium-229	1.7×10 ⁻⁷	Americium-243	1.3×10 ¹
Zirconium-93	1.2	Thorium-230	1.5×10 ⁻⁴	Curium-242	8.7
Niobium-93m	8.2×10 ⁻¹	Thorium-232	1.9×10 ⁻¹⁰	Curium-243	8.3
Niobium-94	5.8×10 ⁻¹	Protactinium-231	1.6×10 ⁻⁵	Curium-244	7.0×10 ²
Technetium-99	7.1	Uranium-232	1.9×10 ⁻²	Curium-245	1.8×10 ⁻¹
Rhodium-102	1.2×10 ⁻³	Uranium-233	3.3×10 ⁻⁵	Curium-246	3.8×10 ⁻²
Ruthenium-106	4.8×10 ⁻³	Uranium-234	6.6×10 ⁻¹	Curium-247	1.3×10 ⁻⁷
Palladium-107	6.3×10 ⁻²	Uranium-235	8.4×10 ⁻³	Curium-248	3.9×10 ⁻⁷
Tin-126	4.4×10 ⁻¹	Uranium-236	1.4×10 ⁻¹	Californium-252	3.1×10 ⁻⁸
Iodine-129	1.8×10 ⁻²	Uranium-238	1.5×10 ⁻¹		

a. Source: DOE (1992, page 1.1-1).

b. Burnup = 39,560 MWd/MTHM, enrichment = 3.69 percent, decay time = 25.9 years.

Table A-9. Radionuclide activity for typical boiling-water reactor fuel assemblies.^{a,b}

Isotope	Curies per assembly	Isotope	Curies per assembly	Isotope	Curies per assembly
Hydrogen-3	3.4×10 ¹	Cesium-134	3.4	Neptunium-237	7.3×10 ⁻²
Carbon-14	3.0×10 ⁻¹	Cesium-135	1.0×10 ⁻¹	Plutonium-238	5.5×10 ²
Chlorine-36	2.2×10 ⁻³	Cesium-137	1.1×10 ⁴	Plutonium-239	6.3×10 ¹
Cobalt-60	3.7×10 ¹	Samarium-151	6.6×10 ¹	Plutonium-240	9.5×10 ¹
Nickel-59	3.5×10 ⁻¹	Lead-210	9.4×10 ⁻⁸	Plutonium-241	7.5×10 ³
Nickel-63	4.6×10 ¹	Radium-226	3.7×10 ⁻⁷	Plutonium-242	4.0×10 ⁻¹
Selenium-79	7.9×10 ⁻²	Radium-228	4.7×10 ⁻¹¹	Americium-241	6.8×10 ²
Krypton-85	2.9×10 ²	Actinium-227	3.1×10 ⁻⁶	Americium-242/242m	4.6
Strontium-90	7.1×10 ³	Thorium-229	6.1×10 ⁻⁸	Americium-243	4.9
Zirconium-93	4.8×10 ⁻¹	Thorium-230	5.8×10 ⁻⁵	Curium-242	3.8
Niobium-93m	3.5×10 ⁻¹	Thorium-232	6.9×10 ⁻¹¹	Curium-243	3.1
Niobium-94	1.9×10 ⁻²	Protactinium-231	6.0×10 ⁻⁶	Curium-244	2.5×10 ²
Technetium-99	2.5	Uranium-232	5.5×10 ⁻³	Curium-245	6.3×10 ⁻²
Rhodium-102	2.8×10 ⁻⁴	Uranium-233	1.1×10 ⁻⁵	Curium-246	1.3×10 ⁻²
Ruthenium-106	6.7×10 ⁻⁴	Uranium-234	2.4×10 ⁻¹	Curium-247	4.3×10 ⁻⁸
Palladium-107	2.4×10 ⁻²	Uranium-235	3.0×10 ⁻³	Curium-248	1.2×10 ⁻⁷
Tin-126	1.5×10 ⁻¹	Uranium-236	4.8×10 ⁻²	Californium-252	6.0×10 ⁻⁹
Iodine-129	6.3×10 ⁻³	Uranium-238	6.2×10 ⁻²		

a. Source: DOE (1992, page 1.1-1).

b. Burnup = 32,240 MWd/MTHM, enrichment = 3.00 percent, decay time = 27.2 years.

Typical pressurized-water and boiling-water reactor fuels consist of uranium dioxide with a zirconium alloy cladding. Some assemblies, however, are clad in stainless-steel 304. Specifically, 2,187 assemblies, or 727 MTHM (1.15 percent of the MTHM included in the Proposed Action) are stainless-steel clad (Cole 1998b, all). These assemblies have been discharged from Haddam Neck, Yankee-Rowe, Indian Point, San Onofre, and LaCrosse. Table A-11 lists the number of assemblies discharged, MTHM, and storage sites for each plant.

Tables A-12 and A-13 list the postirradiation elemental distributions for typical fuels. The data in these tables include the fuel, cladding material, and assembly hardware.

Table A-10. Total projected radionuclide inventories.^a

Isotope	Pressurized-water reactor			Boiling-water reactor			Grand totals (curies)	
	Curies per MTHM ^b	Total curies		Curies per MTHM	Total curies		Proposed Action	Additional modules
		Proposed Action	Additional modules		Proposed Action	Additional modules		
Hydrogen-3	2.1×10 ²	8.6×10 ⁶	1.4×10 ⁷	1.7×10 ²	3.8×10 ⁶	6.4×10 ⁶	1.2×10 ⁷	2.1×10 ⁷
Carbon-14	1.4	5.7×10 ⁴	9.5×10 ⁴	1.5	3.4×10 ⁴	5.7×10 ⁴	9.1×10 ⁴	1.5×10 ⁵
Chlorine-36	1.2×10 ⁻²	4.7×10 ²	7.9×10 ²	1.1×10 ⁻²	2.5×10 ²	4.1×10 ²	7.2×10 ²	1.2×10 ³
Cobalt-60	3.2×10 ²	1.3×10 ⁷	2.2×10 ⁷	1.9×10 ²	4.2×10 ⁶	7.0×10 ⁶	1.7×10 ⁷	2.9×10 ⁷
Nickel-59	2.8	1.1×10 ⁵	1.9×10 ⁵	1.8	4.0×10 ⁴	6.6×10 ⁴	1.5×10 ⁵	2.6×10 ⁵
Nickel-63	3.8×10 ²	1.6×10 ⁷	2.6×10 ⁷	2.3×10 ²	5.1×10 ⁶	8.6×10 ⁶	2.1×10 ⁷	3.5×10 ⁷
Selenium-79	4.9×10 ⁻¹	2.0×10 ⁴	3.3×10 ⁴	4.0×10 ⁻¹	8.9×10 ³	1.5×10 ⁴	2.9×10 ⁴	4.8×10 ⁴
Krypton-85	2.0×10 ³	8.2×10 ⁷	1.4×10 ⁸	1.5×10 ³	3.3×10 ⁷	5.5×10 ⁷	1.1×10 ⁸	1.9×10 ⁸
Strontium-90	4.6×10 ⁴	1.9×10 ⁹	3.1×10 ⁹	3.6×10 ⁴	8.0×10 ⁸	1.3×10 ⁹	2.7×10 ⁹	4.5×10 ⁹
Zirconium-93	2.5	1.0×10 ⁵	1.7×10 ⁵	2.4	5.4×10 ⁴	9.0×10 ⁴	1.6×10 ⁵	2.6×10 ⁵
Niobium-93m	1.8	7.3×10 ⁴	1.2×10 ⁵	1.8	3.9×10 ⁴	6.6×10 ⁴	1.1×10 ⁵	1.9×10 ⁵
Niobium-94	1.3	5.1×10 ⁴	8.6×10 ⁴	9.8×10 ⁻²	2.2×10 ³	3.6×10 ³	5.3×10 ⁴	8.9×10 ⁴
Technetium-99	1.5×10 ¹	6.3×10 ⁵	1.1×10 ⁶	1.3×10 ¹	2.9×10 ⁵	4.8×10 ⁵	9.2×10 ⁵	1.5×10 ⁶
Rhodium-102	2.6×10 ⁻³	1.1×10 ²	1.8×10 ²	1.4×10 ⁻³	3.2×10 ¹	5.3×10 ¹	1.4×10 ²	2.3×10 ²
Ruthenium-106	1.0×10 ⁻²	4.2×10 ²	7.0×10 ²	3.4×10 ⁻³	7.5×10 ¹	1.3×10 ²	5.0×10 ²	8.3×10 ²
Palladium-107	1.4×10 ⁻¹	5.6×10 ³	9.4×10 ³	1.2×10 ⁻¹	2.7×10 ³	4.5×10 ³	8.3×10 ³	1.4×10 ⁴
Tin-126	9.4×10 ⁻¹	3.8×10 ⁴	6.4×10 ⁴	7.9×10 ⁻¹	1.7×10 ⁰	2.9×10 ⁴	5.6×10 ⁴	9.3×10 ⁴
Iodine-129	3.8×10 ⁻²	1.5×10 ³	2.6×10 ³	3.2×10 ⁻²	7.0×10 ²	1.2×10 ³	2.2×10 ³	3.8×10 ³
Cesium-134	3.5×10 ¹	1.4×10 ⁶	2.4×10 ⁶	1.7×10 ¹	3.8×10 ⁵	6.4×10 ⁵	1.8×10 ⁶	3.0×10 ⁶
Cesium-135	5.5×10 ⁻¹	2.3×10 ⁴	3.8×10 ⁴	5.1×10 ⁻¹	1.1×10 ⁴	1.9×10 ⁴	3.4×10 ⁴	5.6×10 ⁴
Cesium-137	6.7×10 ⁴	2.8×10 ⁹	4.6×10 ⁹	5.4×10 ⁴	1.2×10 ⁹	2.0×10 ⁹	4.0×10 ⁹	6.6×10 ⁹
Samarium-151	4.0×10 ²	1.6×10 ⁷	2.7×10 ⁷	3.4×10 ²	7.4×10 ⁰	1.2×10 ⁷	2.4×10 ⁷	4.0×10 ⁷
Lead-210	4.8×10 ⁻⁷	2.0×10 ⁻²	3.3×10 ⁻²	4.8×10 ⁻⁷	1.1×10 ⁻²	1.8×10 ⁻²	3.0×10 ⁻²	5.1×10 ⁻²
Radium-226	2.0×10 ⁻⁶	8.2×10 ⁻²	1.4×10 ⁻¹	1.9×10 ⁻⁶	4.2×10 ⁻²	7.0×10 ⁻²	1.2×10 ⁻¹	2.1×10 ⁻¹
Radium-228	2.8×10 ⁻¹⁰	1.1×10 ⁻⁵	1.9×10 ⁻⁵	2.4×10 ⁻¹⁰	5.3×10 ⁻⁶	8.9×10 ⁻⁶	1.7×10 ⁻⁵	2.8×10 ⁻⁵
Actinium-227	1.7×10 ⁻⁵	6.9×10 ⁻¹	1.2	1.6×10 ⁻⁵	3.5×10 ⁻¹	5.8×10 ⁻¹	1.0	1.7
Thorium-229	3.8×10 ⁻⁷	1.5×10 ⁻²	2.6×10 ⁻²	3.1×10 ⁻⁷	6.9×10 ⁻³	1.2×10 ⁻²	2.2×10 ⁻²	3.7×10 ⁻²
Thorium-230	3.3×10 ⁻⁴	1.4×10 ¹	2.3×10 ¹	3.0×10 ⁻⁴	6.6×10	1.1×10 ¹	2.0×10 ¹	3.4×10 ¹
Thorium-232	4.1×10 ⁻¹⁰	1.7×10 ⁻⁵	2.8×10 ⁻⁵	3.5×10 ⁻¹⁰	7.8×10 ⁻⁶	1.3×10 ⁻⁵	2.5×10 ⁻⁵	4.1×10 ⁻⁵
Protactinium-231	3.4×10 ⁻⁵	1.4	2.3	3.1×10 ⁻⁵	6.8×10 ⁻¹	1.1	2.1	3.5
Uranium-232	4.0×10 ⁻²	1.6×10 ³	2.7×10 ³	2.8×10 ⁻²	6.2×10 ²	1.0×10 ³	2.3×10 ³	3.8×10 ³
Uranium-233	7.1×10 ⁻⁵	2.9	4.9	5.4×10 ⁻⁵	1.2	2.0	4.1	6.9
Uranium-234	1.4	5.8×10 ⁴	9.7×10 ⁴	1.2	2.7×10 ⁴	4.5×10 ⁴	8.5×10 ⁴	1.4×10 ⁵
Uranium-235	1.8×10 ⁻²	7.4×10 ²	1.2×10 ³	1.5×10 ⁻²	3.4×10 ²	5.6×10 ²	1.1×10 ³	1.8×10 ³
Uranium-236	3.0×10 ⁻¹	1.2×10 ⁴	2.1×10 ⁴	2.4×10 ⁻¹	5.4×10 ³	9.0×10 ³	1.8×10 ⁴	3.0×10 ⁴
Uranium-238	3.1×10 ⁻¹	1.3×10 ⁴	2.2×10 ⁴	3.2×10 ⁻¹	7.0×10 ³	1.2×10 ⁴	2.0×10 ⁴	3.3×10 ⁴
Neptunium-237	4.9×10 ⁻¹	2.0×10 ⁴	3.4×10 ⁴	3.7×10 ⁻¹	8.2×10 ³	1.4×10 ⁴	2.8×10 ⁴	4.7×10 ⁴
Plutonium-238	3.6×10 ³	1.5×10 ⁸	2.5×10 ⁸	2.8×10 ³	6.1×10 ⁷	1.0×10 ⁸	2.1×10 ⁸	3.5×10 ⁸
Plutonium-239	3.9×10 ²	1.6×10 ⁷	2.7×10 ⁷	3.2×10 ²	7.1×10 ⁶	1.2×10 ⁷	2.3×10 ⁷	3.9×10 ⁷
Plutonium-240	5.8×10 ²	2.4×10 ⁷	4.0×10 ⁷	4.9×10 ²	1.1×10 ⁷	1.8×10 ⁷	3.4×10 ⁷	5.8×10 ⁷
Plutonium-241	4.4×10 ⁴	1.8×10 ⁹	3.0×10 ⁹	3.8×10 ⁴	8.4×10 ⁸	1.4×10 ⁹	2.6×10 ⁹	4.4×10 ⁹
Plutonium-242	2.1	8.7×10 ⁴	1.5×10 ⁵	2.0	4.5×10 ⁴	7.5×10 ⁴	1.3×10 ⁵	2.2×10 ⁵
Americium-241	3.7×10 ³	1.5×10 ⁸	2.5×10 ⁸	3.5×10 ³	7.7×10 ⁷	1.3×10 ⁸	2.3×10 ⁸	3.8×10 ⁸
Americium-242/242m	2.3×10 ¹	9.3×10 ⁵	1.6×10 ⁶	2.3×10 ¹	5.2×10 ⁵	8.7×10 ⁵	1.4×10 ⁶	2.4×10 ⁶
Americium-243	2.7×10 ¹	1.1×10 ⁶	1.9×10 ⁶	2.5×10 ¹	5.5×10 ⁵	9.2×10 ⁵	1.7×10 ⁶	2.8×10 ⁶
Curium-242	1.9×10 ¹	7.7×10 ⁵	1.3×10 ⁶	1.9×10 ¹	4.3×10 ⁵	7.1×10 ⁵	1.2×10 ⁶	2.0×10 ⁶
Curium-243	1.8×10 ¹	7.3×10 ⁵	1.2×10 ⁶	1.6×10 ¹	3.5×10 ⁵	5.8×10 ⁵	1.1×10 ⁶	1.8×10 ⁶
Curium-244	1.5×10 ³	6.2×10 ⁷	1.0×10 ⁸	1.3×10 ³	2.8×10 ⁷	4.7×10 ⁷	9.0×10 ⁷	1.5×10 ⁸
Curium-245	3.9×10 ⁻¹	1.6×10 ⁴	2.7×10 ⁴	3.2×10 ⁻¹	7.1×10 ³	1.2×10 ⁴	2.3×10 ⁴	3.8×10 ⁴
Curium-246	8.2×10 ⁻²	3.4×10 ³	5.6×10 ³	6.5×10 ⁻²	1.4×10 ³	2.4×10 ³	4.8×10 ³	8.0×10 ³
Curium-247	2.9×10 ⁻⁷	1.2×10 ⁻²	2.0×10 ⁻²	2.2×10 ⁻⁷	4.8×10 ⁻³	8.1×10 ⁻³	1.6×10 ⁻²	2.8×10 ⁻²
Curium-248	8.3×10 ⁻⁷	3.4×10 ⁻²	5.7×10 ⁻²	6.1×10 ⁻⁷	1.4×10 ⁻²	2.3×10 ⁻²	4.8×10 ⁻²	8.0×10 ⁻²
Californium-252	6.7×10 ⁻⁸	2.8×10 ⁻³	4.6×10 ⁻³	3.1×10 ⁻⁸	6.8×10 ⁻⁴	1.1×10 ⁻³	3.4×10 ⁻³	5.7×10 ⁻³

a. Source: Compilation of Tables A-8 and A-9.

b. MTHM = metric tons of heavy metal.

Table A-11. Stainless-steel-clad spent nuclear fuel inventory.^a

Discharging reactor	Storage location	Assemblies	MTHM ^b
Yankee-Rowe	Yankee-Rowe	76	21
San Onofre 1	San Onofre	395	144
San Onofre 1	Morris, Illinois	270	99
Indian Point 1	Indian Point	160	31
LaCrosse	LaCrosse	333	38
Haddam Neck	Haddam Neck	871	360
Haddam Neck	Morris, Illinois	82	34
Totals		2,187	727

a. Source: Cole (1998b, all).

b. MTHM = metric tons of heavy metal.

Table A-12. Elemental distribution of typical pressurized-water reactor fuel.^a

Element	Grams per assembly ^b	Percent total ^c	Element	Grams per assembly ^b	Percent total ^c
Aluminum	47	0.01	Oxygen	62,000	9.35
Americium	600	0.09	Palladium	790	0.12
Barium	1,200	0.18	Phosphorus	85	0.01
Cadmium	77	0.01	Plutonium	4,600	0.69
Carbon	77	0.01	Praseodymium	610	0.09
Cerium	1,300	0.20	Rhodium	230	0.04
Cesium	1,100	0.17	Rubidium	200	0.03
Chromium	4,300	0.65	Ruthenium	1,200	0.18
Cobalt	38	0.01	Samarium	470	0.07
Europium	72	0.01	Silicon	170	0.03
Gadolinium	81	0.01	Silver	40	0.01
Iodine	130	0.02	Strontium	330	0.05
Iron	12,000	1.85	Technetium	420	0.06
Krypton	190	0.03	Tellurium	270	0.04
Lanthanum	670	0.10	Tin	1,900	0.29
Manganese	330	0.05	Titanium	51	0.01
Molybdenum	2,000	0.31	Uranium	440,000	65.78
Neodymium	2,200	0.33	Xenon	2,900	0.43
Neptunium	330	0.05	Yttrium	250	0.04
Nickel	5,000	0.75	Zirconium	120,000	17.77
Niobium	330	0.05			
Nitrogen	49	0.01	Totals	668,637	99.99

a. Source: DOE (1992, page 1.1-1).

b. To convert grams to ounces, multiply by 0.035274.

c. Table only includes elements that constitute at least 0.01 percent of the total; therefore, the total of the percentage column is slightly less than 100 percent.

A.2.1.5.4 Thermal Output

Heat generation rates are available as a function of spent fuel type, enrichment, burnup, and decay time in the Light-Water Reactor Radiological Database, which is an integral part of the *Characteristics Potential Repository Wastes* (DOE 1992, page 1.1-1). Table A-14 lists the thermal profiles for the typical pressurized-water reactor and boiling-water reactor assemblies from the Light-Water Reactor Radiological Database. For the EIS analysis, the typical thermal profile, applied across the proposed inventory, yields a good approximation of the expected thermal load in the repository. Figure A-6 shows these profiles as a function of time.

Table A-13. Elemental distribution of typical boiling-water reactor fuel.^a

Element	Grams per assembly ^b	Percent total ^c	Element	Grams per assembly ^b	Percent total ^c
Aluminum	31	0.01	Nitrogen	25	0.01
Americium	220	0.07	Oxygen	25,000	7.82
Barium	390	0.12	Palladium	270	0.09
Cadmium	27	0.01	Plutonium	1,500	0.48
Carbon	36	0.01	Praseodymium	200	0.06
Cerium	430	0.14	Rhodium	79	0.03
Cesium	390	0.12	Rubidium	64	0.02
Chromium	1,900	0.60	Ruthenium	410	0.13
Cobalt	26	0.01	Samarium	160	0.05
Europium	24	0.01	Silicon	80	0.03
Gadolinium	310	0.10	Strontium	110	0.03
Iodine	43	0.01	Technetium	140	0.04
Iron	5,100	1.63	Tellurium	91	0.03
Krypton	62	0.02	Tin	1,600	0.50
Lanthanum	220	0.07	Titanium	83	0.03
Manganese	160	0.05	Uranium	170,000	55.35
Molybdenum	630	0.20	Xenon	950	0.30
Neodymium	730	0.23	Yttrium	81	0.03
Neptunium	97	0.03	Zirconium	96,000	30.52
Nickel	3,000	0.94			
Niobium	29	0.01	Totals	310,698	99.94

a. Source: DOE (1992, page 1.1-1).

b. To convert grams to ounces, multiply by 0.035274.

c. Table only includes elements that contribute at least 0.01 percent of the total; therefore, the total of the percentage column is slightly less than 100 percent.

Table A-14. Typical assembly thermal profiles.^a

Years after discharge	Pressurized-water reactor		Boiling-water reactor	
	W/MTHM ^b	W/assembly ^c	W/MTHM	W/assembly ^d
1	10,500	4,800	8,400	1,500
3	3,700	1,700	3,000	550
5	2,200	1,000	1,800	340
10	1,500	670	1,200	220
26	990	450	820	150
30	920	420	770	140
50	670	310	570	100
100	370	170	320	58
300	160	73	140	26
500	120	53	100	19
1,000	66	31	58	11
2,000	35	16	30	5
5,000	22	10	19	3
10,000	16	8	13	3

a. Source: DOE (1992, page 1.1-1).

b. W/MTHM = watts per metric ton of heavy metal; to convert metric tons to tons, multiply by 1.1023.

c. W/assembly = watts per assembly; assumes 0.46 MTHM per assembly.

d. Assumes 0.18 MTHM per assembly.

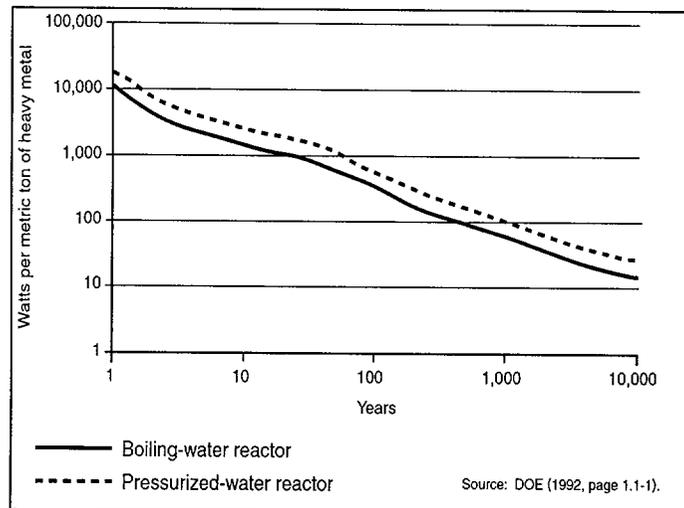


Figure A-6. Typical thermal profiles over time.

A.2.1.5.5 Physical Parameters

Table A-15 lists reference characteristics of typical pressurized-water and boiling-water reactor fuel assemblies. These data are from the *Integrated Data Base Report* (DOE 1997b, page 1-8) and reflect characteristics of unirradiated assemblies.

Table A-15. Reference characteristics for unirradiated typical fuel assemblies.^a

Characteristics ^b	Boiling-water reactor	Pressurized-water reactor
Overall assembly length (meters)	4.5	4.1
Cross section (centimeters)	14 × 14	21 × 21
Fuel rod length (meters)	4.1	3.9
Active fuel height (meters)	3.8	3.7
Fuel rod outer diameter (centimeters)	1.3	0.95
Fuel rod array	8 × 8	17 × 17
Fuel rods per assembly	63	264
Assembly total weight (kilograms)	320	660
Uranium per assembly (kilograms)	180	460
Uranium oxide per assembly (kilograms)	210	520
Zirconium alloy per assembly (kilograms)	100 ^c	110 ^d
Hardware per assembly (kilograms)	8.6 ^e	26 ^f
Nominal volume per assembly (cubic meters)	0.086 ^g	0.19 ^g

a. Source: DOE (1997b, page 1-8).

b. To convert meters to feet, multiply by 3.2808; to convert centimeters to inches, multiply by 0.3937; to convert kilograms to pounds, multiply by 2.2046; to convert cubic meters to cubic feet, multiply by 35.314.

c. Includes zirconium alloy fuel rod spacers and fuel channels.

d. Includes zirconium alloy control rod guide thimbles.

e. Includes stainless-steel tie plates, Inconel springs, and plenum springs.

f. Includes stainless-steel nozzles and Inconel-718 grids.

g. Based on overall outside dimension; includes spacing between the stacked fuel rods of the assembly.

For additional details, the Light-Water Reactor Assembly Database contains individual physical descriptions of the fuel assemblies and fuel pins. The Light-Water Reactor Nonfuel Assembly Hardware Database contains physical and radiological descriptions of nonfuel assembly hardware. These databases are integral parts of the *Characteristics of Potential Repository Wastes* (DOE 1992, Section 2.8).

A.2.2 DOE SPENT NUCLEAR FUEL

A.2.2.1 Background

At present, DOE stores most of its spent nuclear fuel at three primary locations: the Hanford Site in Washington State, the Idaho National Engineering and Environmental Laboratory in Idaho, and the Savannah River Site in South Carolina. Some DOE spent nuclear fuel is stored at the Fort St. Vrain dry storage facility in Colorado. Much smaller quantities remain at other locations (LMIT 1997, all). DOE issued the *Record of Decision – Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* on June 1, 1995 (DOE 1995b, all) and amended it in March 1996 (DOE 1996, all). The Record of Decision and its amendment specify three primary locations as storage sites for DOE spent nuclear fuel. With the exception of Fort St. Vrain, which will retain its spent nuclear fuel in dry storage, DOE will ship all its spent nuclear fuel from other sites to one of the three primary sites for storage and preparation for ultimate disposition.

During the last four decades, DOE and its predecessor agencies have generated more than 200 varieties of spent nuclear fuel from weapons production, nuclear propulsion, and research missions. A method described by Fillmore (1998, all) allows grouping of these many varieties of spent nuclear fuel into 16 categories for the repository Total System Performance Assessment. The grouping method uses regulatory requirements to identify the parameters that would affect the performance of DOE spent nuclear fuel in the repository and meet analysis needs for the repository License Application. Three fuel parameters (fuel matrix, fuel compound, and cladding condition) would influence repository performance behavior. The grouping methodology presents the characteristics of a select number of fuel types in a category that either bound or represent a particular characteristic of the whole category. Table A-16 lists these spent nuclear fuel categories.

Table A-16 includes sodium-bonded fuel (Category 14); however, DOE is considering a proposal to treat and manage sodium-bonded spent nuclear fuel for disposal. Alternatives being considered include processing and converting some or all of its sodium-bonded fuel to a high-level radioactive waste form before shipment. Section A.2.3, which covers data associated with high-level radioactive waste, includes data on waste produced from potential future treatment of Category 14 spent nuclear fuel (Dirkmaat 1997b, page 7).

A.2.2.2 Sources

The DOE National Spent Fuel Program maintains a spent nuclear fuel data base (LMIT 1997, all). Table A-16 provides a brief description of each of the fuel categories and a typical fuel. Section A.2.2.5.3 provides more detail on the chemical makeup of each category.

A.2.2.3 Present Storage and Generation Status

Table A-17 lists storage locations and inventory information on DOE spent nuclear fuels. During the preparation of the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995c, all), DOE evaluated and categorized all the materials listed in the table as spent nuclear fuel, in accordance with the definition in the Nuclear Waste Policy Act, as amended.

Table A-16. DOE spent nuclear fuel categories.^{a,b}

	DOE SNF category	Typically from	Description of fuel
1.	Uranium metal	N-Reactor	Uranium metal fuel compounds with aluminum or zirconium alloy cladding
2.	Uranium-zirconium	HWCTR	Uranium alloy fuel compounds with zirconium alloy cladding
3.	Uranium-molybdenum	Fermi	Uranium-molybdenum alloy fuel compounds with zirconium alloy cladding
4.	Uranium oxide, intact	Commercial PWR	Uranium oxide fuel compounds with zirconium alloy or stainless-steel cladding in fair to good condition
5.	Uranium oxide, failed/declad/aluminum clad	TMI core debris	Uranium oxide fuel compounds: (1) without cladding; (2) clad with zirconium alloy, Hastelloy, nickel-chromium, or stainless steel in poor or unknown condition; or (3) nondegraded aluminum clad
6.	Uranium-aluminide	ATR	Uranium-aluminum alloy fuel compounds with aluminum cladding
7.	Uranium-silicide	FRR MTR	Uranium silicide fuel compounds with aluminum cladding
8.	Thorium/uranium carbide, high-integrity	Fort St. Vrain	Thorium/uranium carbide fuel compounds with graphite cladding in good condition
9.	Thorium/uranium carbide, low-integrity	Peach Bottom	Thorium/uranium carbide fuel compounds with graphite cladding in unknown condition
10.	Plutonium/uranium carbide, nongraphite	FFTF carbide	Uranium carbide or plutonium-uranium carbide fuel compounds with or without stainless-steel cladding
11.	Mixed oxide	FFTF oxide	Plutonium/uranium oxide fuel compounds in zirconium alloy, stainless-steel, or unknown cladding
12.	Uranium/thorium oxide	Shippingport LWBR	Uranium/thorium oxide fuel compounds with zirconium alloy or stainless-steel cladding
13.	Uranium-zirconium hydride	TRIGA	Uranium-zirconium hydride fuel compounds with or without Incalloy, stainless-steel, or aluminum cladding
14.	Sodium-bonded	EBR-II driver and blanket, Fermi-I blanket	Uranium and uranium-plutonium metallic alloy with predominantly stainless-steel cladding
15.	Naval fuel	Surface ship/submarine	Uranium-based with zirconium alloy cladding
16.	Miscellaneous	Not specified	Various fuel compounds with or without zirconium alloy, aluminum, Hastelloy, tantalum, niobium, stainless-steel or unknown cladding

a. Source: Fillmore (1998, all).

b. Abbreviations: SNF = spent nuclear fuel; HWCTR = heavy-water cooled test reactor; PWR = pressurized-water reactor; TMI = Three Mile Island; ATR = Advanced Test Reactor; FRR MTR = foreign research reactor – material test reactor; FFTF = Fast Flux Test Facility; LWBR = light-water breeder reactor; TRIGA = Training Research Isotopes – General Atomic; EBR-II = Experimental Breeder Reactor II.

A.2.2.4 Final Spent Nuclear Fuel Form

For all spent nuclear fuel categories except 14, the expected final spent nuclear fuel form does not differ from the current or planned storage form. Before its disposal in the repository, candidate material would be in compliance with approved acceptance criteria.

DOE has prepared an EIS at the Savannah River Site (DOE 1998d, all) to evaluate potential treatment alternatives for spent nuclear fuel and its ultimate disposal in the repository. The products of any proposed treatment of the Savannah River Site aluminum-based fuels are adequately represented by the

Table A-17. National Spent Nuclear Fuel Database projection of DOE spent nuclear fuel locations and inventories to 2035.^{a,b}

Fuel category and name	Storage Site	No. of units ^c	Mass (kilograms) ^d	Volume (cubic meters) ^e	Fissile mass (kilograms)	Equivalent uranium mass (kilograms)	MTHM
1. Uranium metal ^f	INEEL	85	4,500	0.7	13	1,700	1.7
	Hanford	100,000	2,160,000	200	25,000	2,100,000	2100
	SRS	350	120,000	18	110	17,000	17
	<i>Totals</i>	<i>100,435</i>	<i>2,284,500</i>	<i>218.7</i>	<i>25,123</i>	<i>2,118,700</i>	<i>2119</i>
2. Uranium-zirconium	INEEL	69	120	0.7	34	40	0.04
3. Uranium-molybdenum	INEEL	29,000	4,600	0.3	970	3,800	3.8
4. Uranium oxide, intact	INEEL	14,000	150,000	41	2,200	80,000	80
	Hanford	87	44,000	11	240	18,000	18
	<i>Totals</i>	<i>14,087</i>	<i>194,000</i>	<i>52</i>	<i>2,440</i>	<i>98,000</i>	<i>99</i>
5. Uranium oxide, failed/declad/aluminum clad	INEEL	2,000	340,000	140	2,200	83,000	84
	Hanford	13	270	4.2	4	160	0.2
	SRS	7,600	58,000	96	2,600	3,200	3.2
	<i>Totals</i>	<i>9,613</i>	<i>398,270</i>	<i>240.2</i>	<i>4,804</i>	<i>86,360</i>	<i>87</i>
6. Uranium-aluminide	SRS	18,000	130,000	150	6,000	8,800	8.7
7. Uranium-silicide	SRS	7,400	47,000	53	1,200	12,000	12
8. Thorium/uranium carbide, high-integrity	FSV	1,500	190,000	130	640	820	15
	INEEL	1,600	130,000	82	350	440	9.9
	<i>Totals</i>	<i>3,100</i>	<i>320,000</i>	<i>212</i>	<i>990</i>	<i>1,260</i>	<i>25</i>
9. Thorium/uranium carbide, low-integrity	INEEL	810	55,000	17	180	210	1.7
10. Plutonium/uranium carbide, nongraphite	INEEL	130	140	0	10	73	0.08
	Hanford	2	330	0.1	11	64	0.07
	<i>Totals</i>	<i>132</i>	<i>470</i>	<i>0.1</i>	<i>21</i>	<i>137</i>	<i>0.2</i>
11. Mixed oxide	INEEL	2,000	6,100	2.4	240	2,000	2.1
	Hanford	620	110,000	33	2,400	8,000	10
	<i>Totals</i>	<i>2,620</i>	<i>116,100</i>	<i>35.1</i>	<i>2,640</i>	<i>10,000</i>	<i>12</i>
12. Uranium/thorium oxide	INEEL	260	120,000	18	810	810	50
13. Uranium-zirconium hydride	INEEL	9,800	33,000	8.1	460	2,000	2
	Hanford	190	660	33	7	36	0.04
	<i>Totals</i>	<i>9,990</i>	<i>33,660</i>	<i>8.3</i>	<i>467</i>	<i>2,036</i>	<i>2</i>
15. Naval fuel ^{g,h}	INEEL	300	4,400,000	888	64,000	65,000	65
16. Miscellaneous	INEEL	1,500	33,000	11	360	5,500	7.7
	Hanford	73	1,700	0.2	30	130	0.2
	SRS	8,800	9,200	8.2	550	2,900	2.9
	<i>Totals</i>	<i>10,373</i>	<i>43,900</i>	<i>19.4</i>	<i>940</i>	<i>8,530</i>	<i>11</i>
Grand totals		210,000	8,150,000	1,900	110,000	2,420,000	2,500

- Source: Dirkmaat (1998a, all); individual values and totals rounded to two significant figures.
- Abbreviations: SNF = spent nuclear fuel; INEEL = Idaho National Engineering and Environmental Laboratory; SRS = Savannah River Site; FSV = Fort St. Vrain.
- Unit is defined as an assembly, bundle of elements, can of material, etc., depending on the particular spent nuclear fuel category.
- To convert kilograms to pounds, multiply by 2.2046; to convert metric tons to tons, multiply by 1.1023.
- To convert cubic meters to cubic yards, multiply by 1.3079.
- N-Reactor fuel is stored in aluminum or stainless-steel cans at the K-East and K-West Basins. The mass listed in this table does not include the storage cans.
- Information supplied by the Navy (Dirkmaat 1997a, Attachment, page 2).
- A naval fuel unit consists of a naval dual-purpose canister that contains multiple assemblies.

properties of the present aluminum-based fuel (Categories 6, 7, and part of 5) for this Yucca Mountain EIS. They are bounded by the same total radionuclide inventory, heat generation rates, dissolution rates, and number of canisters. No additional data about the products will be required to ensure that they are represented in the EIS inventory.

A.2.2.5 Spent Nuclear Fuel Characteristics

A.2.2.5.1 Mass and Volume

Table A-17 lists total volume, mass, and MTHM for each DOE spent nuclear fuel category from the National Spent Nuclear Fuel Database (LMIT 1997, all).

A.2.2.5.2 Amount and Nature of Radioactivity

ORIGEN2 (Oak Ridge Isotope Generation), an accepted computer code for calculating spent nuclear fuel radionuclide inventories, was used to generate activity data for radionuclides in the DOE spent nuclear fuel inventory. The inventory came from the 1997 version of the National Spent Nuclear Fuel Database (LMIT 1997, all).

Table A-18 lists the activities expressed in terms of curies per handling unit for the radionuclides of interest (uranium, fission products and actinides). The table lists activity estimates decayed to 2030 for all categories except 15. A handling unit for DOE is a spent nuclear fuel canister, while for Category 15 naval fuels, it is a naval dual-purpose canister.

The activity for naval spent nuclear fuel is provided for typical submarine (15a) or surface ship (15b) spent nuclear fuels. Dirkmaat (1997a, Attachment, pages 3 to 5) provided these activities for 5 years after shutdown, which would be the minimum cooling time before naval fuel would reach the repository. The power history assumed operations at power for a full core life. The assumptions about the power history and minimum cooling time conservatively bound the activity for naval fuel that would be emplaced in a monitored geologic repository. In addition, ORIGEN2 was used to calculate the activity associated with activation products in the cladding, which are listed in Table A-18. For completeness, the data also include the activity that would be present in the activated corrosion products deposited on the fuel.

A.2.2.5.3 Chemical Composition

This section discusses the chemical compositions of each of the 16 categories of DOE spent nuclear fuel (Dirkmaat 1998a, all).

- **Category 1: Uranium metal.** The fuel in this category consists primarily of uranium metal. N-reactor fuel represents the category because its mass is so large that the performance of the rest of the fuel in the category, even if greatly different from N-Reactor fuel, would not change the overall category performance. The fuel is composed of uranium metal about 1.25 percent enriched in uranium-235, and is clad with a zirconium alloy. Approximately 50 percent of the fuel elements are believed to have failed cladding. This fuel typically has low burnup. Other contributors to this category include the Single Pass Reactor fuel at Hanford and declad Experimental Breeder Reactor-II blanket material at the Savannah River Site.
- **Category 2: Uranium-zirconium.** The fuel in this category consists primarily of a uranium- (91-percent) zirconium alloy. The Heavy Water Components Test Reactor fuel is the representative fuel because it is the largest part of the inventory. This fuel is approximately 85-percent enriched in uranium-235 and is clad with a zirconium alloy.
- **Category 3: Uranium molybdenum.** The fuel in this category consists of uranium- (10 percent)-molybdenum alloy and 25-percent enriched in uranium-235, and is clad with a zirconium alloy. Fermi driver core 1 and 2 are the only fuels in the category. The fuel is currently in an aluminum container. The proposed disposition would include the aluminum container.

Table A-18. Radionuclide activity by DOE spent nuclear fuel category^a (page 1 of 2).

Storage site ^b	Category ^c															
	1	2	3	4	5	6	7	8	9	10	11	12	13	15a ^d	15b	16
	Number of handling units															
Hanford	440	0	0	34	1	0	0	0	0	2	324	0	3	0	0	5
INEEL	6	8	70	195	406	0	0	503 ^e	60	3	43	71	97	200	100	39
SRS	9	0	0	0	425	750	225	0	0	0	0	0	0	0	0	2
Totals	455	8	70	229	832	750	225	503	60	5	367	71	100	200	100	46
	Curies per handling unit															
Radio-nuclide ^f																
Ac-227	2.2×10 ⁻⁵	4.8×10 ⁻⁹	6.9×10 ⁻⁶	1.7×10 ⁻⁴	1.4×10 ⁻⁵	3.4×10 ⁻⁷	2.3×10 ⁻⁷	0	2.8×10 ⁻³	8.9×10 ⁻⁹	1.5×10 ⁻⁹	4.3×10 ⁻¹	5.6×10 ⁻⁸	1.3×10 ⁻⁴	1.6×10 ⁻⁴	6.8×10 ⁻⁷
Am-241	1.1×10 ³	3.9×10 ⁻¹	4.6×10 ⁻⁵	1.6×10 ³	7.3	3.3	3.6×10 ¹	3.7	2.7	2.4×10 ²	4.3×10 ²	8.3×10 ¹	2.0×10 ⁻¹	4.9×10 ¹	6.7×10 ¹	1.2×10 ²
Am-242m	6.6×10 ⁻²	1.2×10 ⁻³	0	2.6	1.4×10 ⁻²	2.3×10 ⁻³	1.3×10 ⁻²	1.0×10 ⁻³	1.4×10 ⁻³	4.1×10 ⁻¹	7.5×10 ⁻¹	8.7×10 ⁻³	2.3×10 ⁻³	6.6×10 ⁻¹	8.5×10 ⁻¹	1.5×10 ⁻¹
Am-243	2.8×10 ⁻¹	3.8×10 ⁻³	7.3×10 ⁻¹³	8.3	2.2×10 ⁻²	2.5×10 ⁻³	3.6×10 ⁻²	2.7×10 ⁻²	1.3×10 ⁻³	6.7×10 ⁻³	1.8×10 ⁻¹	1.7×10 ⁻³	2.5×10 ⁻⁴	6.2×10 ⁻¹	1.1	4.9×10 ⁻¹
C-14	1.5	8.2×10 ⁻⁶	2.2×10 ⁻³	1.0×10 ⁻¹	1.1×10 ⁻³	9.9×10 ⁻⁷	1.8×10 ⁻⁵	2.2×10 ⁻¹	3.7×10 ⁻²	1.5×10 ⁻⁵	9.9×10 ⁻⁴	6.7×10 ⁻¹	8.5×10 ⁻²	2.7×10 ¹	4.6×10 ¹	1.7×10 ⁻³
Cf-252	-- ^f	--	--	--	--	--	--	--	--	--	--	--	--	2.8×10 ⁸	1.4×10 ⁷	--
Cl-36	0	0	5.6×10 ⁻⁶	3.5×10 ⁻⁴	1.7×10 ⁻⁵	0	0	2.7×10 ⁻³	1.1×10 ⁻³	0	1.1×10 ⁻⁵	1.5×10 ⁻²	2.6×10 ⁻³	1.0	1.8	4.2×10 ⁻⁶
Cm-242	< 7.4×10 ¹	< 7.4×10 ¹	0	< 7.4×10 ¹	< 7.3×10 ¹	< 7.4×10 ¹	1.5	2.2	< 7.4×10 ¹							
Cm-243	--	--	--	--	--	--	--	--	--	--	--	--	--	7.4×10 ⁻¹	2.8×10 ⁻²	--
Cm-244	8.5	1.6×10 ⁻¹	6.8×10 ⁻¹⁴	3.5×10 ²	9.3×10 ⁻¹	2.1×10 ⁻²	3.0×10 ⁻¹	8.3×10 ⁻¹	3.5×10 ⁻²	2.8×10 ⁻¹	7.6	1.6×10 ⁻¹	6.8×10 ⁻³	4.6×10 ¹	9.9×10 ¹	1.9×10 ¹
Cm-245	3.6×10 ⁻³	8.0×10 ⁻⁶	1.9×10 ⁻¹⁹	1.4×10 ⁻¹	3.8×10 ⁻⁴	1.8×10 ⁻⁶	2.0×10 ⁻⁵	1.4×10 ⁻⁴	4.0×10 ⁻⁶	1.4×10 ⁻⁵	3.1×10 ⁻³	3.3×10 ⁻⁵	1.4×10 ⁻⁷	3.8×10 ⁻³	9.1×10 ⁻³	7.1×10 ⁻³
Cm-246	5.3×10 ⁻⁴	5.5×10 ⁻⁷	6.1×10 ⁻²³	2.4×10 ⁻²	6.4×10 ⁻⁵	8.6×10 ⁻⁸	1.5×10 ⁻⁶	6.9×10 ⁻⁵	1.3×10 ⁻⁷	9.7×10 ⁻⁷	5.3×10 ⁻⁴	2.2×10 ⁻⁶	3.9×10 ⁻⁹	6.6×10 ⁻⁴	1.9×10 ⁻³	1.2×10 ⁻³
Cm-247	--	--	--	--	--	--	--	--	--	--	--	--	--	1.6×10 ⁻⁹	5.1×10 ⁻⁹	--
Cm-248	--	--	--	--	--	--	--	--	--	--	--	--	--	3.1×10 ⁻⁹	1.1×10 ⁻⁸	--
Co-60	1.4×10 ⁻¹	0	1.1×10 ⁻²	1.8×10 ¹	1.6×10 ⁻¹²	1.2×10 ⁻¹¹	2.0×10 ⁻¹⁰	0	2.5×10 ⁻²	1.8	1.4	4.3	1.8×10 ⁻¹	9.0×10 ²	1.6×10 ³	7.6×10 ⁻⁴
Cs-134	2.7×10 ⁻¹	4.6×10 ⁻²	1.9×10 ⁻⁸	9.6×10 ⁻²	8.3×10 ⁻³	1.7×10 ⁻¹	3.7×10 ⁻¹	7.6×10 ⁻³	3.6×10 ⁻⁷	3.4×10 ⁻²	7.5×10 ⁻³	6.0×10 ⁻³	3.3×10 ⁻⁴	3.1×10 ¹	5.5×10 ¹	5.7×10 ⁻¹
Cs-135	1.8×10 ⁻¹	7.7×10 ⁻³	4.5×10 ⁻³	1.8×10 ⁻¹	2.9×10 ⁻²	2.8×10 ⁻²	1.9×10 ⁻²	1.7×10 ⁻²	2.6×10 ⁻²	1.4×10 ⁻²	3.2×10 ⁻³	2.0×10 ⁻¹	3.2×10 ⁻²	3.9	4.7	1.4×10 ⁻¹
Cs-137	2.0×10 ⁴	7.4×10 ³	0	2.9×10 ⁴	3.6×10 ³	3.8×10 ³	8.1×10 ³	2.4×10 ³	1.9×10 ³	1.5×10 ⁴	4.0×10 ³	2.5×10 ³	3.1×10 ³	4.4×10 ⁵	5.5×10 ⁵	8.7×10 ⁴
H-3	2.3×10 ¹	4.4	8.6×10 ⁻²	3.6×10 ¹	1.3	5.9×10 ⁻¹	1.3×10 ¹	2.0	1.5	7.3	2.8	2.3×10 ¹	9.6×10 ⁻¹	1.5×10 ³	1.8×10 ³	1.3×10 ¹
I-129	1.6×10 ⁻²	1.6×10 ⁻³	1.2×10 ⁻⁴	1.8×10 ⁻²	7.5×10 ⁻⁴	1.8×10 ⁻³	3.8×10 ⁻³	2.1×10 ⁻³	7.3×10 ⁻⁴	2.9×10 ⁻³	3.6×10 ⁻⁴	1.1×10 ⁻²	7.2×10 ⁻⁴	1.1×10 ⁻¹	1.4×10 ⁻¹	2.3×10 ⁻²
Kr-85	3.6×10 ²	9.3×10 ¹	7.7×10 ⁻¹	3.1×10 ²	2.7×10 ¹	1.3×10 ²	2.6×10 ²	6.0×10 ¹	7.2	4.8×10 ¹	2.4×10 ¹	6.2×10 ²	1.7×10 ¹	3.8×10 ⁴	4.7×10 ⁴	4.2×10 ²
Nb-93m	8.0×10 ⁻¹	8.7×10 ⁻³	4.6×10 ⁻³	6.7×10 ⁻¹	1.1×10 ⁻²	1.6×10 ⁻²	3.1×10 ⁻²	9.2×10 ⁻³	4.6×10 ⁻²	1.5×10 ⁻²	1.3×10 ⁻²	3.1×10 ⁻¹	7.1×10 ⁻³	8.5	1.3×10 ¹	1.7×10 ⁻¹
Nb-94	5.7×10 ⁻⁶	1.6×10 ⁻⁶	8.4×10 ⁻⁴	7.3×10 ⁻³	4.2×10 ⁻⁵	3.1×10 ⁻⁶	7.4×10 ⁻⁶	1.3×10 ⁻⁴	4.9×10 ⁻⁴	2.9×10 ⁻⁶	1.9×10 ⁻⁵	1.6×10 ⁻²	4.6×10 ⁻³	2.1×10 ²	3.7×10 ²	3.5×10 ⁻⁵
Ni-59	8.2×10 ⁻²	0	6.9×10 ⁻³	9.4×10 ⁻²	2.3×10 ⁻⁴	0	0	1.7×10 ⁻²	1.5×10 ⁻³	0	2.1×10 ⁻³	5.1×10 ⁻²	5.0×10 ⁻¹	1.2	2.0	8.2×10 ⁻⁴
Ni-63	7.7	0	1.4×10 ⁻¹	3.0×10 ²	2.5×10 ⁻²	2.3×10 ⁻²²	0	4.1×10 ⁻¹	1.5×10 ⁻¹	5.0	8.7	6.2	6.2×10 ¹	1.3×10 ²	2.3×10 ²	1.0×10 ⁻¹
Np-237	1.7×10 ⁻¹	2.0×10 ⁻²	3.3×10 ⁻⁴	1.8×10 ⁻¹	3.1×10 ⁻³	1.2×10 ⁻²	1.8×10 ⁻²	1.6×10 ⁻²	7.4×10 ⁻³	3.7×10 ⁻²	6.5×10 ⁻³	7.1×10 ⁻⁴	1.9×10 ⁻³	2.9	4.0	2.4×10 ⁻¹
Pa-231	5.8×10 ⁻⁵	2.3×10 ⁻⁷	2.0×10 ⁻⁵	3.0×10 ⁻⁴	2.6×10 ⁻⁵	4.2×10 ⁻⁶	2.8×10 ⁻⁶	1.9×10 ⁻²	4.8×10 ⁻³	4.1×10 ⁻⁷	1.2×10 ⁻⁷	1.1	9.0×10 ⁻⁷	6.4×10 ⁻⁴	7.9×10 ⁻⁴	1.0×10 ⁻⁵
Pb-210	3.2×10 ⁻¹⁰	8.6×10 ⁻¹³	1.4×10 ⁻¹⁰	9.0×10 ⁻⁸	5.2×10 ⁻⁹	2.1×10 ⁻¹¹	1.2×10 ⁻¹¹	4.6×10 ⁻⁶	2.6×10 ⁻⁷	1.5×10 ⁻¹²	3.1×10 ⁻¹⁰	7.8×10 ⁻⁵	1.4×10 ⁻¹²	7.6×10 ⁻⁷	9×10 ⁻⁷	7.51×10 ⁻¹⁰

Table A-18. Radionuclide activity by DOE spent nuclear fuel category^a (page 2 of 2).

Radio-nuclide ^f	Category ^b															
	1	2	3	4	5	6	7	8	9	10	11	12	13	15a ^c	15b	16
	Curies per handling unit															
Pd-107	3.3×10 ⁻²	1.1×10 ⁻³	1.3×10 ⁻⁴	4.8×10 ⁻²	8.3×10 ⁻⁴	9.3×10 ⁻⁴	3.5×10 ⁻³	8.7×10 ⁻⁴	4.8×10 ⁻⁴	2.0×10 ⁻³	1.0×10 ⁻³	2.4×10 ⁻³	6.0×10 ⁻⁴	7.9×10 ⁻²	9.9×10 ⁻²	1.8×10 ⁻²
Pu-238	2.5×10 ²	4.3×10 ¹	1.7×10 ⁻²	1.2×10 ³	5.8	1.7×10 ¹	2.8×10 ¹	8.1×10 ¹	1.8×10 ¹	1.1×10 ²	7.9×10 ¹	2.8	2.1	1.4×10 ⁴	2.3×10 ⁴	5.3×10 ²
Pu-239	5.1×10 ²	1.1	2.0	1.5×10 ²	1.3×10 ¹	2.4	2.2×10 ¹	2.3×10 ⁻¹	4.1×10 ⁻¹	1.9×10 ²	3.2×10 ²	1.8×10 ⁻¹	4.5	1.3×10 ¹	1.8×10 ¹	5.2×10 ¹
Pu-240	3.0×10 ²	6.1×10 ⁻¹	6.1×10 ⁻³	2.4×10 ²	4.4	1.2	1.6×10 ¹	3.8×10 ⁻¹	3.2×10 ⁻¹	1.6×10 ²	2.8×10 ²	1.0×10 ⁻¹	1.8	9.9	1.4×10 ¹	3.7×10 ¹
Pu-241	3.8×10 ³	2.1×10 ²	6.0×10 ⁻⁴	1.4×10 ⁴	2.9×10 ²	6.3×10 ¹	7.0×10 ²	0	3.0×10 ¹	1.7×10 ³	2.6×10 ³	2.4×10 ¹	1.3×10 ²	4.2×10 ³	5.9×10 ³	3.5×10 ³
Pu-242	1.6×10 ⁻¹	9.2×10 ⁻⁴	3.8×10 ⁻¹¹	9.1×10 ⁻¹	3.0×10 ⁻³	9.9×10 ⁻⁴	1.6×10 ⁻²	0	4.2×10 ⁻⁴	1.6×10 ⁻³	2.0×10 ⁻²	2.3×10 ⁻⁴	2.5×10 ⁻⁴	5.7×10 ⁻²	9.0×10 ⁻²	7.0×10 ⁻²
Ra-226	4.6×10 ⁻⁶	2.2×10 ⁻¹²	6.5×10 ⁻¹⁰	2.6×10 ⁻⁷	2.0×10 ⁻⁸	3.8×10 ⁻¹⁰	2.3×10 ⁻¹⁰	4.9×10 ⁻⁶	9.3×10 ⁻⁷	2.3×10 ⁻⁹	5.3×10 ⁻⁹	4.5×10 ⁻⁵	2.3×10 ⁻¹²	5.6×10 ⁻⁶	6.3×10 ⁻⁶	4.1×10 ⁻⁹
Ra-228	3.7×10 ⁻¹⁰	1.2×10 ⁻¹³	4.0×10 ⁻⁹	1.3×10 ⁻⁴	1.1×10 ⁻⁵	7.3×10 ⁻¹³	1.1×10 ⁻¹²	6.5×10 ⁻³	2.4×10 ⁻³	6.9×10 ⁻¹³	2.0×10 ⁻¹¹	7.1×10 ⁻²	3.5×10 ⁻⁹	3.0×10 ⁻⁷	5.3×10 ⁻⁷	1.5×10 ⁻¹¹
Rh-102	--	--	--	--	--	--	--	--	--	--	--	--	--	1.1	1.5	--
Ru-106	3.1×10 ⁻⁵	6.3×10 ⁻⁷	3.1×10 ⁻¹⁵	3.9×10 ⁻⁷	1.2×10 ⁻⁶	1.3×10 ⁻⁵	4.2×10 ⁻⁵	3.2×10 ⁻⁹	3.0×10 ⁻¹⁵	2.6×10 ⁻⁶	3.1×10 ⁻⁸	2.2×10 ⁻¹⁰	1.5×10 ⁻⁹	4.2	7.1	5.7×10 ⁻⁵
Se-79	2.6×10 ⁻¹	3.0×10 ⁻²	1.7×10 ⁻³	1.9×10 ⁻¹	1.6×10 ⁻²	5.0×10 ⁻²	1.0×10 ⁻¹	2.9×10 ⁻²	1.4×10 ⁻²	5.2×10 ⁻²	3.6×10 ⁻³	2.5×10 ⁻¹	1.3×10 ⁻²	2.2	2.7	4.7×10 ⁻¹
Sm-151	3.3×10 ²	2.7×10 ¹	6.9	5.3×10 ²	2.5×10 ¹	4.2×10 ¹	3.4×10 ¹	4.5×10 ¹	2.6×10 ¹	1.8×10 ²	2.4×10 ²	9.1×10 ¹	2.4×10 ¹	1.2×10 ³	1.3×10 ³	3.8×10 ²
Sn-126	3.5×10 ⁻¹	2.6×10 ⁻²	3.8×10 ⁻³	2.4×10 ⁻¹	1.2×10 ⁻²	1.7×10 ⁻²	4.1×10 ⁻²	1.4×10 ⁻²	1.2×10 ⁻²	4.7×10 ⁻²	4.8×10 ⁻³	2.8×10 ⁻¹	1.2×10 ⁻²	1.9	2.4	3.3×10 ⁻¹
Sr-90	1.6×10 ⁴	7.1×10 ³	0	2.1×10 ⁴	3.2×10 ³	3.7×10 ³	7.6×10 ³	2.3×10 ³	1.8×10 ³	1.3×10 ⁴	1.6×10 ³	2.6×10 ³	2.9×10 ³	4.2×10 ⁵	5.2×10 ⁵	8.3×10 ⁴
Tc-99	7.7	9.9×10 ⁻¹	4.5×10 ⁻²	6.6	4.2×10 ⁻¹	1.0	2.2	7.4×10 ⁻¹	4.1×10 ⁻¹	1.8	1.3×10 ⁻¹	2.3	4.3×10 ⁻¹	6.7×10 ¹	8.2×10 ¹	1.4×10 ¹
Th-229	3.9×10 ⁻⁸	1.1×10 ⁻¹⁰	2.4×10 ⁻⁹	4.0×10 ⁻⁴	3.2×10 ⁻⁵	2.2×10 ⁻⁹	1.2×10 ⁻⁹	2.8×10 ⁻²	6.8×10 ⁻³	2.5×10 ⁻¹⁰	1.7×10 ⁻⁹	1.8×10 ⁻¹	1.2×10 ⁻⁹	6.1×10 ⁻⁶	9.9×10 ⁻⁶	8.7×10 ⁻⁹
Th-230	4.4×10 ⁻⁶	8.6×10 ⁻⁹	1.2×10 ⁻⁷	3.7×10 ⁻⁵	2.9×10 ⁻⁶	1.8×10 ⁻⁷	1.2×10 ⁻⁷	1.9×10 ⁻³	1.3×10 ⁻⁴	5.1×10 ⁻⁷	1.2×10 ⁻⁶	6.9×10 ⁻³	3.9×10 ⁻⁹	1.9×10 ⁻³	2.1×10 ⁻³	1.2×10 ⁻⁶
Th-232	5.1×10 ⁻¹⁰	2.0×10 ⁻¹²	4.3×10 ⁻⁹	1.4×10 ⁻⁴	1.2×10 ⁻⁵	1.9×10 ⁻¹¹	3.0×10 ⁻¹¹	5.1×10 ⁻³	2.5×10 ⁻³	4.4×10 ⁻¹²	5.5×10 ⁻¹¹	8.4×10 ⁻²	1.0×10 ⁻⁸	3.8×10 ⁻⁷	6.6×10 ⁻⁷	9.8×10 ⁻¹¹
U-232	9.9×10 ⁻⁵	3.5×10 ⁻⁵	1.9×10 ⁻⁶	0	2.2×10 ⁻⁵	1.7×10 ⁻⁴	1.4×10 ⁻⁴	2.3	2.4×10 ⁻¹	0	0	7.1×10 ²	2.4×10 ⁻⁵	3.2×10 ⁻¹	4.9×10 ⁻¹	3.5×10 ⁻⁴
U-233	2.5×10 ⁻⁵	9.1×10 ⁻⁷	9.9×10 ⁻⁷	1.6×10 ⁻¹	1.2×10 ⁻²	2.6×10 ⁻⁶	1.8×10 ⁻⁶	6.9	2.6	1.7×10 ⁻⁶	9.3×10 ⁻⁷	1.2×10 ²	5.6×10 ⁻⁶	1.8×10 ⁻³	3.0×10 ⁻³	1.6×10 ⁻⁵
U-234	2.0	8.6×10 ⁻⁴	5.0×10 ⁻⁴	1.7×10 ⁻¹	1.1×10 ⁻²	2.2×10 ⁻³	1.8×10 ⁻³	5.6×10 ⁻¹	4.4×10 ⁻¹	4.9×10 ⁻³	8.0×10 ⁻³	5.9	2.1×10 ⁻⁴	1.7×10 ¹	1.8×10 ¹	1.8×10 ⁻²
U-235	8.4×10 ⁻²	8.2×10 ⁻³	3.2×10 ⁻²	1.7×10 ⁻²	1.2×10 ⁻²	1.8×10 ⁻²	1.3×10 ⁻²	2.2×10 ⁻³	6.8×10 ⁻³	1.5×10 ⁻²	2.2×10 ⁻⁴	4.0×10 ⁻⁴	9.9×10 ⁻³	2.6×10 ⁻¹	2.5×10 ⁻¹	1.2×10 ⁻¹
U-236	3.3×10 ⁻¹	3.4×10 ⁻²	1.7	1.4×10 ⁻¹	1.2×10 ⁻²	3.7×10 ⁻²	5.9×10 ⁻²	2.1×10 ⁻²	1.7×10 ⁻²	6.0×10 ⁻²	4.1×10 ⁻³	8.1×10 ⁻⁴	1.3×10 ⁻²	3.3	4.0	4.4×10 ⁻¹
U-238	1.6	1.5×10 ⁻⁴	1.4×10 ⁻²	1.3×10 ⁻¹	3.4×10 ⁻²	8.9×10 ⁻⁴	1.6×10 ⁻²	5.4×10 ⁻⁵	7.1×10 ⁻⁵	2.7×10 ⁻⁴	2.7×10 ⁻³	1.3×10 ⁻⁵	5.8×10 ⁻³	1.1×10 ⁻³	1.2×10 ⁻³	2.4×10 ⁻²
Zr-93	1.0	1.5×10 ⁻¹	6.7×10 ⁻³	9.1×10 ⁻¹	5.0×10 ⁻²	1.0×10 ⁻¹	2.1×10 ⁻¹	1.1	6.4×10 ⁻²	2.7×10 ⁻¹	1.7×10 ⁻²	5.7×10 ⁻¹	7.8×10 ⁻²	1.8×10 ¹	2.7×10 ¹	1.9

a. Source: Dirkmaat (1998b, all); values are rounded to two significant figures.

b. INEEL = Idaho National Engineering and Environmental Laboratory; SRS = Savannah River Site.

c. Categories 1-13 and 16 decayed to 2030. Category 15 cooled for 5 years.

d. 15a = naval submarine fuel; 15b = naval surface ship fuel.

e. Includes 334 canisters from Fort St. Vrain.

f. -- = not found in appreciable quantities.

- **Category 4: Uranium oxide, intact.** The fuel in this category consists of uranium oxide that has been formed into pellets or plates and clad with a corrosion-resistant material. Commercial fuel is the representative fuel for this category because it is a large part of the inventory. The fuel is made of uranium oxide, some of which is highly enriched in uranium-235 and some of which is low enriched in uranium-235. The fuel elements are clad with a zirconium alloy.
- **Category 5: Uranium oxide, failed/declad/aluminum clad.** The fuel in this category is chemically similar to the fuels in Category 4, except accident or destructive examination has disrupted it. The failed fuel from Three Mile Island Reactor 2 represents this category because it comprises 96 percent of the total MTHM of the category. The Three Mile Island Reactor 2 fuel is melted uranium oxide. The accident greatly disrupted the cladding. Other fuel in this category is declad or has a large amount of cladding damage. Approximately 4 percent consists of intact aluminum clad fuel included in this category because the aluminum cladding is less corrosion resistant than Category 4 cladding material.
- **Category 6: Uranium-aluminide.** This category consists of fuel with a uranium-aluminum compound dispersed in a continuous aluminum metal phase. The fuel is clad with an aluminum alloy. The uranium-235 enrichment varies from 10 to 93 percent.
- **Category 7: Uranium-silicide.** The fuel in this category is a uranium-silicide compound dispersed in a continuous aluminum metal phase. The fuel is clad with an aluminum alloy. The uranium-235 enrichment varies from 8 to 93 percent, but most are less than 20 percent.
- **Category 8: Thorium/uranium carbide, high-integrity.** This category consists of fuels with thorium carbide or uranium carbide formed into particles with a high-integrity coating. Fort St. Vrain Reactor fuel represents the category because it makes up 95 percent of the mass of the category. This fuel is uranium carbide and thorium carbide formed into particles and coated with layers of pyrolytic carbon and silicon carbide. The particles are bonded in a carbonaceous matrix material and emplaced in a graphite block. The fuel was made with uranium enriched to 93 percent in uranium-235. The thorium was used to generate fissile uranium-233 during irradiation. Some fuel does not have a silicon carbide coating, but its effect on the category is very small. Less than 1 percent of the fuel particles are breached.
- **Category 9: Thorium/uranium carbide, low-integrity.** This category consists of fuels with uranium carbide or thorium carbide made into particles with a coating of an earlier design than that described for Category 8. Peach Bottom Unit 1, Core 1 is the only fuel in this category. This fuel is chemically similar to Category 8 fuel except 60 percent of the particle coating is breached. Peach Bottom Unit 1, Core 2 is included in Category 8 because its fuel particles are basically intact and are more rugged than the Peach Bottom Unit 1, Core 1 particles.
- **Category 10: Plutonium/uranium carbide, nongraphite.** This category consists of fuel that contains uranium carbide. Much of it also contains plutonium carbide. Fast Flux Test Facility carbide assemblies represent this category because they make up 70 percent of the category and contain both uranium and plutonium. The Fast Flux Test Facility carbide fuel was constructed from uncoated uranium and plutonium carbide spheres that were loaded directly into the fuel pins, or pressed into pellets that were loaded into the pins. The pins are clad with stainless steel.
- **Category 11: Mixed oxide.** This category consists of fuels constructed of both uranium oxide and plutonium oxide. The Fast Flux Test Facility mixed-oxide test assembly is the representative fuel because it comprises more than 80 percent of the category. The fuels are a combination of uranium oxide and plutonium oxide pressed into pellets and clad with stainless steel or a zirconium alloy. The

uranium-235 enrichment is low, but the fissile contribution of the plutonium raises the effective enrichment to 15 percent.

- **Category 12: Uranium/thorium oxide.** This category consists of fuels constructed of uranium oxide and thorium oxide. Shippingport light-water breeder reactor fuel is the representative fuel because it comprises more than 75 percent of the inventory. The Shippingport light-water breeder reactor fuel is made of uranium-233, and the irradiation of the thorium produces more uranium-233. The mixture is pressed into pellets and clad with a zirconium alloy.
- **Category 13: Uranium-zirconium hydride.** This category consists of fuels made of uranium-zirconium hydride. Training Research Isotopes-General Atomic fuels comprise more than 90 percent of the mass of this category. The fuel is made of uranium-zirconium hydride formed into rods and clad primarily with stainless steel or aluminum. The uranium is enriched as high as 90 percent in uranium-235, but most is less than 20 percent enriched.
- **Category 14: Sodium-bonded.** For purposes of analysis in this EIS, it is assumed that all Category 14 fuels would be treated during the proposed electrometallurgical treatment that would result in high-level radioactive waste. The chemical composition of the resulting high-level radioactive waste is described in Section A.2.3. Category 14 is included here for completeness.
- **Category 15: 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 (Beckett 1998, Attachment 2).

DOE plans to emplace approximately 300 canisters of naval spent nuclear fuel in the Yucca Mountain repository. There are several different designs for naval nuclear fuel, but all designs employ similar materials and mechanical arrangements. The total weight of the fuel assemblies in a canister of a typical submarine spent reactor fuel, which is representative of the chemical composition of naval spent nuclear fuel, would be 11,000 to 13,000 kilograms (24,000 to 29,000 pounds). Of this total, less than 500 kilograms (1,100 pounds) would be uranium. Approximately 1,000 to 2,000 kilograms (2,200 to 4,400 pounds) of the total weight of these fuel assemblies is from hafnium in the poison devices (primarily control rods) permanently affixed to the fuel assemblies (Beckett 1998, Attachment 2).

There would be approximately 9,000 to 12,000 kilograms (20,000 to 26,500 pounds) of zirconium alloy in the fuel structure in the typical canister. The typical chemical composition of zirconium alloy is approximately 98 percent zirconium, 1.5 percent tin, 0.2 percent iron, and 0.1 percent chromium (Beckett 1998, Attachment 2).

The small remainder of the fuel mass in a typical canister of naval submarine spent nuclear fuel [less than 500 kilograms (1,100 pounds)] would consist of small amounts of such metals and nonmetals as fission products and oxides (Beckett 1998, Attachment 2).

- **Category 16: Miscellaneous.** This category consists of the fuels that do not fit into the previous 15 categories. The largest amount of this fuel, as measured in MTHM, is uranium metal or alloy. The other two primary contributors are uranium alloy and uranium-thorium alloy. These three fuel types make up more than 80 percent of the MTHM in the category. It is conservative to treat the total category as uranium metal. Other chemical compounds included in this category include uranium

oxide, uranium nitride, uranium alloys, plutonium oxide, plutonium nitride, plutonium alloys, and thorium oxide.

Table A-19 lists the primary materials of construction and chemical composition for each category.

A.2.2.5.4 Thermal Output

Table A-20 lists the maximum heat generation per handling unit for each spent nuclear fuel category (Dirkmaat 1997a, Attachment, pages 74 to 77; Dirkmaat 1998b, all). The category 15 (naval fuel) thermal data used the best estimate radionuclide content from Dirkmaat (1997a, Attachment, pages 74 to 77) at a minimum cooling time of 5 years.

A.2.2.5.5 Quantity of Spent Nuclear Fuel Per Canister

Table A-21 lists the projected number of canisters required for each site and category. The amount of fuel per canister would vary widely among categories and would depend on a variety of parameters. The average mass of submarine spent nuclear fuel in a short naval dual-purpose canister would be approximately 13 metric tons (14 tons) with an associated volume of 2.7 cubic meters (95 cubic feet). Surface ship spent nuclear fuel in a long naval dual-purpose canister would have an average mass of approximately 18 metric tons (20 tons) and a volume of 3.5 cubic meters (124 cubic feet) (Dirkmaat 1997a, Attachment, pages 86 to 88).

A.2.2.5.6 Spent Nuclear Fuel Canister Parameters

The Idaho National Engineering and Environmental Laboratory would use a combination of 46- and 61-centimeter (18- and 24-inch)-diameter stainless-steel canisters for spent nuclear fuel disposition. The Savannah River Site would use 18-inch canisters, and Hanford would use 64-centimeter (25.3-inch) multiccanister overpacks and 18-inch canisters. Table A-21 lists the specific number of canisters per site. Detailed canister design specifications for the standard 18- and 24-inch canisters are contained in DOE (1998c, all). Specifications for the Hanford multiccanister overpacks are in Parsons (1999, all).

There are two conceptual dual-purpose canister designs for naval fuel: one with a length of 539 centimeters (212 inches) and one with a length of 475 centimeters (187 inches). Both canisters would have a maximum diameter of 169 centimeters (67 inches) (Dirkmaat 1997a, Attachment, pages 86 to 88). Table A-22 summarizes the preliminary design information.

For both designs, the shield plug, shear ring, and outer seal plate would be welded to the canister shell after the fuel baskets were loaded in the canister. The shield plug, shear ring, and welds, along with the canister shell and bottom plug, would form the containment boundary for the disposable container. The shell, inner cover, and outer cover material for the two canisters would be low-carbon austenitic stainless steel or stabilized austenitic stainless steel. Shield plug material for either canister would be stainless steel or another high-density material sheathed in stainless steel (Dirkmaat 1997a, Attachment, pages 86 to 88).

A.2.3 HIGH-LEVEL RADIOACTIVE WASTE

High-level radioactive waste is the highly radioactive material resulting from the reprocessing of spent nuclear fuel. DOE stores high-level radioactive waste at the Hanford Site, the Savannah River Site, and the Idaho National Engineering and Environmental Laboratory. Between 1966 and 1972, commercial chemical reprocessing operations at the Nuclear Fuel Services plant near West Valley, New York, generated a small amount of high-level radioactive waste at a site presently owned by the New York State

Table A-19. Chemical composition of DOE spent nuclear fuel by category (kilograms).^{a,b}

Fuel	Category															
	1	2	3	4	5	6	7	8	9	10	11	12	13	15	16	
Components																
Uranium	2,120,000	40	3,800	98,000	87,000	8,800	12,000	1,300	210	140	9,900	810	2,000	65,000	8,500	
Aluminum	1,700	(c)				18,000	4,200									
Molybdenum			380										9			
Zirconium	140	440		7,500									23,000			
Thorium								27,000	1,500			48,000			2,200	
Plutonium										16	2,400				8	
Silicon	260						880									
Silicon carbide								53,000								
Carbon	1,200			30				220,000	53,000				1,700			
Cladding and structure																
Aluminum	100		640		18,000	64,000	52,000						11,000		500	
Stainless steel				11,000	3,000				8,000	320	2,400	31,000	17,000		20,000	
Zirconium alloy	160,000	70	280	64,000	58,000						500	12,000	100	3,600,000	100	
Inconel				1,000	1,700											
Container																
Stainless steel	2,640,000	5,600	50,000	165,000	750,000	900,000	270,000	500,000	42,000	3,500	260,000	50,000	70,000	9,900,000	31,000	
Aluminum			660		10,000											
Other																
Concrete					30,000 ^d											
Boron									29							
Silver					1,100											
Cadmium					34											
Indium					280											
Magnesium									430							
Nickel	210															
Rhodium									30							
Ruthenium									30							
Samarium													67			
Gadolinium					530	950	23									
Hafnium														600,000		

a. Source: Dirkmaat (1998a, all); values are rounded to two significant figures.

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

c. Blanks indicate none or less than reportable quantities.

d. Low density converters were added to canisters of Three Mile Island Unit 2 fuel and would remain when shipped to the repository.

Table A-20. Maximum heat generation for DOE spent nuclear fuel (watts per handling unit).^{a,b}

Category and fuel type	Maximum heat generation
1. Uranium metal	18
2. Uranium zirconium	90
3. Uranium molybdenum	4
4. Intact uranium oxide	1,000
5. Failed/declad/aluminum clad uranium oxide	800
6. Uranium aluminide	480
7. Uranium silicide	1,400
8. High-integrity thorium/uranium carbide	250
9. Low-integrity thorium/uranium carbide	37
10. Nongraphite plutonium/uranium carbide	1,800
11. Mixed oxide	1,800
12. Thorium/uranium oxide	120
13. Uranium zirconium hydride	100
14. Sodium-bonded	N/A ^c
15. Naval fuel	4,250
16. Miscellaneous	1,000

- a. Sources: Dirkmaat (1997a, Attachment, pages 74 to 77; Dirkmaat 1998b, all).
 b. Handling unit is a canister or naval dual purpose canister.
 c. N/A = not applicable. Assumed to be treated and therefore part of high-level radioactive waste inventory (see Section A.2.2.1).

Table A-21. Required number of canisters for disposal of DOE spent nuclear fuel.^{a,b}

Category	Hanford		INEEL		SRS	Naval	
	18-inch	25.3-inch	18-inch	24-inch	18-inch	Short DPC ^c	Long DPC
1		440	6		9		
2			8				
3			70				
4	14	20	179	16			
5	1		406		425		
6					750		
7					225		
8			503 ^d				
9			60				
10	2		3				
11	324		43				
12			24	47			
13	3		97				
14 ^e							
15						200	100
16	5		39		2		
Totals	349	460	1,438	63	1,411	200	100

- a. Sources: Dirkmaat (1997b, Attachment, page 2); Dirkmaat (1998a, all).
 b. INEEL = Idaho National Engineering and Environmental Laboratory; SRS = Savannah River Site.
 c. Naval dual-purpose canister.
 d. Includes 334 canisters from Fort St. Vrain.
 e. Assumed to be treated and therefore part of high-level radioactive waste inventory (see Section A.2.2.1).

Energy Research and Development Authority. These operations ceased after 1972. In 1980, Congress passed the West Valley Demonstration Project Act, which authorizes DOE to conduct, with the Research and Development Authority, a demonstration of solidification of high-level radioactive waste for disposal and the decontamination and decommissioning of demonstration facilities (DOE 1992, Chapter 3). This

Table A-22. Preliminary naval dual-purpose canister design parameters.^a

Parameter	Short canister	Long canister
Maximum outside diameter (centimeters) ^{b,c}	169	169
Maximum outer length (centimeters)	475	539
Minimum loaded weight (metric tons) ^d	27	27
Maximum loaded weight (metric tons)	45	45

a. Source: Dirkmaat (1997a, Attachment, pages 86 to 88).

b. To convert centimeters to inches, multiply by 0.3937.

c. Right circular cylinder.

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

section addresses defense high-level radioactive waste generated at the DOE sites (Hanford Site, Idaho National Engineering and Environmental Laboratory, and Savannah River Site) and commercial high-level radioactive waste generated at the West Valley Demonstration Project.

A.2.3.1 Background

In 1985, DOE published a report in response to Section 8 of the Nuclear Waste Policy Act (of 1982) that required the Secretary of Energy to recommend to the President whether defense high-level radioactive waste should be disposed of in a geologic repository along with commercial spent nuclear fuel. That report, *An Evaluation of Commercial Repository Capacity for the Disposal of Defense High-Level Waste* (DOE 1985, all), provided the basis, in part, for the President's determination that defense high-level radioactive waste should be disposed of in a geologic repository. Given that determination, DOE decided to allocate 10 percent of the capacity of the first repository for the disposal of DOE spent nuclear fuel (2,333 MTHM) and high-level radioactive waste (4,667 MTHM) (Dreyfuss 1995, all; Lytle 1995, all).

Calculating the MTHM quantity for spent nuclear fuel is straightforward. It is determined by the actual heavy metal content of the spent fuel. However, an equivalence method for determining the MTHM in defense high-level radioactive waste is necessary because almost all of its heavy metal has been removed. A number of alternative methods for determining MTHM equivalence for high-level radioactive waste have been considered over the years. Four of those methods are described in the following paragraphs.

Historical Method. Table 1-1 of the 1985 DOE report provided a method to estimate the MTHM equivalence for high-level radioactive waste based on comparing the radioactive (curie) equivalence of commercial high-level radioactive waste and defense high-level radioactive waste. The method relies on the relative curie content of a hypothetical (in the early 1980s) canister of defense high-level radioactive waste from the Savannah River, Hanford, or Idaho site, and a hypothetical canister of vitrified waste from reprocessing of high-burnup commercial spent nuclear fuel. Based on commercial high-level radioactive waste containing 2.3 MTHM per canister (heavy metal has not been removed from commercial waste) and defense high-level radioactive waste estimated to contain approximately 22 percent of the radioactivity of a canister of commercial high-level radioactive waste, defense high-level radioactive waste was estimated to contain the equivalent of 0.5 MTHM per canister. Since 1985, DOE has used this 0.5 MTHM equivalence per canister of defense high-level radioactive waste in its consideration of the potential impacts of the disposal of defense high-level radioactive waste, including the analysis presented in this EIS. With this method, less than 50 percent of the total inventory of high-level radioactive waste could be disposed of in the repository within the 4,667 MTHM allocation for high-level radioactive waste. There has been no determination of which waste would be shipped to the repository, or the order of shipments.

Spent Nuclear Fuel Reprocessed Method. Another method of determining MTHM equivalence, based on the quantity of spent nuclear fuel reprocessed, would be to consider the MTHM in the high-level radioactive waste to be the same as the MTHM in the spent nuclear fuel before it was reprocessed. Using

this method, less than 5 percent of the total inventory of high-level radioactive waste could be disposed of in the repository within the 4,667 MTHM allocation for high-level radioactive waste.

Total Radioactivity Method. Another method, the total radioactivity method, would establish equivalence based on a comparison of radioactivity inventory (curies) of defense high-level radioactive waste to that of a standard MTHM of commercial spent nuclear fuel. For this equivalence method the standard spent nuclear fuel characteristics are based on pressurized-water reactor fuel with uranium-235 enrichment of 3.11 percent and 39.65 gigawatt-days per MTHM burnup. Using this method, 100 percent of the total inventory of high-level radioactive waste inventory could be disposed of in the repository within the 4,667 MTHM allocation for high-level radioactive waste.

Radiotoxicity Method. Yet another method, the radiotoxicity method, uses a comparison of the relative radiotoxicity of defense high-level radioactive waste to that of a standard MTHM of commercial spent nuclear fuel, and is thus considered an extension of the total radioactivity method. Radiotoxicity compares the inventory of specific radionuclides to a regulatory release limit for that radionuclide, and uses these relationships to develop an overall radiotoxicity index. For this equivalence, the standard spent nuclear fuel characteristics are based on pressurized-water reactor fuel with uranium-235 enrichment of 3.11 percent, 39.65 gigawatt-days per MTHM burnup. Using this method, 100 percent of the total inventory of high-level radioactive waste could be disposed of in the repository within the 4,667 MTHM allocation for high-level radioactive waste.

A recent report (Knecht et al. 1999, all) describes four equivalence calculation methods and notes that, under the Total Radioactivity Method or the Radiotoxicity Method, all DOE high-level radioactive waste could be disposed of under the Proposed Action. Using different equivalence methods would shift the proportion of high-level radioactive waste that could be disposed of between the Proposed Action and Inventory Module 1 analyzed in Chapter 8, but would not change the cumulative impacts analyzed in this EIS. Regardless of the equivalence method used, the EIS analyzes the impacts from disposal of the entire inventory of high-level radioactive waste in inventory Module 1.

A.2.3.2 Sources

A.2.3.2.1 Hanford Site

The Hanford high-level radioactive waste materials discussed in this EIS are those in the Tank Waste Remediation System Disposal Program and include tank waste, strontium capsules, and cesium capsules (Picha 1997, Table RL-1). DOE has not declared other miscellaneous materials or waste at Hanford, either existing or forecasted, to be candidate high-level radioactive waste streams. Before shipment to the repository, DOE would vitrify the high-level radioactive waste into a borosilicate glass matrix and pour it into stainless-steel canisters.

A.2.3.2.2 Idaho National Engineering and Environmental Laboratory

The Idaho National Engineering and Environmental Laboratory has proposed three different high-level radioactive waste stream matrices for disposal at the proposed Yucca Mountain Repository—glass, ceramic, and metal. The glass matrix waste stream would come from the Idaho Nuclear Technology and Engineering Center and would consist of wastes generated from the treatment of irradiated nuclear fuels. The Argonne National Laboratory-West proposed electrometallurgical treatment of DOE sodium-bonded fuels would generate both ceramic and metallic high-level radioactive waste matrices. DOE is preparing an EIS [DOE/EIS-0287 (Notice of Intent, 62 FR 49209, September 19, 1997)] to support decisions on managing the high-level radioactive waste at the Idaho Nuclear Technology and Engineering Center. DOE is preparing a separate EIS on managing sodium-bonded spent nuclear fuel at Argonne National

Laboratory-West and elsewhere, under which electrometallurgical treatment as well as alternative terminologies are being considered [DOE/EIS-0306 (Notice of Intent, 64 *FR* 8553, February 22, 1999)].

A.2.3.2.3 Savannah River Site

Savannah River Site high-level radioactive waste consists of wastes generated from the treatment of irradiated nuclear fuels. These wastes include various chemicals, radionuclides, and fission products that DOE maintains in liquid, sludge, and saltcake forms. The Defense Waste Processing Facility at the Savannah River Site mixes the high-level radioactive waste with glass-forming materials, converts it to a durable borosilicate glass waste form, pours it into stainless-steel canisters, and seals the canisters with welded closure plugs (Picha 1997, Attachment 4, page 2).

Another source of high-level radioactive waste at the Savannah River Site is the immobilized plutonium addressed in Section A.2.4.

A.2.3.2.4 West Valley Demonstration Project

The West Valley Demonstration Project is responsible for solidifying high-level radioactive waste that remains from the commercial spent nuclear fuel reprocessing plant operated by Nuclear Fuel Services. The Project mixes the high-level radioactive waste with glass-forming materials, converts it to a durable borosilicate glass waste form, pours it into stainless-steel canisters, and seals the canisters with welded closure plugs.

A.2.3.3 Present Status

A.2.3.3.1 Hanford Site

The Hanford Site stores high-level radioactive waste in underground carbon-steel tanks. This analysis assumed that before vitrification, strontium and cesium capsules currently stored in water basins at Hanford would be blended with the liquid high-level radioactive waste. To date, Hanford has immobilized no high-level radioactive waste. Before shipping waste to a repository, DOE would vitrify it into an acceptable glass form. DOE has scheduled vitrification to begin in 2007 with an estimated completion in 2028.

A.2.3.3.2 Idaho National Engineering and Environmental Laboratory

Most of the high-level radioactive waste at the Idaho Nuclear Technology and Engineering Center (formerly the Idaho Chemical Processing Plant) is in calcined solids (calcine) stored at the Idaho National Engineering and Environmental Laboratory. The calcine, an interim waste form, is in stainless-steel bins in concrete vaults. Before shipment to a repository, DOE proposes to immobilize the high-level radioactive waste in a vitrified (glass) waste form. The Idaho Nuclear Technology and Engineering Center proposes to implement its vitrification program in 2020 and complete it in 2035 (LMIT 1998, pages A-39 to A-42).

As discussed in Section A.2.2.1, DOE is evaluating treatment of sodium-bonded fuels at Argonne National Laboratory-West. If electrometallurgical treatment were to be chosen, DOE would stabilize the high-level radioactive waste generated from the treatment of its sodium-bonded fuel in the Fuel Conditioning Facility and Hot Fuel Examination Facility into ceramic and metal waste forms in the same facilities. The Radioactive Scrap and Waste Facility at Argonne National Laboratory-West would provide interim storage for these waste forms. There are several technologies being considered for waste

treatment (for example, electrometallurgical treatment, melt and dilute, Purex). If a decision was made to implement this proposal, DOE would begin stabilization in 2000.

A.2.3.3.3 Savannah River Site

DOE stores high-level radioactive waste in underground tanks in the F- and H-Areas at the Savannah River Site. High-level radioactive waste that has been converted to a borosilicate glass form is stored in the Glass Waste Storage Building in the S-Area. DOE projects completion of the vitrification of the stored high-level radioactive waste by 2022 (Davis and Wells 1997, all).

A.2.3.3.4 West Valley Demonstration Project

High-level radioactive waste is stored in underground tanks at the West Valley site. High-level radioactive waste that has been converted into a borosilicate glass waste form is stored in the converted Chemical Process Cell in the Process Building, referred to as the Interim High-Level Radioactive Waste Storage Facility. West Valley plans to complete its vitrification program by the Fall of 2002 (DOE 1992, Chapter 3).

A.2.3.4 Final Waste Form

The final waste form for high-level radioactive waste from the Hanford Site, Savannah River Site, Idaho Nuclear Technology and Engineering Center, and West Valley Demonstration Project would be a vitrified glass matrix in a stainless-steel canister.

The waste forms from Argonne National Laboratory-West could be ceramic and metallic waste matrices depending on decisions to be based on an ongoing EIS. These could be in stainless-steel canisters similar to those used for Savannah River Site and Idaho Nuclear Technology and Engineering Center glass wastes.

A.2.3.5 Waste Characteristics

A.2.3.5.1 Mass and Volume

Hanford Site. The estimated volume of borosilicate glass generated by high-level radioactive waste disposal actions at Hanford will be 15,700 cubic meters (554,000 cubic feet); the estimated mass of the glass is 44,000 metric tons (48,500 tons) (Picha 1998a, Attachment 1). The volume calculation assumes that strontium and cesium compounds from capsules currently stored in water basins would be blended with tank wastes before vitrification with no increase in product volume. This volume of glass would require 14,500 canisters, nominally 4.5 meters (15 feet) long with a 0.61-meter (2-foot) diameter (Picha 1998a, Attachment 1).

Idaho National Engineering and Environmental Laboratory. Table A-23 lists the volumes, masses, densities, and estimated number of canisters for the three proposed waste streams.

Savannah River Site. Based on Revision 8 of the High-Level Waste System Plan (Davis and Wells 1997, all), the Savannah River Site would generate an estimated 5,978 canisters of high-level radioactive waste (Picha 1997, Attachment 1). The canisters have a nominal outside diameter of 0.61 meter (2 feet) and a nominal height of 3 meters (10 feet). They would contain a total of approximately 4,240 cubic meters (150,000 cubic feet) of glass. The estimated total mass of high-level radioactive waste for repository disposal would be 11,600 metric tons (12,800 tons) (Picha 1997, Attachment 1). Section A.2.4.5.2.1 addresses the additional high-level radioactive waste canisters that DOE would generate at the

Table A-23. Physical characteristics of high-level radioactive waste at the Idaho National Engineering and Environmental Laboratory.^{a,b}

Physical quantities	INTEC glass matrix	ANL-W ceramic matrix	ANL-W metal matrix
Volume (cubic meters) ^c	743	60.0	1.2
Mass (kilograms) ^d	1,860,000	144,000	9,000
Density (kilograms per cubic meter)	2,500	2,400	7,750
Number of canisters [range] ^e	1,190	96 [80 - 125]	6 [2 - 10]

- a. Sources: Picha (1997, Attachment 1); Goff (1998a, all); Goff (1998b, all).
- b. INTEC = Idaho Nuclear Technology and Engineering Center; ANL-W = Argonne National Laboratory-West.
- c. To convert cubic meters to cubic yards, multiply by 1.3079.
- d. To convert kilograms to pounds, multiply by 2.2046.
- e. Canister would be nominally 3 meters (10 feet) by 0.6 meter (2 feet). Canisters would be filled to approximately 0.625 cubic meter (22 cubic feet).

Savannah River Site as a result of immobilizing surplus plutonium. As discussed in that section, 77 additional canisters would be required if the assumed 18 metric tons (20 tons) of plutonium is immobilized. If the entire 50 metric tons (55 tons) of surplus plutonium was immobilized, 210 additional high-level radioactive waste canisters would be required.

West Valley Demonstration Project. The West Valley Demonstration Project will generate between 260 and 300 canisters of high-level radioactive waste. The canisters have a nominal outside diameter of 0.61 meter (2 feet) and a nominal height of 3 meters (10 feet) (Picha 1997, Attachment 1). They will contain approximately 200 cubic meters (7,060 cubic feet) of glass. The estimated total mass of this high-level radioactive waste will be between 540 and 630 metric tons (595 and 694 tons) (Picha 1998c, page 3).

Summary. Table A-24 summarizes the information in the previous paragraphs to provide the total mass and volume projected to be disposed of at the repository.

Table A-24. High-level radioactive waste mass and volume summary.

Parameter	Total ^{a,b}
Mass	58,000 metric tons
Volume	21,000 cubic meters
Number of canisters	22,147 - 22,280 ^c

- a. Sources: Picha (1997, Attachment 1); Picha (1998a, Attachment 1).
- b. To convert metric tons to tons, multiply by 1.1023; to convert cubic meters to cubic yards, multiply by 1.3079.
- c. The number of canisters depends on the amount of surplus weapons-usable plutonium immobilized (see Section A.2.4.5.2.1).

A.2.3.5.2 Amount and Nature of Radioactivity

The following paragraphs present radionuclide inventory information for the individual sites. They present the best available data at varying dates; however, in most cases, the data are conservative because the inventories are for dates earlier than the date of disposal, and additional radioactive decay would occur before disposal. Any differences due to varying amounts of radioactive decay are small.

Hanford Site. Table A-25 lists the estimated radionuclide inventory for Hanford high-level radioactive glass waste, including strontium-90 and cesium-137 currently stored in capsules (Picha 1997, Table RL-1). With the exception of hydrogen-3 and carbon-14, this table makes the conservative assumption that 100 percent of a radionuclide in Hanford's 177 tanks and existing capsules is vitrified. Consistent with Hanford modeling for the Integrated Data Base (DOE 1997b, page 2-24), pretreatment and vitrification would separate hydrogen-3 and carbon-14 from the high-level radioactive waste stream such

Table A-25. Radionuclide distribution for Hanford Site high-level radioactive waste.^{a,b}

Radionuclide	Total curies	Curies per canister	Radionuclide	Total curies	Curies per canister
Hydrogen-3	-- ^c	--	Thorium-229	1.8	1.3×10 ⁻⁴
Carbon-14	9.6×10 ⁻²	6.6×10 ⁻⁶	Thorium-230	--	--
Chlorine-36	--	--	Thorium-232	2.1	1.5×10 ⁻⁴
Nickel-59	9.3×10 ²	6.4×10 ⁻²	Protactinium-231	1.6×10 ²	1.1×10 ⁻²
Nickel-63	9.2×10 ⁴	6.3	Uranium-232	1.2×10 ²	8.5×10 ⁻³
Cobalt-60	1.2×10 ⁴	8.5×10 ⁻¹	Uranium-233	4.8×10 ²	3.3×10 ⁻²
Selenium-79	7.7×10 ²	5.3×10 ⁻²	Uranium-234	3.5×10 ²	2.4×10 ⁻²
Krypton-85	--	--	Uranium-235	1.5×10 ¹	1.0×10 ⁻³
Strontium-90	9.7×10 ⁷	6.7×10 ³	Uranium-236	9.6	6.6×10 ⁻⁴
Niobium-93m	2.7×10 ³	1.9×10 ⁻¹	Uranium-238	3.2×10 ²	2.2×10 ⁻²
Niobium-94	--	--	Neptunium-237	1.4×10 ²	9.7×10 ⁻³
Zirconium-93	3.6×10 ³	2.5×10 ¹	Plutonium-238	2.8×10 ³	1.9×10 ⁻¹
Technetium-99	3.3×10 ⁴	2.3	Plutonium-239	3.9×10 ⁴	2.7
Rhodium-101	--	--	Plutonium-240	8.9×10 ³	6.2×10 ⁻¹
Rhodium-102	--	--	Plutonium-241	2.3×10 ⁵	1.6×10 ¹
Ruthenium-106	1.0×10 ⁵	7.2	Plutonium-242	1.2	8.0×10 ⁻⁵
Palladium-107	--	--	Americium-241	7.0×10 ⁴	4.8
Tin-126	1.2×10 ³	8.2×10 ⁻²	Americium-242m	--	--
Iodine-129	3.2×10 ¹	2.2×10 ⁻³	Americium-243	9.3	6.4×10 ⁻⁴
Cesium-134	8.9×10 ⁴	6.1	Curium-242	7.7×10 ¹	5.3×10 ⁻³
Cesium-135	--	--	Curium-243	1.0×10 ¹	6.9×10 ⁻⁴
Cesium-137	1.1×10 ⁸	7.7×10 ³	Curium-244	2.4×10 ²	1.7×10 ⁻²
Samarium-151	2.8×10 ⁶	1.9×10 ²	Curium-245	--	--
Lead-210	--	--	Curium-246	--	--
Radium-226	6.3×10 ⁻²	4.4×10 ⁻⁶	Curium-247	--	--
Radium-228	7.7×10 ¹	5.3×10 ⁻³	Curium-248	--	--
Actinium-227	8.8×10 ¹	6.0×10 ⁻³	Californium-252	--	--

a. Sources: Picha (1997, Table RL-1); Picha (1998a, Attachment 1).

b. Decayed to January 1, 1994.

c. -- = not found in appreciable quantities.

that essentially 0.0 percent and 0.002 percent of each, respectively, would be present in the glass. A large portion of iodine-129 could also be separated, but the analysis assumed a conservative 50-percent retention (Picha 1998a, Attachment 1). Table A-25 uses the estimated number of canisters (14,500) to develop the curies-per-canister value.

Idaho National Engineering and Environmental Laboratory. Table A-26 contains a baseline radionuclide distribution for the three Idaho National Engineering and Environmental Laboratory high-level radioactive waste streams. For each waste stream, the total radionuclide inventory is provided, as is the worst-case value for curies per canister. For Idaho Nuclear Technology and Engineering Center glass, the calculated inventories are decayed to 2035. For Argonne National Laboratory-West waste matrices, the calculated inventories are decayed to 2000.

Savannah River Site. The Waste Qualification Report details the projected radionuclide distribution in the high-level radioactive waste from the Savannah River Site (Plodinec and Marra 1994, page 10). Table A-27 lists the quantities of individual radionuclides in 2015, the expected time of shipment (Pearson 1998, all). The curie-per-canister values were obtained by dividing the total radionuclide projection by the expected number of canisters (5,978).

West Valley Demonstration Project. DOE used the ORIGEN2 computer code to estimate the radionuclide inventory for the West Valley Demonstration Project, simulating each Nuclear Fuel Services

Table A-26. Radionuclide distribution for Idaho National Engineering and Environmental Laboratory high-level radioactive waste.^{a,b}

Radionuclides	INTEC glass		ANL-W ceramic ^c		ANL-W metal ^c	
	Total curies for 2035	Curies per canister ^d	Total curies for 2000	Curies per canister ^d	Total curies for 2000	Curies per canister ^d
Hydrogen-3	3.6×10 ³	4.3	-- ^e	--	--	--
Carbon-14	2.8×10 ⁻²	8.3×10 ⁻⁵	--	--	4.3	4.3
Chlorine-36	--	--	--	--	--	--
Cobalt-60	3.2×10 ¹	3.6×10 ⁻²	--	--	3.2×10 ³	3.2×10 ³
Nickel-59	--	--	--	--	1.1×10 ¹	1.1×10 ¹
Nickel-63	--	--	--	--	4.1×10 ²	3.9×10 ²
Selenium-79	--	--	--	--	--	--
Krypton-85	--	--	--	--	--	--
Strontium-90	7.0×10 ⁶	1.2×10 ⁴	7.1×10 ⁵	4.7×10 ⁴	--	--
Niobium-93	4.7×10 ²	1.4	--	--	2.9×10 ¹	2.9×10 ¹
Niobium-94	5.4×10 ⁻³	1.6×10 ⁻⁵	--	--	2.7	2.7
Zirconium-93	--	--	--	--	--	--
Technetium-99	3.4×10 ³	9.9	--	--	1.3×10 ²	1.3×10 ²
Rhodium-101	--	--	--	--	--	--
Rhodium-102	2.0×10 ⁻⁵	2.2×10 ⁻⁸	--	--	--	--
Ruthenium-106	1.0×10 ⁻⁹	8.7×10 ⁻¹³	--	--	2.1×10 ⁴	2.1×10 ⁴
Palladium-107	--	--	--	--	--	--
Tin-126	8.9×10 ¹	2.6×10 ⁻¹	--	--	2.8	2.1
Iodine-129	5.6	1.7×10 ⁻²	3.4×10 ⁻¹	1.8×10 ⁻²	--	--
Cesium-134	3.3×10 ⁻²	3.6×10 ⁻⁵	7.9×10 ³	5.1×10 ²	--	--
Cesium-135	1.6×10 ²	2.5×10 ⁻¹	1.6×10 ¹	8.8×10 ⁻¹	--	--
Cesium-137	6.0×10 ⁶	1.2×10 ⁴	8.5×10 ⁵	5.3×10 ⁴	--	--
Samarium-151	--	--	--	--	--	--
Lead-210	--	--	--	--	--	--
Radium-226	9.7×10 ⁻³	7.2×10 ⁻⁵	3.0×10 ⁻⁵	2.1×10 ⁻⁶	--	--
Radium-228	--	--	--	--	--	--
Actinium-227	--	--	--	--	--	--
Thorium-229	--	--	--	--	--	--
Thorium-230	4.0×10 ⁻¹	2.8×10 ⁻³	4.7×10 ⁻³	8.9×10 ⁻⁴	--	--
Thorium-232	9.9×10 ⁻⁸	5.0×10 ⁻¹⁰	2.3×10 ⁻⁹	1.3×10 ⁻¹⁰	--	--
Protactinium-231	--	--	--	--	--	--
Uranium-232	4.6×10 ⁻³	5.2×10 ⁻⁶	2.6×10 ⁻³	1.8×10 ⁻⁴	1.2×10 ⁻⁴	1.2×10 ⁻⁴
Uranium-233	1.3×10 ⁻³	6.1×10 ⁻⁶	2.0×10 ⁻⁴	1.4×10 ⁻⁵	5.8×10 ⁻⁵	5.8×10 ⁻⁵
Uranium-234	1.0×10 ²	1.1×10 ⁻¹	2.8	1.9×10 ⁻¹	7.7×10 ⁻¹	7.7×10 ⁻¹
Uranium-235	5.9×10 ⁻¹	6.6×10 ⁻⁴	8.8×10 ⁻²	5.9×10 ⁻³	2.5×10 ⁻²	2.5×10 ⁻²
Uranium-236	1.5	1.7×10 ⁻³	6.3×10 ⁻²	4.2×10 ⁻³	1.8×10 ⁻²	1.8×10 ⁻²
Uranium-238	2.9×10 ⁻²	3.3×10 ⁻⁵	2.8×10 ⁻¹	4.9×10 ⁻³	9.7×10 ⁻²	8.8×10 ⁻²
Neptunium-237	6.3	2.8×10 ⁻²	1.3	5.8×10 ⁻²	2.4×10 ⁻⁵	2.3×10 ⁻⁵
Plutonium-238	9.0×10 ⁴	1.0×10 ²	3.6×10 ²	2.9×10 ¹	6.6×10 ⁻³	6.6×10 ⁻³
Plutonium-239	1.8×10 ³	2.0	1.7×10 ⁴	8.1×10 ²	3.3×10 ⁻¹	3.3×10 ⁻¹
Plutonium-240	1.6×10 ³	1.8	1.5×10 ³	6.9×10 ¹	2.9×10 ⁻²	2.9×10 ⁻²
Plutonium-241	1.9×10 ⁴	2.2×10 ¹	1.1×10 ⁴	1.3×10 ³	1.9×10 ⁻¹	1.9×10 ⁻¹
Plutonium-242	3.4	3.8×10 ⁻³	1.2×10 ⁻¹	2.3×10 ⁻²	2.0×10 ⁻⁶	2.0×10 ⁻⁶
Americium-241	1.3×10 ⁴	1.4×10 ¹	1.6×10 ³	3.4×10 ¹	3.1×10 ⁻²	2.1×10 ⁻²
Americium-242/242m	1.5×10 ⁻²	9.4×10 ⁻⁵	1.4×10 ¹	2.1×10 ⁻¹	2.7×10 ⁻⁴	2.1×10 ⁻⁴
Americium-243	1.4×10 ⁻²	1.1×10 ⁻⁴	2.8×10 ⁻¹	1.9×10 ⁻²	4.8×10 ⁻⁶	4.8×10 ⁻⁶
Curium-242	1.2×10 ⁻²	7.7×10 ⁻⁵	1.2×10 ¹	1.8×10 ⁻¹	2.3×10 ⁻⁴	1.8×10 ⁻⁴
Curium-243	4.7×10 ⁻⁴	3.4×10 ⁻⁶	1.6×10 ⁻¹	3.1×10 ⁻³	3.0×10 ⁻⁶	2.1×10 ⁻⁶
Curium-244	1.0×10 ⁻²	7.7×10 ⁻⁵	1.9	1.3×10 ⁻¹	3.1×10 ⁻⁵	3.1×10 ⁻⁵
Curium-245	3.7×10 ⁻⁶	2.8×10 ⁻⁸	6.8×10 ⁻⁵	4.7×10 ⁻⁶	1.1×10 ⁻⁹	1.1×10 ⁻⁹
Curium-246	8.7×10 ⁻⁸	6.6×10 ⁻¹⁰	4.2×10 ⁻⁷	2.9×10 ⁻⁸	7.1×10 ⁻¹²	7.1×10 ⁻¹²
Curium-247	3.1×10 ⁻¹⁴	2.4×10 ⁻¹⁶	2.4×10 ⁻¹³	1.6×10 ⁻¹⁴	4.0×10 ⁻¹⁸	4.0×10 ⁻¹⁸
Curium-248	9.4×10 ⁻¹⁵	7.2×10 ⁻¹⁷	2.6×10 ⁻¹⁴	1.8×10 ⁻¹⁵	4.4×10 ⁻¹⁹	4.4×10 ⁻¹⁹
Californium-252	--	--	6.5×10 ⁻¹⁹	1.6×10 ⁻¹⁹	--	--

a. Sources: Picha (1997, Table ID-2); Goff (1998a, all).

b. INTEC = Idaho Nuclear Technology and Engineering Center; ANL-W = Argonne National Laboratory-West.

c. Matrices based on treating all sodium-bonded fuels. Waste input streams and associated radioactivity for 2000 averaged for total number of canisters produced. Curie values based on calculated data from stored material.

d. Curie per canister values were provided as worst case rather than a homogenous mixture.

e. -- = not found in appreciable quantities.

Table A-27. Radionuclide distribution for Savannah River Site high-level radioactive waste (2015).^a

Radionuclide	Total (curies)	Curies per canister	Radionuclide	Total (curies)	Curies per canister
Hydrogen-3	-- ^b	--	Thorium-229	--	--
Carbon-14	--	--	Thorium-230	2.4×10^{-2}	4.0×10^{-6}
Chlorine-36	--	--	Thorium-232	--	--
Nickel-59	1.1×10^2	1.8×10^{-2}	Protactinium-231	--	--
Nickel-63	1.2×10^4	2.1	Uranium-232	--	--
Cobalt-60 ^c	--	4.5×10^1	Uranium-233	--	--
Selenium-79	1.1×10^3	1.8×10^{-1}	Uranium-234	1.6×10^2	2.7×10^{-2}
Krypton-85	--	--	Uranium-235	--	--
Strontium-90	1.7×10^8	2.9×10^4	Uranium-236	--	--
Niobium-93m	1.3×10^4	2.2	Uranium-238	5.0×10^1	8.3×10^{-3}
Niobium-94	--	--	Neptunium-237	4.1×10^2	6.8×10^{-2}
Zirconium-93	3.0×10^4	5.0	Plutonium-238	3.0×10^6	5.0×10^2
Technetium-99	1.5×10^4	2.5	Plutonium-239	3.7×10^4	6.2
Rhodium-101	--	--	Plutonium-240	2.5×10^4	4.1
Rhodium-102	--	--	Plutonium-241	3.3×10^6	5.4×10^2
Ruthenium-106 ^c	--	2.4	Plutonium-242	3.5×10^1	5.8×10^{-3}
Palladium-107	7.3×10^1	1.2×10^{-2}	Americium-241	1.6×10^5	2.6×10^1
Tin-126	2.6×10^3	4.3×10^{-1}	Americium-242m	--	--
Iodine-129	--	--	Americium-243	1.1×10^3	1.8×10^{-1}
Cesium-134 ^c	--	1.2×10^1	Curium-242	--	--
Cesium-135	4.0×10^2	6.7×10^{-2}	Curium-243	--	--
Cesium-137	1.5×10^8	2.4×10^4	Curium-244	4.9×10^5	8.3×10^1
Samarium-151	3.3×10^6	5.5×10^2	Curium-245	--	--
Lead-210	--	--	Curium-246	--	--
Radium-226	--	--	Curium-247	--	--
Radium-228	--	--	Curium-248	--	--
Actinium-227	--	--	Californium-252	--	--

a. Sources: Plodinec and Marra (1994, page 10); Pearson (1998, all).

b. -- = not found in appreciable quantities.

c. Total curie content not provided for these nuclides; curie per canister values provided for 10 years after production.

irradiated fuel campaign. A detailed description of the development of these estimates is in the West Valley Demonstration Project Waste Qualification Report (WVNS 1996, WQR-1.2, Appendix 1). Table A-28 lists the estimated activity by nuclide and provides the total curies, as well as the curies per canister, based on 260 canisters.

A.2.3.5.3 Chemical Composition

Hanford Site. The Integrated Data Base (DOE 1997b, page 2-29) provides the best available information for the proposed representative chemical composition of future high-level radioactive waste glass from Hanford. Table A-29 combines the percentages by weight of chemical constituents obtained from the Integrated Data Base with the estimated mass to present the expected chemical composition of the glass in terms of mass per chemical compound.

Idaho National Engineering and Environmental Laboratory

Idaho Nuclear Technology and Engineering Center Glass Matrix. This waste stream is composed of three primary sources—zirconium calcine, aluminum calcine, and sodium-bearing waste.

The distribution of these sources is 55 percent, 15 percent, and 30 percent, respectively (Heiser 1998, all). Table A-30 lists the chemical composition of the total waste stream.

Table A-28. Radionuclide distribution for West Valley Demonstration Project high-level radioactive waste (2015).^a

Radionuclide	Total curies	Curies per canister	Radionuclide	Total curies	Curies per canister
Hydrogen-3	2.0×10 ¹	7.8×10 ⁻²	Thorium-229	2.3×10 ⁻¹	8.9×10 ⁻⁴
Carbon-14	1.4×10 ²	5.3×10 ⁻¹	Thorium-230	6.0×10 ⁻²	2.3×10 ⁻⁴
Chlorine-36	-- ^b	--	Thorium-232	1.6	6.3×10 ⁻³
Nickel-59	1.1×10 ²	4.1×10 ⁻¹	Protactinium-231	1.5×10 ¹	5.9×10 ⁻²
Nickel-63	7.1×10 ³	2.7×10 ¹	Uranium-232	5.9	2.3×10 ⁻²
Cobalt-60	2.9×10 ¹	1.1×10 ⁻¹	Uranium-233	9.5	3.7×10 ⁻²
Selenium-79	6.0×10 ¹	2.3×10 ⁻¹	Uranium-234	5.0	1.9×10 ⁻²
Krypton-85	--	--	Uranium-235	1.0×10 ⁻¹	3.9×10 ⁻⁴
Strontium-90	3.7×10 ⁶	1.4×10 ⁴	Uranium-236	3.0×10 ⁻¹	1.1×10 ⁻³
Niobium-93m	2.5×10 ²	9.5×10 ⁻¹	Uranium-238	8.5×10 ⁻¹	3.3×10 ⁻³
Niobium-94	--	--	Neptunium-237	2.4×10 ¹	9.2×10 ⁻²
Zirconium-93	2.7×10 ²	1.1	Plutonium-238	7.0×10 ³	2.7×10 ¹
Technetium-99	1.7×10 ³	6.5	Plutonium-239	1.7×10 ³	6.4
Rhodium-101	--	--	Plutonium-240	1.2×10 ³	4.7
Rhodium-102	--	--	Plutonium-241	2.5×10 ⁴	9.5×10 ¹
Ruthenium-106	5.0×10 ⁻⁷	1.9×10 ⁻⁹	Plutonium-242	1.7	6.4×10 ⁻³
Palladium-107	1.1×10 ¹	4.2×10 ⁻²	Americium-241	5.3×10 ⁴	2.0×10 ²
Tin-126	1.0×10 ²	4.0×10 ⁻¹	Americium-242m	2.7×10 ²	1.0
Iodine-129	2.1×10 ⁻¹	8.1×10 ⁻⁴	Americium-243	3.5×10 ²	1.3
Cesium-134	1.2	4.4×10 ⁻³	Curium-242	2.2×10 ²	8.4×10 ⁻¹
Cesium-135	1.6×10 ²	6.2×10 ⁻¹	Curium-243	7.3×10 ¹	2.8×10 ⁻¹
Cesium-137	4.1×10 ⁶	1.6×10 ⁴	Curium-244	2.9×10 ³	1.1×10 ¹
Samarium-151	7.0×10 ⁴	2.7×10 ²	Curium-245	8.8×10 ⁻¹	3.4×10 ⁻³
Lead-210	--	--	Curium-246	1.0×10 ⁻¹	3.9×10 ⁻⁴
Radium-226	--	--	Curium-247	--	--
Radium-228	1.6	6.3×10 ⁻³	Curium-248	--	--
Actinium-227	1.2×10 ¹	4.6×10 ⁻²	Californium-252	--	--

a. Source: WVNS (1996, WQR-1.2, Appendix 1).

b. -- = not found in appreciable quantities.

Table A-29. Expected chemical composition of Hanford high-level radioactive waste glass (kilograms).^{a,b}

Compound	Mass	Compound	Mass
Aluminum oxide	4,100,000	Sodium oxide	5,190,000
Boron oxide	3,090,000	Sodium sulfate	44,000
Bismuth trioxide	510,000	Nickel monoxide	480,000
Calcium oxide	370,000	Phosphorous pentaoxide	690,000
Ceric oxide	500,000	Lead monoxide	62,000
Chromic oxide	160,000	Silicon oxide	20,300,000
Ferric oxide	1,980,000	Strontium oxide	79,000
Potassium oxide	75,000	Thorium dioxide	4,400
Lanthanum oxide	48,000	Uranium oxide	2,940,000
Lithium oxide	880,000	Zirconium dioxide	1,630,000
Manganese dioxide	510,000	Other	75,000
Sodium fluoride	280,000	Total	44,000,000

a. Sources: DOE (1997b, page 2-29); Picha (1998a, Attachment 1).

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

Argonne National Laboratory-West Ceramic and Metal Matrices. Electrometallurgical processing of DOE spent nuclear fuel containing thermal-bond sodium would result in two high-level radioactive waste forms for repository disposal, depending on decisions to be based on an going EIS [DOE/EIS-0306

Table A-30. Expected glass matrix chemical composition at Idaho Nuclear Technology and Engineering Center (kilograms).^{a,b}

Compound or element	Mass	Compound or element	Mass
Aluminum oxide	130,000	Silicon oxide	1,020,000
Ammoniummolybdophosphate	26,000	Zirconium dioxide	18,000
Boron oxide	200,000	Arsenic	100
Calcium fluoride	140,000	Cadmium	42,000
Calcium oxide	4,100	Chromium	14,000
Ceric oxide	300	Mercury ^c	200
Ferric oxide	800	Nickel	1,400
Sodium oxide	250,000	Lead	1,800
Phosphorous pentaoxide	1,000	Total^d	1,860,000

- a. Sources: Picha (1997, Table ID-3); Heiser (1998, all).
- b. Masses are rounded to the nearest 100 kilograms; to convert kilograms to pounds, multiply by 2.2046.
- c. Assumes only 0.1 percent capture of original mercury in the feed materials.
- d. Trace amounts of antimony, beryllium, barium, selenium, silver, and thallium were also reported.

(Notice of Intent, 64 *FR* 8553, February 22, 1999)]. The first form would be a glass-bonded ceramic composite.

It would stabilize the alkali, alkaline earth, lanthanide, halide, and transuranic materials in processed spent nuclear fuel. These elements would be present as halides after fuel treatment. For disposal, these compounds would be stabilized in a zeolite-based material (Goff 1998a, all).

The chemical formula for zeolite-4A, the typical starting material, is $\text{Na}_{12}[(\text{AlO}_2)_{12}(\text{SiO}_2)_{12}]$. In the waste form, zeolite would contain approximately 10 to 12 percent of the halide compounds by weight. The zeolite mixture typically would be combined with 25-percent glass frit by weight, placed in a stainless-steel container, and processed into a solid monolith using a hot isostatic press. The zeolite would convert to the mineral sodalite in the process (Goff 1998a, all). Table A-31 lists the composition of the waste form.

Table A-31. Expected ceramic waste matrix chemical composition at Argonne National Laboratory-West (kilograms).^{a,b}

Component	Mass	Component	Mass
Zeolite-4A	92,000	Potassium iodide	10
Silicon oxide	24,000	Cesium chloride	160
Boron oxide	6,800	Barium chloride	70
Aluminum oxide	2,500	Lanthium chloride	90
Sodium oxide	2,700	Ceric chloride	140
Potassium oxide	140	Praseodymium chloride	70
Lithium-potassium chloride	13,000	Neodymium chloride	240
Sodium chloride	980	Samarium chloride	40
Rubidium chloride	20	Yttrium chloride	0
Strontium chloride	70	Total^c	14,000

- a. Source: Goff (1998a, all).
- b. To convert kilograms to pounds, multiply by 2.2046.
- c. Includes trace amounts of potassium bromide and europium chloride.

The halide composition would depend on the fuel processed. The final bulk composition of the ceramic waste form by weight percentages would be 25 percent glass, 63 to 65 percent zeolite-4A, and 10 to 12 percent halide salts.

Table A-32 lists the estimated composition of the second high-level radioactive waste form, which is a metal matrix waste form. The table combines percentage weight distribution with the total expected mass of the metal waste form to achieve a distributed mass by element (Goff 1998a, all).

Savannah River Site. Fowler et al. (1995, page 4) describes the chemical composition of the Defense Waste Processing Facility glass in detail. Table A-33 lists the distributed mass of the chemical constituents that comprise the current design-basis glass for the Savannah River Site. These values are based on a total mass of the glass of 11,600 metric tons (12,800 tons) (Picha 1997, Attachment 1).

West Valley Demonstration Project. The West Valley Demonstration Project will produce a single type of vitrified high-level radioactive waste. WVNS (1996, WQR-1.1, page 7) provides a target composition for all chemical constituents in the high-level radioactive waste. Table A-34 lists the expected chemical composition based on this target composition and the upper range of the projected total glass mass, 630 metric tons (694 tons).

Table A-32. Expected metal waste matrix chemical composition at Argonne National Laboratory-West (kilograms).^a

Component	Mass
Iron	4,200
Chromium	1,500
Nickel	1,100
Manganese	180
Molybdenum	220
Silicon	90
Zirconium	1,400
NMFPs ^b	360
Others ^c	20
Total	9,000

- a. Source: Goff (1998a, all); to convert kilograms to pounds, multiply by 2.2046.
- b. NMFPs = Noble metal fission products; includes silver, niobium, palladium, rhodium, ruthenium, antimony, tin, tantalum, technetium, and cobalt in small amounts.
- c. Others include trace amounts of carbon, phosphorus, and sulfur.

A.2.3.5.4 Thermal Output

Hanford Site. The estimated total thermal power from radioactive decay in the 14,500 reference canisters would be 1,190 kilowatts (as of January 1, 1994). This total heat load equates to an average power of 82 watts per canister. These values represent the hypothetical situation in which washed sludges from 177 tanks, cesium concentrates from the decontamination of low-level supernates, and strontium and cesium materials from capsules would be uniformly blended before vitrification. Realistically, uniform blending would not be likely. Current planning calls for merging all capsule materials with tank wastes from 2013 through 2016, which would create much hotter canisters during these years. In the extreme, the nonuniform blending of cesium concentrates and capsule materials into a relatively small volume of sludge waste could produce a few canisters with specific powers as high as 2,540 watts, which is the limit for the nominally 4.5-meter (15-foot) Hanford canisters in the Civilian Radioactive Waste Management System Baseline (Picha 1997, Attachment 1, page 2; Taylor 1997, all).

Table A-33. Expected Savannah River Site high-level radioactive waste chemical composition (kilograms).^{a,b}

Glass component	Mass	Glass component	Mass
Aluminum oxide	460,000	Sodium chloride	22,000
Barium sulfate	31,000	Neodymium	13,000
Calcium oxide	110,000	Nickel monoxide	100,000
Calcium sulfate	9,300	Neptunium	100
Cadmium	140	Promethium	210
Cerium	6,800	Praseodymium	3,300
Chromic oxide	14,000	Rubidium	120
Cesium oxide	14,000	Selenium	270
Copper oxide	51,000	Silicon oxide	5,800,000
Europium	200	Samarium	2,200
Ferric oxide	1,200,000	Tin	120
Potassium oxide	450,000	Tellurium	2,200
Lanthanum	3,500	Thorium dioxide	22,000
Lithium oxide	510,000	Titanium dioxide	100,000
Magnesium oxide	160,000	Uranium oxide	250,000
Manganese oxide	230,000	Zirconium	13,000
Molybdenum	14,000	Other ^c	58,000
Sodium oxide	1,000,000		
Sodium sulfate	12,000	Total	11,600,000

- a. Sources: Fowler et al. (1995, page 4); Picha (1997, Attachment 1).
- b. To convert kilograms to pounds, multiply by 2.2046.
- c. Includes trace amounts of silver, americium, cobalt, and antimony.

Table A-34. Expected West Valley Demonstration Project chemical composition (kilograms).^{a,b}

Compound	Mass	Compound	Mass
Aluminum oxide	38,000	Nickel monoxide	1,600
Boron oxide	82,000	Phosphorous pentaoxide	7,600
Barium oxide	1,000	Rubidium oxide	500
Calcium oxide	3,000	Silicon oxide	260,000
Ceric oxide	2,000	Strontium oxide	100
Chromic oxide	900	Thorium dioxide	23,000
Ferric oxide	76,000	Titanium dioxide	4,300
Potassium oxide	32,000	Uranium oxide	3,000
Lithium oxide	24,000	Zinc oxide	100
Magnesium oxide	5,600	Zirconium dioxide	7,100
Manganese oxide	5,200	Others	3,900
Sodium oxide	51,000		
Neodymium oxide	900	Total	630,000

- a. Sources: WVNS (1996, WQR-1.1, page 7); Picha (1998c, page 3).
- b. To convert kilograms to pounds, multiply by 2.2046.

Idaho National Engineering and Environmental Laboratory. The Laboratory has three proposed high-level radioactive waste streams. Table A-35 lists the thermal output of these waste streams per waste canister.

Savannah River Site. The radionuclide inventories reported for the Savannah River Site high-level radioactive waste in Section A.2.3.5.2 were used to calculate projected heat generation rates for single canisters.

For the design-basis waste form, the heat generation rates 10 and 20 years after production are 465 and 302 watts per canister, respectively (Plodinec, Moore, and Marra 1993, pages 8 and 9).

Table A-35. Idaho National Engineering and Environmental Laboratory waste stream thermal output (watts).^{a,b}

Output per waste canister	INTEC glass matrix	ANL-W ceramic matrix	ANL-W metal matrix
Average ^c	7.1	160	170
Worst case ^d	180	620	410

a. Source: Picha (1997, Attachment 1, page 2).

b. INTEC = Idaho Nuclear Technology and Engineering Center; ANL-W = Argonne National Laboratory-West.

c. Based on average case; 2035 used as base year for Idaho Nuclear Technology and Engineering Center glass and 2000 for ANL-W matrices.

d. Based on worst case; 2020 used as base year for Idaho Nuclear Technology and Engineering Center glass and 2000 for ANL-W matrices.

West Valley Demonstration Project. West Valley has calculated heat generation rates for a nominal West Valley canister after several different decay times (WVNS 1996, WQR-3.8, page 2). In the nominal case, the ORIGEN2-computed heat generation rate was 324 watts at the calculational base time in 1988. The heat generation rate would decrease continuously from 324 watts to about 100 watts after 50 years of additional decay.

A.2.3.5.5 Quantity of Waste Per Canister

Table A-36 lists the estimated mass of glass per waste canister for each high-level radioactive waste stream.

Table A-36. Mass of high-level radioactive waste glass per canister (kilograms).^a

Waste stream ^b	Mass per canister	Source
<i>Hanford</i>	3,040	Picha (1997, Attachment 1, page 2)
<i>INEEL</i>		
INTEC	1,560	Picha (1997, Attachment 1, page 2)
ANL-W ceramic ^c	960 - 1,500	Goff (1998a, all)
ANL-W metal ^c	1,500 - 4,850	Goff (1998a, all)
<i>Savannah River Site</i>	2,000	Pearson (1998, all)
<i>WVDP</i>	2,000	Picha (1997, Attachment 1, page 2)

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

b. INEEL = Idaho National Engineering and Environmental Laboratory; INTEC = Idaho Nuclear Technology and Engineering Center; ANL-W = Argonne National Laboratory-West; WVDP = West Valley Demonstration Project.

c. These values are estimates. ANL-W is evaluating waste package configurations compatible with existing storage and remote hot cell facilities. The geometries would be compatible with the Defense Waste Processing Facility high-level radioactive waste canister.

A.2.3.5.6 High-Level Radioactive Waste Canister Parameters

Hanford Site. Table A-37 lists preliminary physical parameters for a Hanford Tank Waste Remediation System standard canister (Picha 1997, Table RL-3).

Idaho National Engineering and Environmental Laboratory. The Idaho Nuclear Technology and Engineering Center would use stainless-steel canisters identical in design to those used at the Savannah River Site in the Defense Waste Processing Facility. A similar canister would also be used to contain the ceramic and metal waste matrices resulting from the proposed high-level radioactive waste processing at Argonne National Laboratory-West (Picha 1997, Table ID-1).

Table A-37. Parameters of proposed Tank Waste Remediation System standard canister for Hanford high-level radioactive waste disposal.^a

Parameter	Value ^b	Comments ^c
Length	4.50 meters	1.5 meters longer than DWPF and WVDP canisters - nominal 4.5-meter length
Outer diameter	0.61 meter	Same as DWPF and WVDP canisters
Material	304 stainless steel	Same as DWPF and WVDP canisters
Wall thickness	0.95 centimeter	Same as DWPF
Canister weight	720 kilograms	
Flange opening	0.41 meters	Same as WVDP canister; large opening
Dished bottom	Yes	Same as DWPF and WVDP
Available volume	1.2 cubic meters	
Nominal percent fill	90 percent	Provides approximately same void volume as WVDP canister
Glass volume	1.1 cubic meters	

a. Source: Picha (1997, Table RL-3).

b. To convert meters to feet, multiply by 3.2808; to convert centimeters to inches, multiply by 0.3937; to convert kilograms to tons, multiply by 0.0011023; to convert cubic meters to cubic feet, multiply by 35.314.

c. DWPF = Defense Waste Processing Facility; WVDP = West Valley Demonstration Project.

Savannah River Site. The fabrication specifications of the Defense Waste Processing Facility high-level radioactive waste canisters are described in detail in Marra, Harbour, and Plodinec (1995, all). The canisters are fabricated from four basic pieces of A240 304L austenitic stainless steel—the main cylinder, the bottom head, the top head, and a nozzle. The nominal wall thickness of the canister is 0.95 centimeter (0.37 inch).

West Valley Demonstration Project. The West Valley canister is designed, fabricated, and handled in accordance with the specifications in the West Valley Demonstration Project Waste Qualification Report (WVNS 1996, WQR-2.2, all). The West Valley canisters are fabricated from four principal 304L austenitic stainless-steel components. The nominal wall thickness of the canister is 0.34 centimeter (0.13 inch).

A.2.3.5.7 Nonstandard Packages

Each site that would ship high-level radioactive waste to the repository has provided additional data on an estimate of nonstandard packages for possible inclusion in the candidate waste material. The mass, volume, and radioactivity of potential nonstandard packages would be dominated by failed melters from the vitrification facilities. Final disposition plans for these melters are in development and vary from site to site. The EIS used the following assumptions to estimate the potential inventory.

Hanford Site. DOE could need to ship such nonstandard high-level radioactive waste packages as failed melters and failed contaminated high-level radioactive waste processing equipment to the repository. For this EIS, the estimated volume of nonstandard packages available for shipment to the repository from the Hanford Site would be equivalent to that described below for the Savannah River Site.

Idaho National Engineering and Environmental Laboratory. DOE proposes to treat and dispose of nonstandard packages under existing regulations. However, to bound the number of failed melters the Idaho National Engineering and Environmental Laboratory could ship to the repository, this EIS uses the same ratio of failed melters to the number of canisters produced as the Savannah River Site (Palmer 1997, page 2). The Idaho National Engineering and Environmental Laboratory would produce approximately 20 percent of the number of canisters produced at the Savannah River Site, which assumes 10 failed

melting. Therefore, the Idaho National Engineering and Environmental Laboratory assumes two failed melter. The volumes and other parameters would then be twice the values listed in Table A-38 for an individual melter.

Table A-38. Parameters of nonstandard packages from Savannah River Site.^a

Parameter	Value
Volume	10 melters based on current planning to 2021
Activity	4.5 equivalent DWPF ^b canisters for each melter
Mass	1,000 metric tons ^c for 10 melters (filled melter: 100 metric tons)
Chemical composition	Glass (see Section A.2.3.5.3) Melter – Refractory brick Aluminum Stainless steel Inconel
Quantity per disposal package	1 melter per disposal package
Thermal generation	4.5 times the heat generation of a single canister for each melter

a. Source: Pearson (1997, Attachment 1, pages 3 and 4).

b. DWPF = Defense Waste Processing Facility.

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

Savannah River Site. Table A-38 lists the estimated parameters of nonstandard packages for repository shipment from the Savannah River Site.

West Valley Demonstration Project. The West Valley Demonstration Project anticipates that it would send only one melter to the repository at the end of the waste solidification campaign. It would be treated as a nonstandard waste package. Table A-39 lists the estimated parameters of nonstandard packages from the West Valley Demonstration Project.

Table A-39. Parameters of nonstandard packages from West Valley Demonstration Project.^a

Parameter	Value ^b
Volume	1 melter (24 cubic meters)
Activity	1.1 equivalent West Valley canisters
Mass	52 metric tons
Chemical composition	Melter refractories (38 metric tons) Inconel (11 metric tons) Stainless steel (1.6 metric tons) Glass (see Table A-34)
Quantity per disposal package	1 melter per package
Thermal generator	1.1 times the heat generation of a single canister (A.2.3.5.4)

a. Source: Rowland (1997, all).

b. To convert cubic meters to cubic feet, multiply by 35.314; to convert metric tons to tons, multiply by 1.1023.

A.2.4 SURPLUS WEAPONS-USABLE PLUTONIUM

A.2.4.1 Background

The President has declared approximately 50 metric tons (55 tons) of weapons-usable plutonium to be surplus to national security needs (DOE 1998a, page 1-1). This material includes the following:

- Purified plutonium in various forms (metal, oxide, etc.)
- Nuclear weapons components (pits)

- High-purity materials that DOE could process in the future to produce purified plutonium
- Plutonium residues that DOE previously saved for future recovery of purified plutonium

These materials are currently stored at the Pantex Plant, the Rocky Flats Environmental Technology Site, the Savannah River Site, the Hanford Site, the Idaho National Engineering and Environmental Laboratory (Argonne National Laboratory-West), and the Oak Ridge, Los Alamos, and Lawrence Livermore National Laboratories. DOE would draw the specific surplus weapons-usable plutonium it ultimately disposed of from the larger inventory primarily stored at these sites.

DOE could process the surplus weapons-usable plutonium as two material streams. One stream would be an immobilized plutonium ceramic form that DOE would dispose of using a can-in-canister technique with high-level radioactive waste. The second stream would be mixed uranium and plutonium oxide fuel assemblies that would be used for power production in light-water reactors and disposed of as commercial spent nuclear fuel. The Surplus Plutonium Disposition Environmental Impact Statement (DOE 1998a, page 1-1) evaluates the quantity of plutonium processed in each stream. This EIS assumes that approximately 18 metric tons (20 tons) of surplus weapons-usable plutonium would be immobilized and approximately 32 metric tons (35 tons) would be made into mixed-oxide commercial nuclear fuel. The actual split could include the immobilization of between 18 and 50 metric tons (55 tons).

A.2.4.2 Sources

DOE would produce the immobilized plutonium and/or mixed-oxide fuel at sites determined in a Record of Decision for the Surplus Plutonium Disposition Environmental Impact Statement (DOE 1998a, page 1-9). The Department has selected for further environmental review six alternative commercial light-water reactors in which it proposes to irradiate the mixed-oxide fuel: both units at Catawba in York, South Carolina; both units at McGuire in Huntersville, North Carolina; and both units at North Anna Power Station in Mineral Springs, Virginia (DOE 1999, all).

A.2.4.3 Present Storage and Generation Status

DOE would begin production of the immobilized plutonium in 2006 with an estimated completion by 2016. The immobilization of 18 metric tons (20 tons) of plutonium would produce an estimated 77 additional canisters of high-level radioactive waste, which the production location would store until shipment to the repository. The immobilization of 50 metric tons (55 tons) of plutonium would produce an estimated 210 additional canisters of high-level radioactive waste. This EIS assumes that the production location would be the Savannah River Site and, therefore, used the physical dimensions of the Defense Waste Processing Facility canisters to calculate these values (DOE 1998a, pages 2-26 and 2-27).

Commercial light-water reactors would use mixed-oxide fuel assemblies for power production starting as early as 2007. This fuel would replace the low-enriched uranium fuel that normally would be in the reactors. After the fuel assemblies were discharged from the reactors as spent mixed-oxide fuel, the reactor sites would store them until shipment to the repository. Mixed-oxide fuel use would produce an insignificant number of additional spent nuclear fuel assemblies (less than 0.1 percent) (DOE 1998a, page 4-378).

A.2.4.4 Final Waste Form

The final waste form would be immobilized plutonium or spent mixed-oxide fuel. Section A.2.4.5 discusses the characteristics of these materials. The spent mixed-oxide fuel discussed here has different characteristics than the mixed-oxide fuel included in the National Spent Fuel Program (LMIT 1997, all) and described in Section A.2.2.

A.2.4.5 Material Characteristics

A.2.4.5.1 Mixed-Oxide Fuel

A.2.4.5.1.1 Mass and Volume. The EIS on surplus weapons-usable plutonium disposition (DOE 1998a, page 1-9) evaluates the disposal of approximately 32 metric tons (35 tons) of plutonium as mixed-oxide fuel. The amount of plutonium and uranium measured in metric tons of heavy metal going to a repository would depend on the average percentage of plutonium in the fuel. The percentage of plutonium would be influenced by the fuel design. DOE has chosen pressurized-water reactors for the proposed irradiation of these assemblies. For pressurized-water reactors, the expected average plutonium percentages would be approximately 4.6 percent; however, they could range between 3.5 and 6 percent (Stevenson 1997, pages 5 and 6). Table A-40 lists estimates and ranges for the total metric tons of heavy metal (uranium and plutonium) that would result from disposing of 32 metric tons (35 tons) of plutonium in mixed-oxide fuel. The table also lists a corresponding estimate for the number of assemblies required, based on using the typical assemblies described in Section A.2.1.4. The ranges of metric tons of heavy metal account for the proposed range in potential plutonium percentage.

Table A-40. Estimated spent nuclear fuel quantities for disposition of 32 metric tons of plutonium in mixed-oxide fuel.^{a,b}

Reactor and fuel type	Plutonium percentage	Best estimate (MTHM)	Assemblies required	Range (MTHM)
Pressurized-water reactor	4.56	700	1,500	500-900

a. Source: Stevenson (1997, pages 5 and 6).

b. MTHM = metric tons of heavy metal; to convert metric tons to tons, multiply by 1.1023.

DOE assumed that each spent mixed-oxide assembly irradiated and disposed of would replace an energy-equivalent, low-enriched uranium assembly originally intended for the repository. The mixed-oxide assemblies would be part of the 63,000 metric tons (69,000 tons) that comprise the commercial spent nuclear fuel disposal amount in the Proposed Action (Person 1998, all). DOE also assumes that the average burnup levels for the pressurized-water reactor would be the same as that for the energy-equivalent, low-enriched uranium fuel. Table A-41 lists the assumed burnup levels and the amount of heavy metal in an assembly.

Table A-41. Assumed design parameters for typical mixed-oxide assembly.^a

Parameter	Pressurized-water reactor
Mixed-oxide and low-enriched uranium burnup (MWd/MTHM) ^b	45,000
Mixed-oxide assembly mass (kilograms ^c of heavy metal)	450
Mixed-oxide assembly percentage of plutonium	4.56

a. Source: Stevenson (1997, page 7).

b. MWd/MTHM = megawatt days per metric ton of heavy metal; to convert metric tons to tons, multiply by 1.1023.

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

The analysis assumed that the mixed-oxide spent nuclear fuel would replace the low-enriched uranium fuel. Because of the similarities in the two fuel types, impacts to the repository would be small. Nuclear criticality, radionuclide release rates, and heat generation comparisons are evaluated in Stevenson (1997, pages 35 to 37).

A.2.4.5.1.2 Amount and Nature of Radioactivity. Tables A-42 and A-43 list isotopic composition data for spent mixed-oxide fuel assemblies. The tables reflect SCALE data files from an Oak Ridge National Laboratory report used with computer simulation to project the characteristics of spent mixed-oxide fuel in pressurized-water reactors (Ryman, Hermann, and Murphy 1998, Volume 3, Appendix B). The tables summarize data for two different potential fuel assemblies: a typical pressurized-water reactor,

Table A-42. Radionuclide activity for typical pressurized-water reactor spent mixed-oxide assembly.^a

Isotope	Curies per assembly	Isotope	Curies per assembly
Hydrogen-3	2.0×10 ²	Samarium-151	5.3×10 ²
Carbon-14	3.4×10 ⁻¹	Uranium-234	4.9×10 ⁻²
Cobalt-60	1.7×10 ³	Uranium-235	1.0×10 ⁻³
Nickel-59	1.1	Uranium-236	6.4×10 ⁻³
Nickel-63	1.4×10 ²	Uranium-238	1.4×10 ⁻¹
Krypton-85	1.9×10 ³	Plutonium-238	1.2×10 ³
Strontium-90	1.7×10 ⁴	Plutonium-239	6.6×10 ²
Zirconium-93	6.5×10 ⁻²	Plutonium-240	8.6×10 ²
Niobium-93m	2.8×10 ¹	Plutonium-241	2.0×10 ⁵
Niobium-94	6.8×10 ⁻¹	Americium-241	2.2×10 ³
Technetium-99	6.3	Americium-242/242m	3.4×10 ¹
Ruthenium-106	1.6×10 ⁴	Americium-243	2.4×10 ¹
Iodine-129	2.1×10 ⁻²	Curium-242	6.0×10 ¹
Cesium-134	1.4×10 ⁴	Curium-243	3.2×10 ¹
Cesium-137	4.7×10 ⁴	Curium-244	2.6×10 ³

a. Source: Ryman, Hermann, and Murphy (1998, Volume 3, Appendix B).

Table A-43. Radionuclide activity for high-burnup pressurized-water reactor spent mixed-oxide assembly.^a

Isotope	Curies per assembly	Isotope	Curies per assembly
Hydrogen-3	2.9×10 ²	Uranium-234	6.8×10 ⁻²
Carbon-14	5.4×10 ⁻¹	Uranium-235	6.7×10 ⁻⁴
Cobalt-60	2.4×10 ³	Uranium-236	7.7×10 ⁻³
Nickel-59	1.7	Uranium-238	1.5×10 ⁻¹
Nickel-63	2.3×10 ²	Plutonium-238	2.7×10 ³
Krypton-85	2.6×10 ³	Plutonium-239	4.6×10 ²
Strontium-90	2.4×10 ⁴	Plutonium-240	8.8×10 ²
Niobium-93m	3.9×10 ¹	Plutonium-241	2.2×10 ⁵
Niobium-94	9.8×10 ⁻¹	Americium-241	2.5×10 ³
Technetium-99	9.0	Americium-242/242m	4.9×10 ¹
Ruthenium-106	1.8×10 ⁴	Americium-243	5.6×10 ¹
Iodine-129	3.0×10 ⁻²	Curium-242	1.0×10 ²
Cesium-134	2.5×10 ⁴	Curium-243	8.5×10 ¹
Cesium-137	7.0×10 ⁴	Curium-244	8.9×10 ³
Samarium-151	5.4×10 ²		

a. Sources: Ryman, Hermann, and Murphy (1998, Volume 3, Appendix B).

and a high-burnup pressurized-water reactor. A high burnup pressurized-water assembly would be irradiated for three cycles in comparison to the two cycles for the typical assemblies. For each of these assemblies, the tables provide radioactivity data for the common set of nuclides used in this EIS for the assumed 5-year minimum cooling time.

A.2.4.5.1.3 Chemical Composition. Tables A-44 and A-45 list the elemental distributions for the typical and high-burnup pressurized-water reactor spent mixed-oxide fuel assemblies.

A.2.4.5.1.4 Thermal Output. Table A-46 lists the decay heat from the representative mixed-oxide spent fuel assemblies at a range of times after discharge.

A.2.4.5.1.5 Physical Parameters. Because the mixed-oxide fuel would replace low-enriched uranium fuel in existing reactors, Section A.2.1.5.5 describes the physical parameters, with the exception of uranium and plutonium content, which are listed in Table A-41.

Table A-44. Elemental distribution of typical burn-up pressurized-water reactor spent mixed-oxide assembly.^a

Element	Grams per assembly ^b	Percent ^c	Element	Grams per assembly	Percent
Americium	770	0.12	Palladium	1,200	0.19
Barium	750	0.12	Phosphorus	140	0.02
Carbon	67	0.01	Plutonium	17,000	2.59
Cerium	1,100	0.16	Praseodymium	500	0.08
Cesium	1,500	0.23	Rhodium	360	0.05
Chromium	2,300	0.36	Rubidium	91	0.01
Europium	90	0.01	Ruthenium	1,300	0.20
Iodine	150	0.02	Samarium	440	0.07
Iron	4,600	0.71	Silicon	66	0.01
Krypton	100	0.02	Strontium	210	0.03
Lanthanum	540	0.08	Technetium	370	0.06
Manganese	110	0.02	Tellurium	260	0.04
Molybdenum	1,700	0.27	Tin	1900	0.28
Neodymium	1,700	0.26	Uranium	428,000	65.92
Neptunium	72	0.01	Xenon	2500	0.38
Nickel	4,400	0.68	Yttrium	110	0.02
Niobium	330	0.05	Zirconium	111,000	17.10
Oxygen	62,000	9.56	Totals	648,000	99.73

a. Source: Murphy (1998, all).

b. To convert grams to ounces, multiply by 0.035274.

c. Table includes only elements that constitute at least 0.01 percent of the total; therefore, total is slightly less than 100 percent.

Table A-45. Elemental distribution of high burn-up pressurized-water reactor spent mixed-oxide assembly.^a

Element	Grams per assembly ^b	Percent ^c	Element	Grams per assembly	Percent
Americium	1,000	0.16	Palladium	2,000	0.30
Barium	1,200	0.18	Phosphorus	140	0.02
Carbon	70	0.01	Plutonium	14,000	2.22
Cerium	1,600	0.24	Praseodymium	750	0.11
Cesium	2,100	0.33	Rhodium	460	0.07
Chromium	2,300	0.36	Rubidium	140	0.02
Europium	140	0.02	Ruthenium	2,000	0.31
Iodine	220	0.03	Samarium	630	0.10
Iron	4,600	0.71	Silicon	66	0.01
Krypton	150	0.02	Strontium	300	0.05
Lanthanum	810	0.12	Technetium	520	0.08
Manganese	100	0.02	Tellurium	390	0.06
Molybdenum	2,500	0.39	Tin	1,900	0.29
Neodymium	2,500	0.39	Uranium	421,000	64.84
Neptunium	93	0.01	Xenon	3,700	0.57
Nickel	4,400	0.68	Yttrium	170	0.03
Niobium	330	0.05	Zirconium	111,000	17.10
Oxygen	62,000	9.56	Totals	646,000	99.46

a. Source: Murphy (1998, all).

b. To convert grams to ounces, multiply by 0.035274.

c. Table includes only elements that constitute at least 0.01 percent of the total; therefore, total is slightly less than 100 percent.

Table A-46. Mixed-oxide spent nuclear fuel thermal profile (watts per assembly).^a

Years	Typical PWR ^b	High-burnup PWR
1	6,100	8,000
5	1,000	1,600
10	670	1,100
15	610	970
30	540	780
100	370	430
300	240	260
1,000	110	110
3,000	42	38
10,000	25	22
30,000	10	7.9
100,000	1.5	1.3
250,000	0.5	0.6

a. Source: Ryman, Hermann, and Murphy (1998, Volume 3, Appendix B).

b. PWR = pressurized-water reactor.

A.2.4.5.2 Immobilized Plutonium

At present, approximately 50 metric tons (55 tons) of weapons-usable plutonium have been declared to be surplus to national needs. DOE has not yet determined the total quantity of plutonium for immobilization. The Department assumes that approximately 32 metric tons (35 tons) is “clean” metal suitable for use in mixed-oxide fuel, and that it could dispose of this material by burning it in reactors (DOE 1998a, page 1-1). The remaining surplus plutonium would require considerable additional chemical processing to make it suitable for reactor use. This EIS evaluates two cases, one in which DOE immobilizes only the “impure” materials (base case) and a second in which it immobilizes the entire 50-metric-ton surplus inventory. The base case is evaluated for the Proposed Action because it is DOE’s preferred alternative (DOE 1998a, page 1-1). The EIS evaluates the second case for potential cumulative impacts (Modules 1 and 2) because it would conservatively predict the largest number of required high-level radioactive waste canisters.

A.2.4.5.2.1 Mass and Volume. In DOE’s preferred disposition alternative, immobilized plutonium would arrive at the repository in canisters of vitrified high-level radioactive waste that would be externally identical to standard canisters from the Defense Waste Processing Facility at the Savannah River Site. Smaller cans containing immobilized plutonium in ceramic disks would be embedded in each canister of high-level radioactive waste glass. This is the *can-in-canister* concept. Because the design of the can-in-canister is not final, DOE has not determined final waste loadings per canister, volume displaced by the cans, or other specifications. The current baseline concept calls for cylindrical cans that are 53 centimeters (21 inches) high with a 7.6-centimeter (3-inch) diameter. The gross volume of each can would be 2.4 liters (150 cubic inches). DOE estimates that each canister would contain 28 cans, but has not yet finalized the actual number. One of the limitations on the number of cans is determined by the ability to ensure that the high-level radioactive waste glass would fill completely around the cans; increasing the volume that the cans would occupy in a canister could increase the difficulty of achieving this. Final confirmation of the design will be confirmed by actual test pours at scale (Stevenson 1997, page 41).

Marra, Harbour, and Plodinec (1995, page 2) describes the volume of a high-level radioactive waste canister. Each canister has a design capacity of 2,000 kilograms (4,400 pounds) of high-level radioactive waste glass. A nominal glass density of 2.7 grams per cubic centimeter (0.10 pound per cubic inch)

yields a design glass volume of 620 liters (22 cubic feet). The 28 cans containing plutonium would displace 68 liters (2.4 cubic feet), or about 11 percent of the available volume. The rack holding the cans would displace about an additional 1 percent of the available volume, yielding a total displacement of about 12 percent.

Each plutonium can would contain 20 cylindrical pellets, 6.7 centimeters (2.6 inches) in diameter and 2.5 centimeters (1 inch) in height. The pellets would have an average density of 5.5 grams per cubic centimeter (0.20 pound per cubic inch) and would contain 10.5 percent of plutonium by weight. Each can, therefore, would contain about 1 kilogram (2.2 pounds) of plutonium, yielding a total of about 28 kilograms (62 pounds) per canister (1 kilogram of plutonium per can multiplied by 28 cans per canister).

Table A-47 lists the number of high-level radioactive waste canisters required to dispose of immobilized surplus plutonium using the loading and volumetric assumptions given above for both the base and 50-metric-ton (55-ton) cases. It also lists the number of additional canisters DOE would have to produce (in addition to those the high-level radioactive waste producer would already have produced) due to the displacement of high-level radioactive waste glass by the plutonium-containing canisters. The total number of required canisters would be a function of both the number of cans in each canister and the plutonium loading of the immobilization form. The number of additional canisters would depend only on the plutonium loading of the immobilization form.

Table A-47. Number of canisters required for immobilized plutonium disposition.^{a,b}

Canisters	Base case	50-metric-ton case
Containing plutonium	635	1,744
In excess of those required for DWPF ^c (12% of total canisters)	77	210
Additional ^d	1.3%	3.5%

a. Source: DOE (1998a, pages 2-26 and 2-27).

b. Assumes 28 kilograms (62 pounds) of plutonium per canister and displacement of 12 percent of the high-level radioactive waste glass by plutonium cans and rack.

c. DWPF = Defense Waste Processing Facility.

d. As percentage of total planned DWPF canisters (about 6,000).

A.2.4.5.2.2 Amount and Nature of Radioactivity. Assuming the current 10.5-percent plutonium loading in the ceramic (Stevenson 1997, page 49), the expected isotopic composition of the various materials in the feedstream for ceramic production, and the nominal quantity of ceramic in each canister, Stevenson (1997, page 49) calculated the activity of the immobilized material in each high-level radioactive waste canister. The figures do not include the radioactivity of the vitrified high-level radioactive waste that would surround the cans of immobilized plutonium. Calculation of the total radioactivity of a canister requires the subtraction of approximately 12 percent from the radioactivity of a full high-level radioactive waste canister to account for the displacement of the immobilized plutonium and its rack. Those reduced numbers, added to the appropriate figures in Table A-48, produce the total activity of a plutonium-containing high-level radioactive waste canister.

Values for the base case and the 50-metric-ton case are different because the plutonium in the base case contains more transuranic radionuclides, other than plutonium-239, than does the remainder of the plutonium [32 metric tons (35 tons)]. Thus, the "other" transuranic radionuclides are diluted in the 50-metric-ton case. From a thermal output and radiological impact standpoint, the base case is a more severe condition and, therefore, DOE has used it for the Proposed Action analysis.

Section A.2.3.5.2 contains information on the radioactivity contained in a standard Defense Waste Processing Facility high-level radioactive waste canister.

Table A-48. Average total radioactivity of immobilized plutonium ceramic in a single canister in 2010 (curies).^{a,b}

Nuclide	Base case	50-metric-ton case
Plutonium-238	120	60
Plutonium-239	1,600	1,700
Plutonium-240	550	430
Plutonium-241	4,700	2,800
Plutonium-242	0.098	0.046
Americium-241	720	430
Uranium-234	< 0.000015 ^c	< 0.000005
Uranium-235	0.0024	< 0.0011
Uranium-238	0.019	0.019
Thorium-232	< 0.00003	< 0.00003
Totals	7,700	5,400

- a. Source: Stevenson (1997, page 49).
 b. Assumes 10.5 percent of plutonium by weight in ceramic form, 1:2 molar ratio of plutonium to uranium, and 28 kilograms (62 pounds) of plutonium per canister. These values account only for the radioactivity in the immobilized form; they do not include that in the surrounding high-level radioactive waste glass.
 c. < = less than.

A.2.4.5.2.3 Chemical Composition. The current design for a ceramic immobilization form is a multiphase titanate ceramic, with a target bulk composition listed in Table A-49. The neutron absorbers, hafnium and gadolinium, are each present at a 1-to-1 atomic ratio to plutonium, and the atomic ratio of uranium to plutonium is approximately 2-to-1. For the base case, the presence of impurities in some categories of surplus weapons-usable plutonium would result in the presence of a few weight percent of other nonradioactive oxides in some of the actual ceramic; Table A-49 does not list these impurities (Stevenson 1997, page 51).

Table A-49. Chemical composition of baseline ceramic immobilization form.^a

Oxide	Approximate percent by weight
Titanium oxide	36
Hafnium oxide	10
Calcium oxide	10
Gadolinium oxide	8
Plutonium oxide	12
Uranium oxide	24

- a. Source: Stevenson (1997, page 51).

The ceramic phase assemblage is mostly Hf-pyrochlore [(CaGd)(Gd,Pu,U,Hf)Ti₂O₇], with subsidiary Hf-zirconolite [(CaGd)(Gd,Pu,U,Hf)Ti₂O₇], and minor amounts of brannerite [(U,Pu,Gd)Ti₂O₆] and rutile [(Ti,Hf)O₂]. Pyrochlore and zirconolite differ in their crystalline structures. The presence of silicon as an impurity in the plutonium could lead to the formation of a minor amount of a silicate glass phase in the ceramic. This phase could contain a trace amount of the immobilized plutonium. Some residual plutonium oxide (less than 0.5 percent of the total quantity of plutonium) could also be present. The residual plutonium oxide contains uranium with smaller amounts of gadolinium and hafnium as a result of partial reaction with the other constituents of the ceramic (Stevenson 1997, page 51). Section A.2.3.5.3 describes the chemical composition of the high-level radioactive waste glass surrounding the plutonium-containing cans.

A.2.4.5.2.4 Thermal Output. Stevenson (1997, page 49) has presented the heat generation of the immobilized ceramic. These figures represent only the heat from the ceramic; they do not account for the heat from the surrounding high-level radioactive waste glass. The total heat from a Defense Waste Processing Facility canister containing high-level radioactive waste and immobilized plutonium would be the value listed in Table A-50 combined with 88 percent of the value listed in Section A.2.3.5.4 for the heat from a Defense Waste Processing Facility canister.

Table A-50. Thermal generation from immobilized plutonium ceramic in a single canister in 2010 (watts per canister).^a

Case	Thermal production
Base case	8.6
50-metric-ton ^b case	7.0

a. Source: Stevenson (1997, page 49).

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

A.2.4.5.2.5 Quantity of Material Per Canister. As discussed in Section A.2.4.5.2.1, DOE has yet to determine the actual configuration of the can-in-canister disposal package. Although the final configuration could use either the Savannah River Site or Hanford canisters, this EIS assumes the use of the Savannah River Site canister. The current baseline concept (described above) would result in a per-canister loading of 28 kilograms (62 pounds) of plutonium. Table A-48 lists the radioactivities of these materials. Section A.2.3.5.5 discusses the quantity of high-level radioactive waste associated with each Defense Waste Processing Facility canister. The quantity of high-level radioactive waste in each plutonium-containing canister would be less than the nominal content of a standard Defense Waste Processing Facility canister because the displacement of the plutonium cans and the support rack would amount to an estimated 12 percent of the net canister volume.

The canisters would differ internally from normal Defense Waste Processing Facility canisters due to the presence of the stainless-steel cans of immobilized plutonium and a stainless-steel rack holding the cans in place during pouring of molten high-level radioactive waste glass into the canister.

A.2.5 COMMERCIAL GREATER-THAN-CLASS-C LOW-LEVEL WASTE

A.2.5.1 Background

Title 10 of the Code of Federal Regulations, Part 61 (10 CFR Part 61), establishes disposal requirements for three classes of waste—A, B, and C—suitable for near-surface disposal. Class C has the highest level of radioactivity and therefore the most rigorous disposal specifications. Wastes with concentrations above Class C limits (listed in 10 CFR 61.55 Tables 1 and 2 for long and short half-life radionuclides, respectively) are called Greater-Than-Class-C low-level waste, and are not generally suitable for near-surface disposal (DOE 1994, all).

Commercial nuclear powerplants, research reactors, radioisotope manufacturers, and other manufacturing and research institutions generate waste that exceeds the Nuclear Regulatory Commission Class C shallow-land-burial disposal limits. Public Law 99-240 assigns the Federal Government, specifically DOE, the responsibility for disposing of this Greater-Than-Class-C waste. DOE could use a number of techniques for the disposal of these wastes, including engineered near-surface disposal, deep borehole disposal, intermediate-depth burial, and disposal in a deep geologic repository (DOE 1994, all).

The activities of nuclear electric utilities and other radioactive waste generators to date have produced relatively small quantities of Greater-Than-Class-C waste. As the utilities take their reactors out of service and decommission them, they could generate more waste of this type (DOE 1994, all).

Greater-Than-Class-C waste could include the following materials:

- Nuclear powerplant operating wastes
- Nuclear powerplant decommissioning wastes
- Sealed radioisotope sources that exceed Class C limits for waste classification
- DOE-held Greater-Than-Class-C waste (addressed in Section A.2.6)
- Greater-Than-Class-C waste from other generators

This section describes the quantities and characteristics of these waste types.

A.2.5.2 Sources

Sources or categories of Greater-Than-Class-C waste include:

- DOE facilities (addressed in Section A.2.6)
- Nuclear utilities
- Sealed sources
- Other generators

Nuclear utility waste includes activated metals and process wastes from commercial nuclear powerplants. Sealed sources are radioactive materials in small metallic capsules used in measurement and calibration devices. Other generator wastes consist of sludge, activated metals, and other wastes from radionuclide manufacturers, commercial research, sealed-source manufacturers, and similar operations. The decommissioning of light-water reactors probably will generate additional Greater-Than-Class-C waste. Some internal reactor components will exceed Class C disposal limits.

A.2.5.3 Present Status

Nuclear utilities store their Greater-Than-Class-C waste at the generator site, where it will remain until a disposal option becomes available.

Sealed sources are held by a Nuclear Regulatory Commission or Agreement State licensee. Current DOE sealed-source management plans call for the licensees to store their sealed-source wastes until a disposal option becomes available. If storage by a licensee became physically or financially impossible and a threat to public health and safety, the Nuclear Regulatory Commission would determine if the source was a candidate for DOE storage. At that time, the Commission could request that DOE accept the source for storage, reuse, or recycling. The inventory projections do not include such a transfer of material.

In 1993, there were 13 identified "other generators" of Greater-Than-Class-C waste (DOE 1994, Appendix D), which were categorized into seven business types:

- Carbon-14 user
- Industrial research and development
- Irradiation laboratory
- Fuel fabricator
- University reactor
- Sealed-source manufacturer
- Nonmedical academic institution

These generators store their wastes at their sites and will continue to do so until a disposal site becomes operational.

A.2.5.4 Final Waste Form

The final disposition method for Greater-Than-Class-C waste is not known. If DOE was to place such waste in a repository, it is assumed that it would be placed in a disposal package before shipment. The EIS assumes the use of a package similar to the naval dual-purpose canister, which is described in Section A.2.2.5.6, for all shipments by rail and a package similar to the high-level radioactive waste canisters for all shipments by truck.

A.2.5.5 Waste Characteristics

Table A-51 lists existing and projected volumes for the three Greater-Than-Class-C waste generator sources. DOE conservatively projects the volume of nuclear utility wastes to 2055 because that date would include the majority of this waste from the decontamination and decommissioning of commercial nuclear reactors. The projected volumes conservatively reflect the highest potential volume and activity based on inventories, surveys, and industry production rates. DOE projects the other two generator sources (sealed sources and other generators) to 2035 (DOE 1994, all).

Table A-51. Greater-Than-Class-C waste volume by generator source (cubic meters).^{a,b}

Source	1993 volume	Projected volume
Nuclear electric utility	26	1,300
Sealed sources	39	240
Other generators	74	470
Totals	139	2,010

a. Source: DOE (1994, all).

b. To convert cubic meters to cubic feet, multiply by 35.314.

The data concerning the volumes and projections are from Greater-Than-Class-C Waste Characterization: Estimated Volumes, Radionuclide Activities, and Other Characteristics (DOE 1994), Appendix A-1, which provides detailed radioactivity reports for such waste currently stored at nuclear utilities. Table A-52 summarizes the radioactivity data for the primary radionuclides in the waste, projected to 2055.

Table A-52. Commercial light-water reactor Greater-Than-Class-C waste radioactivity (curies) by nuclide (projected to 2055).^a

Nuclide	Radioactivity
Carbon-14	6.8×10^4
Cobalt-60	3.3×10^7
Iron-55	1.8×10^7
Hydrogen-3	1.2×10^4
Manganese-54	3.2×10^4
Niobium-94	9.8×10^2
Nickel-59	2.5×10^5
Nickel-63	3.7×10^7
Transuranics	2.0x1 0 ³
Total	8.8×10^7

a. Source: DOE (1994, Appendix A-1).

Appendix B of DOE (1994) provides detailed radioactivity reports for the sealed sources, which could be candidate wastes for the repository. Table A-53 summarizes the radioactivity data for the radionuclides in these sources, projected to 2035.

Table A-53. Sealed-source Greater-Than-Class-C waste radioactivity (curies) by nuclide (projected to 2035).^a

Nuclide	Radioactivity
Americium-241	8.0×10^4
Curium-244	1.6×10^2
Cesium-137	4.0×10^7
Plutonium-238	1.6×10^4
Plutonium-239	1.1×10^5
Plutonium-241	2.8×10^1
Technetium-99	5.8×10^3
Uranium-238	5.7×10^1
Total	4.2×10^7

a. Source: DOE (1994, Appendix B).

DOE (1994, Section 5) also identifies the 13 other generators and the current and projected volumes and total radioactivity of Greater-Than-Class-C waste held by each. It does not provide specific radionuclide activity by nuclide. DOE used the data to derive a distribution, by user business type, of the specific nuclides that comprise the total radioactivity. Table A-54 lists this distributed radioactivity for other generators.

Table A-54. Other generator Greater-Than-Class-C waste radioactivity (in curies) by nuclide (projected to 2035).^a

Nuclide	Radioactivity
Carbon-14	7.7×10^3
Transuranic	2.2×10^3
Cobalt-60	1.5×10^2
Nickel-63	1.5×10^2
Americium-241	2.4×10^3
Cesium-137	6.6×10^1
Technetium-99	5.1×10^{-2}
Total^b	1.3×10^4

a. Source: Derived from DOE (1994, Appendix D).

b. Total differs from sum of values due to rounding.

A detailed chemical composition by weight percentage for current Greater-Than-Class-C waste is not available. However, Table A-55 lists the typical composition of such wastes by generator.

Table A-55. Typical chemical composition of Greater-Than-Class-C wastes.^a

Source	Typical composition
Nuclear electric utility	Stainless steel-304, and zirconium alloys
Sealed sources	Stainless steel-304 (source material has very small mass contribution)
Other generators	Various materials

a. Source: DOE (1994, all).

The heat generation rates or thermal profiles for this waste type are not included in the source documentation. However, the contribution to the total thermal load at the repository from the Greater-Than-Class-C radioactive waste would be very small in comparison to commercial spent nuclear fuel or high-level radioactive waste.

A.2.6 SPECIAL-PERFORMANCE-ASSESSMENT-REQUIRED LOW-LEVEL WASTE

A.2.6.1 Background

DOE production reactors, research reactors, reprocessing facilities, and research and development activities generate wastes that exceed the Nuclear Regulatory Commission Class C shallow-land-burial disposal limits. The Department is responsible for the safe disposal of such waste, and could use a number of techniques such as engineered near-surface disposal, deep borehole disposal, intermediate-depth burial, or disposal in a deep geologic repository. These wastes have been designated as Special-Performance-Assessment Required wastes.

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

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

A.2.6.2 Sources

DOE has identified Special-Performance-Assessment-Required waste inventories at several locations. Table A-56 lists the generators and amounts of these wastes. These amounts include current and projected inventory. The Department will generate additional waste as it decommissions its nuclear facilities.

Table A-56. Estimated Special-Performance-Assessment-Required low-level waste volume and mass by generator source.^a

Source ^b	Volume (cubic meters) ^c	Mass (kilograms) ^d
Hanford	20	360,000
INEEL ^e	20	280,000
ORNL	2,900	4,700,000
WVDP	550	5,200,000
ANL-E	1	230
Naval Reactors Facility	500	2,500,000
Totals	4,000	13,040,230

a. Source: Picha (1998b, all).

b. INEEL = Idaho National Engineering and Environmental Laboratory (including Argonne National Laboratory-West); ORNL = Oak Ridge National Laboratory; WVDP = West Valley Demonstration Project; ANL-E = Argonne National Laboratory-East.

c. To convert cubic meters to cubic yards, multiply by 1.3079.

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

e. Includes Argonne National Laboratory-West.

A.2.6.3 Present Status

DOE stores its Special-Performance-Assessment-Required waste at the generator sites listed in Table A-56. Tables A-57 through A-60 list the waste inventories at the individual sites. For radionuclides, these tables include only the reported isotopes with inventories greater than 1×10^{-5} curies. Table A-61 lists the chemical composition of this material at each site.

Table A-57. Hanford Special-Performance-Assessment-Required low-level waste radioactivity by nuclide (curies).^a

Nuclide	Radioactivity
Cesium-137	6.0×10^4
Strontium-90	6.0×10^4

a. Source: Picha (1998b, all).

Table A-58. Idaho National Engineering and Environmental Laboratory (including Argonne National Laboratory-West) Special-Performance-Assessment-Required low-level waste radioactivity by nuclide (curies).^a

Nuclide	Radioactivity
Hydrogen-3	5.9×10^6
Carbon-14	8.3×10^2
Cobalt-60	1.1×10^6
Nickel-59	9.0×10^1
Nickel-63	1.3×10^4
Strontium-90	7.4×10^3
Niobium-94	1.4×10^2
Technetium-99	3.3
Cesium-137	3.1×10^1
Radium-226	3.0×10^1
Plutonium-239	2.0×10^1
Americium-241	2.4×10^2

a. Source: Picha (1998b, all).

Table A-59. Oak Ridge National Laboratory Special-Performance-Assessment-Required low-level waste radioactivity by nuclide (curies).^a

Nuclide	Radioactivity
Hydrogen-3	1.9×10^6
Carbon-14	1.0×10^1
Cobalt-60	1.9×10^6
Nickel-59	7.6×10^3
Nickel-63	7.5×10^5
Strontium-90	8.3×10^7
Niobium-94	1.0×10^4
Technetium-99	8.0×10^{-1}
Iodine-129	7.5×10^{-5}
Cesium-137	1.7×10^{-4}

a. Source: Picha (1998b, all).

Table A-60. Radioactivity of naval Special-Performance-Assessment-Required waste (curies per package).^a

Isotope	Short canister	Long canister	Isotope	Short canister	Long canister
Americium-241	5.4×10^{-2}	6.0×10^{-2}	Nickel-59	2.2×10^2	2.5×10^2
Americium-242m	5.8×10^{-4}	6.5×10^{-4}	Nickel-63	2.7×10^4	3.0×10^4
Americium-243	5.8×10^{-4}	6.5×10^{-4}	Plutonium-239	2.1×10^{-2}	2.4×10^{-2}
Carbon-14	3.2	3.6	Plutonium-240	5.4×10^{-3}	6.0×10^{-3}
Chlorine-36	5.3×10^{-2}	6.0×10^{-2}	Plutonium-241	4.1	4.6
Curium-242	1.4×10^{-3}	1.5×10^{-3}	Plutonium-242	4.5×10^{-5}	5.1×10^{-5}
Curium-243	6.6×10^{-4}	7.4×10^{-4}	Ruthenium-106	2.1×10^{-1}	2.3×10^{-1}
Curium-244	7.0×10^{-2}	7.9×10^{-2}	Selenium-79	1.2×10^{-5}	1.3×10^{-5}
Curium-245	1.3×10^{-5}	1.5×10^{-5}	Samarium-151	1.7×10^{-2}	1.9×10^{-2}
Cesium-134	1.6	1.8	Tin-126	1.2×10^{-5}	1.3×10^{-5}
Cesium-135	1.1×10^{-5}	1.2×10^{-5}	Strontium-90	4.2×10^{-1}	4.7×10^{-1}
Cesium-137	1.1	1.3	Technetium-99	5.3×10^{-4}	6.0×10^{-4}
Hydrogen-3	1.5	1.7	Uranium-232	1.2×10^{-4}	1.4×10^{-4}
Krypton-85	4.9×10^{-2}	5.6×10^{-2}	Uranium-233	7.8×10^{-5}	8.8×10^{-5}
Niobium-93m	3.6×10^{-1}	4.1×10^{-1}	Zirconium-93	3.8×10^{-1}	4.3×10^{-1}
Niobium-94	5.9×10^{-1}	6.7×10^{-1}			

a. Source: Beckett (1998, Attachment 1).

Table A-61. Typical chemical composition of Special-Performance-Assessment-Required low-level waste.^a

Source ^b	Composition
Hanford	Vitrified fission products in glass waste form; hot cell waste
INEEL	Activated metal
ORNL	Activated metal; isotope production waste; hot cell waste
WVDP	Activated metal; vitrified transuranic waste
Naval Reactors	Activated metal (zirconium alloy, Inconel, stainless steel)
Other generators	Stainless-steel sealed sources

a. Source: Picha (1998b, all).

b. INEEL = Idaho National Engineering and Environmental Laboratory; ORNL = Oak Ridge National Laboratory; WVDP = West Valley Demonstration Project.

A.2.6.4 Final Waste Form

The final disposal method for DOE Special-Performance-Assessment-Required waste is not known. If the Department disposed of such waste in a repository, it is assumed that the material would be placed in a disposable package before shipment to the repository. The EIS assumes the use of a dual-purpose canister similar to those used for naval fuels for all rail shipments and packages similar to a high-level radioactive waste canister for all truck shipments.

A.2.6.5 Waste Characteristics

The low-level waste from West Valley consists of material in the Head End Cells (5 cubic meters [177 cubic feet]) and remote-handled and contact-handled transuranic waste (545 cubic meters [19,000 cubic feet]). The estimated radioactivity of the material in the Head End Cells is 6,750 curies, while the activity of the remote-handled and contact-handled transuranic waste is not available at present (Picha 1998b, all). The naval Special-Performance-Assessment-Required waste consists primarily of zirconium alloys, Inconel, and stainless steel (Beckett 1998, all); Table A-60 lists the specific radioactivity of the projected material 5 years after discharge.

The specific activity associated with the radium sources at Argonne National Laboratory-East has not been determined. However, in comparison to the other Special-Performance-Assessment-Required waste included in this section, its impact would be small.

REFERENCES

- Beckett 1998 Beckett, T. H., 1998, "Response to Data Request," interoffice memorandum to P. J. Dirkmaat (Idaho Operations Office) and K. G. Picha (Office of Waste Management), Office of Naval Reactors, U.S. Department of Energy, Washington, D.C. [MOL.19990511.0293]
- Cole 1998a Cole, B., 1998a, "EIS Comments," memorandum to J. Rivers (Jason Associates, Inc.), U.S. Department of Energy, Washington, D.C. [MOL.19990511.0303]
- Cole 1998b Cole, B., 1998b, "Stainless Steel Clad SNF," memorandum to J. Rivers (Jason Associates, Inc.), U.S. Department of Energy, Washington, D.C. [MOL.19990511.0302]
- Davis and Wells 1997 Davis, N. R., and M. N. Wells, 1997, *High-Level Waste System Plan Revision 8*, HLW-OVP-97-0068, High-Level Waste Management Division, Westinghouse Savannah River Company, Aiken, South Carolina. [243431]
- Dirkmaat 1997a Dirkmaat, P. J., 1997a, "Repository Environmental Impact Statement (EIS) Data Call (OPE-SFP-97-230)," interoffice memorandum with attachment to K. Skipper (Office of Civilian Radioactive Waste Management), U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho. [MOL.19970725.0067, correspondence; MOL.19970725.0068, attachment]
- Dirkmaat 1997b Dirkmaat, P. J., 1997b, "Revision 1 Response to Repository Environmental Impact Statement (EIS) Data Call, OPE-SFP-97-336," U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho. [MOL.19971119.0151, letter; MOL.19971119.0152, attachment]
- Dirkmaat 1998a Dirkmaat, P. J., 1998a, "Response to Repository Environmental Impact Statement (EIS) Draft Appendix A. Review Action Items; Additional Comments," interoffice memorandum to K. Skipper (Yucca Mountain Site Characterization Office), Idaho Operations Office, U.S. Department of Energy, Idaho Falls, Idaho. [241196]
- Dirkmaat 1998b Dirkmaat, P. J., 1998b, "Response to U.S. Department of Energy (DOE) Appendix A MEPAS Input Parameters Review, and Miscellaneous [*sic*] Data – Yucca Mountain Repository Environmental Impact Statement (EIS)," interoffice memorandum to K. Skipper (Office of Civilian Radioactive Waste Management), OPE-SFP-98-171, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho. [MOL.19990511.0295]

- DOE 1985 DOE (U.S. Department of Energy), 1985, *An Evaluation of Commercial Repository Capacity for the Disposal of Defense High-Level Waste*, DOE/DP/0020/1, Director of Defense Waste and Byproducts, Deputy Assistant Secretary for Nuclear Materials, Assistant Secretary for Defense Programs, Washington, D.C. [235263]
- DOE 1992 DOE (U.S. Department of Energy), 1992, *Characteristics of Potential Repository Wastes*, DOE/RW-0184-R1, Oak Ridge National Laboratory, Oak Ridge, Tennessee. [HQO.19920827.0001, Volume 1; HQO.19920827.0002, Volume 2; HQO.19920827.0003, Volume 3; HQO.19920827.0004, Volume 4]
- DOE 1994 DOE (U.S. Department of Energy), 1994, *Greater-Than-Class C Low-Level Radioactive Waste Characterization: Estimated Volumes, Radionuclide Activities, and Other Characteristics*, DOE/LLW-114, Revision 1, Idaho National Engineering Laboratory, Idaho Falls, Idaho. [231330]
- DOE 1995a DOE (U.S. Department of Energy), 1995a, *Acceptance Priority Ranking and Annual Capacity Report*, DOE/RW-0457, Office of Civilian Radioactive Waste Management, Washington, D.C. [MOV.19960910.0021]
- DOE 1995b DOE (U.S. Department of Energy), 1995b, *Record of Decision – Department of Energy Programmatic Spent Nuclear Fuel Management and the Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs*, Office of Environmental Management, Idaho Operations Office, Idaho Falls, Idaho. [243787]
- DOE 1995c DOE (U.S. Department of Energy), 1995c, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs: Final Environmental Impact Statement*, DOE/EIS-0203-F, Office of Environmental Management, Idaho Operations Office, Idaho Falls, Idaho. [102617]
- DOE 1996 DOE (U.S. Department of Energy), 1996, “Amendment to the Record of Decision for the Department of Energy (DOE) Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory (INEL) Environmental Restoration and Waste Management Programs Environmental Impact Statement (EIS),” Idaho Operations Office, Idaho Falls, Idaho. [243792]
- DOE 1997a DOE (U.S. Department of Energy), 1997a, *Nuclear Power Generation and Fuel Cycle Report 1997*, DOE/EIA-0436 (97), Office of Coal, Nuclear, Electric and Alternate Fuels, Energy Information Administration, Washington, D.C. [243940]

- DOE 1997b DOE (U.S. Department of Energy), 1997b, *Integrated Data Base for 1996: U.S. Spent Nuclear Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, DOE/RW-0006, Revision 13, Office of Environmental Management, Oak Ridge National Laboratories, Oak Ridge, Tennessee. [242471]
- DOE 1998a DOE (U.S. Department of Energy), 1998a, *Surplus Plutonium Disposition Draft Environmental Impact Statement*, DOE/EIS-0283-D, Office of Fissile Materials Disposition, Washington, D.C. [243236]
- DOE 1998b DOE (U.S. Department of Energy), 1998b, *Viability Assessment of a Repository at Yucca Mountain*, DOE/RW-0508, Office of Civilian Radioactive Waste Management, Washington, D.C. [U.S. Government Printing Office, MOL.19981007.0027, Overview; MOL.19981007.0028, Volume 1; MOL.19981007.0029, Volume 2; MOL.19981007.0030, Volume 3; MOL.19981007.0031, Volume 4; MOL.19981007.0032, Volume 5]
- DOE 1998c DOE (U.S. Department of Energy), 1998c, *Preliminary Design Specification for Department of Energy Standardized Spent Nuclear Fuel Canisters, Volume 1 – Design Specification*, DOE/SNF/REP-011, Revision 0, Office of Spent Fuel Management and Special Projects, Assistant Secretary for Environmental Management, Idaho Operations Office, Idaho Falls, Idaho. [240539]
- DOE 1998d DOE (U.S. Department of Energy), 1998d, *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement*, DOE/EIS-0279-D, Savannah River Operations Office, Aiken, South Carolina. [243456]
- DOE 1999 DOE (U.S. Department Of Energy), 1999, *Supplement to the Surplus Plutonium Disposition Draft Environmental Impact Statement*, DOE/EIS-0283-DS, Office of Fissile Materials Disposition, Washington, D.C. [244066]
- Dreyfus 1995 Dreyfus, D. A., 1995, "Proposed Mix Of DOE-Owned High Level Waste And Spent Nuclear Fuel," interoffice memorandum to J. E. Lytle (Office of Environmental Management), November 9, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, Washington, D.C. [MOL.19990319.0341]
- Fillmore 1998 Fillmore, D. L., 1998, *Parameter Selection For Department of Energy Spent Nuclear Fuel To Be Used in the Yucca Mountain Viability Assessment*, INEEL/EXT-98-00666, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Corporation, Idaho Falls, Idaho. [MOL.19990511.0296]
- Fowler et al. 1995 Fowler, J. R., R. E. Edwards, S. L. Marra, and M. J. Plodinec, 1995, *Chemical Composition Projections for the DWPF Product (U)*, WSRC-IM-91-116-1, Revision 1, Westinghouse Savannah River Company, Aiken, South Carolina. [232731]

- Goff 1998a Goff, K. M., 1998a, "Revision to Original INEEL Response to Yucca Mountain Site Characterization Office Data Call for High-Level Waste," memorandum to M. B. Heiser (Lockheed Martin Idaho Technologies Corporation), Argonne National Laboratory-West, Idaho Falls, Idaho. [MOL.19990608.0032]
- Goff 1998b Goff, K. M., 1998b, "ANL-West Comments from Review of Appendix A - Yucca Mountain Repository Environmental Impact Statement," memorandum to M. B. Heiser (Lockheed Martin Idaho Technologies Corporation), Argonne National Laboratory-West, Idaho Falls, Idaho. [MOL.19990511.0377]
- Heath 1998 Heath, C. A., 1998, "DE-AC08-91RW00134; OCRWM Fiscal Year 1998 Annual Work Plan ...," letter to D. Shelor (Office of Civilian Radioactive Waste Management, U.S. Department of Energy), September 24, TRW Environmental Safety Systems Inc., Vienna, Virginia. [MOV.19981005.0009]
- Heiser 1998 Heiser, M. B., 1998, "INEL HLW vit Breakdown," facsimile to J. Rivers (Jason Associates, Inc.), March 5, Lockheed Martin Idaho Technologies Corporation, Idaho Falls, Idaho. [MOL.19990511.0370]
- Knecht et al. 1999 Knecht, D. A., J. H. Valentine, A. J. Luptak, M.D. Staiger, H. H. Loo, and T. L. Wichmann, 1999, *Options for Determining Equivalent MTHM for DOE High-Level Waste*, INEEL/EXT-99-00317, Revision 1, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho. [244063]
- LMIT 1997 LMIT (Lockheed Martin Idaho Technologies Corporation), 1997, *DOE National Spent Nuclear Fuel Database*, Version 3.2, Idaho Falls, Idaho. [DTN: M09906DOESFVER32.000]
- LMIT 1998 LMIT (Lockheed Martin Idaho Technologies Corporation), 1998, *Accelerating Cleanup: Paths to Closure, Idaho Operations Office*, PNL-177, Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho. [243437]
- Lytle 1995 Lytle, J. E., 1995, "Disposal of DOE-owned High Level Waste and Spent Nuclear Fuel," interoffice memorandum to D. A. Dreyfus (Office of Civilian Radioactive Waste Management), Office of Environmental Management, U.S. Department of Energy, Washington, D.C. [HQO.19951116.0015]
- Marra, Harbour, and Plodinec 1995 Marra, S. L., J. R. Harbour, and M. J. Plodinec, 1995, *DWPF Canister Procurement, Control, Drop-Test, and Closure (U)*, WSRC-IM-91-116-8, Revision 1, Westinghouse Savannah River Company, Aiken, South Carolina. [240797]
- Murphy 1998 Murphy, B. D., 1998, "EIS, Jason requests," internal memorandum to K. A. Williams, June 4, Computational Physics and Engineering Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee. [MOL.19990511.0288]

- Palmer 1997 Palmer, W. B., 1997, "Clarification to Yucca Mountain Site Characterization Office Data Call for High Level Waste-WBP-13-97," memorandum to T. L. Wichman (Idaho Operations Office, U.S. Department of Energy), November 13, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho. [MOL.19990526.0031]
- Parsons 1999 Parsons Infrastructure and Technology Group, Inc., 1999, *Multi-Canister Overpack Fabrication Specification*, HNF-S-0453, Revision 3, Richland, Washington. [243785]
- Pearson 1997 Pearson, W. D., 1997, "Repository Environmental Impact Statement (EIS) Data Call for High-Level Waste (HLW)," memorandum to K. G. Picha (Office of Planning and Analysis), October 22, Savannah River Operations Office, U.S. Department of Energy, Aiken, South Carolina. [MOL.19990303.0336]
- Pearson 1998 Pearson, W. D., 1998, "SRS Data Request Followup," electronic communication to J. Rivers (Jason Associates Corporation), February 18, U.S. Department of Energy, Savannah River Site, Aiken, South Carolina. [MOL.19990511.0281]
- Person 1998 Person, R., 1998, "Status of MOx in RFP," memorandum to J. Rivers (Jason Associates Corporation), May 4, U.S. Department of Energy, Office of Fissile Material Disposition, Washington, D.C. [MOL.19990511.0286]
- Picha 1997 Picha, K. G., Jr., 1997, "Response to Repository Environmental Impact Statement Data Call for High-Level Waste," interoffice memorandum to W. Dixon (Yucca Mountain Site Characterization Office), September 5, Office of Waste Management, U.S. Department of Energy, Washington, D.C. [MOL.19970917.0273]
- Picha 1998a Picha, K. G., Jr., 1998a, "Clarification of High-Level Waste and Special Performance Assessment Required Data for Repository Environmental Impact Statement," interoffice memorandum with attachments to K. Skipper (Yucca Mountain Site Characterization Office), May 8, Office of Waste Management, U.S. Department of Energy, Washington, D.C. [MOL.19990610.0297]
- Picha 1998b Picha, K. G., Jr., 1998b, "Special Performance Assessment Required Waste Supplement for the Yucca Mountain Repository Environmental Impact Statement," interoffice memorandum with attachments to W. Dixon (Yucca Mountain Site Characterization Office), May 8, Office of Waste Management, U.S. Department of Energy, Washington, D.C. [MOL.19990319.0331, correspondence; MOL.19990319.0332, attachment]
- Picha 1998c Picha, K. G., Jr., 1998c, "Follow Up Response to Repository EIS Data Call for High-Level Waste," interoffice memorandum to W. Dixon, U.S. Department of Energy, Office of Waste Management, Washington, D.C. [MOL.19981006.0206]

- Plodinec and Marra 1994 Plodinec, M. J., and S. L. Marra, 1994, *Projected Radionuclide Inventories and Radiogenic Properties of the DWPF Product (U)*, WSRC-IM-91-116-3, Revision 0, Westinghouse Savannah River Company, Aiken, South Carolina. [242337]
- Plodinec, Moore, and Marra 1993 Plodinec, M. J., F. S. Moore, and S. L. Marra, 1993, *Reporting Dose and Heat Generation Rates of the DWPF Product (U)*, WSRC-IM-91-116-12, Revision 0, Westinghouse Savannah River Company, Aiken, South Carolina. [232736]
- Raddatz and Waters 1996 Raddatz, M. G., and M. D. Waters, 1996, *Information Handbook on Independent Spent Fuel Storage Installations*, NUREG-1571, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. [231666]
- Rowland 1997 Rowland, T. J., 1997, "Repository Environmental Impact Statement Data Call for High-Level Waste," interoffice memorandum with Attachment A to K. G. Picha (Office of Waste Management), November 26, West Valley Demonstration Project, U.S. Department of Energy, West Valley, New York. [MOL.19990608.0048]
- Ryman, Hermann, and Murphy 1998 Ryman, J. C., O. W. Hermann, and B. D. Murphy, 1998, *Characteristics of Spent Fuel from Plutonium Disposition Reactors*, Volumes 2 and 3, ORNL/TM-13170/V2 and V3, Computational Physics and Engineering Division, Oak Ridge National Laboratory, Oak Ridge, Tennessee. [239236, Volume 2; 237138, Volume 3]
- Stevenson 1997 Stevenson, B., 1997, "Delivery of Data Reports," interoffice memorandum to W. Dixon (Yucca Mountain Site Characterization Office), U.S. Department of Energy, Office of Fissile Materials Disposition, Washington, D.C. [MOL.19971119.0155]
- Taylor 1997 Taylor, W. J., 1997, "Response to Clarification Data for the Repository Environmental Impact Statement (EIS) Data Call Memorandum Dated October 3, 1997," interoffice memorandum to K. J. Picha (Office of Waste Management), November 17, U.S. Department of Energy, Richland Operations Office, Richland, Washington. [MOL.19990610.0295]
- TRW 1997 TRW (TRW Environmental Safety Systems Inc.), 1997, *Waste Quantity, Mix and Throughput Study Report*, B00000000-01717-5705-00059, Revision 01, TRW, Las Vegas, Nevada. [MOL.19971210.0628]
- TRW 1998 TRW (TRW Environmental Safety Systems Inc.), 1998, *Controlled Design Assumptions Document*, B00000000-01717-4600-00032, Revision 05, Las Vegas, Nevada. [MOL.19980804.0481]
- USN 1996 USN (U.S. Navy), 1996, *Department of the Navy Final Environmental Impact Statement for a Container System for the Management of Naval Spent Nuclear Fuel*, DOE/EIS-0251, in cooperation with the U.S. Department of Energy, Naval Nuclear Propulsion Program, U.S. Department of the Navy, U.S. Department of Defense, Arlington, Virginia. [227671]

WVNS 1996

WVNS (West Valley Nuclear Services, Inc.), 1996, *WVDP Waste Form Qualification Report*, WVDP-186, Revision 1, West Valley, New York. [242094]



Appendix B

Federal Register Notices

(NEPA) of 1969 (42 U.S.C. § 4321 *et seq.*), the Council on Environmental Quality regulations that implement the procedural provisions of NEPA (40 CFR Parts 1500-1508), and the DOE procedures for implementing NEPA (10 CFR Part 1021). DOE invites Federal, State, and local agencies, Native American tribal organizations, and other interested parties to participate in determining the scope and content of the EIS.

The NWSA directs DOE to evaluate the suitability of the Yucca Mountain site in southern Nevada as a potential site for a geologic repository for the disposal of spent nuclear fuel and high-level radioactive waste. If the Secretary of Energy determines that the Yucca Mountain site is suitable, the Secretary may then recommend that the President approve the site for development of a repository. Under the NWSA, any such recommendation shall be considered a major Federal action and must be accompanied by a final environmental impact statement. Accordingly, DOE is preparing this EIS in conjunction with any potential DOE recommendation regarding the development of a repository at Yucca Mountain.

The NWSA provides that the environmental impact statement need not consider the need for a repository, the alternatives to geologic disposal, or alternative sites to the Yucca Mountain site. Therefore, this environmental impact statement will evaluate a proposal to construct, operate, and eventually close a repository at Yucca Mountain. The EIS will evaluate reasonable alternatives for implementing such a proposal in accordance with the NWSA.

The NWSA also provides that the Nuclear Regulatory Commission shall, to the extent practicable, adopt DOE's EIS in connection with any subsequent construction authorization and license that the Commission issues to DOE for a repository. The EIS process is scheduled to be completed in September 2000 and is separate from the licensing process that would be initiated by any submission of a license application by DOE to the Commission in June 2001.

The EIS will be prepared over a five-year period in conjunction with DOE's separate but parallel site suitability evaluation and potential license application. DOE is beginning the EIS process early to ensure that the appropriate data gathering and tests are performed to adequately assess potential environmental impacts, and to allow the public sufficient time to consider this complex program and to provide input.

DATES: DOE invites and encourages comments and suggestions on the scope of the EIS to ensure that all relevant environmental issues and reasonable alternatives are addressed. Public scoping meetings are discussed below in the **SUPPLEMENTARY INFORMATION** section. DOE will carefully consider all comments and suggestions received during the 120-day public scoping period that ends on December 5, 1995. Comments and suggestions received after the close of the public scoping period will be considered to the extent practicable.

ADDRESSES: Written comments on the scope of this EIS, requests to pre-register to speak at any of the public scoping meetings, questions concerning the proposed action and EIS, or requests for additional information on the EIS, should be directed to: Wendy R. Dixon, EIS Project Manager, Yucca Mountain Site Characterization Office, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, 101 Convention Center Drive Suite P-110, MS 010, Las Vegas, NV 89109. Telephone: 1-800-967-3477. Facsimile: 1-800-967-0739.

FOR FURTHER INFORMATION CONTACT: For more information about this EIS, please contact Wendy R. Dixon at the address above. For information on DOE's NEPA process, please contact: Carol M. Borgstrom, Director, Office of NEPA Policy and Assistance (EH-42), U.S. Department of Energy, 1000 Independence Avenue, S.W., Washington, D.C. 20585. Telephone: 1-202-586-4600 or leave a message at 1-800-472-2756.

SUPPLEMENTARY INFORMATION:

Public Participation

All interested persons, including Federal agencies, Native American tribal organizations, State and local government agencies, public interest groups, transportation interests, industry and utility organizations, regulators, and the general public are encouraged to take part in the EIS scoping process. Because of the anticipated public interest and national scope of the program, DOE will provide several methods for people to express their views and provide comments, request additional information and copies of the EIS, or pre-register to speak at the scoping meetings. Comments submitted by any of these means will become part of the official record for scoping.

DEPARTMENT OF ENERGY

Preparation of an 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

AGENCY: Department of Energy.

ACTION: Notice of intent.

SUMMARY: The U.S. Department of Energy (DOE) announces its intent to prepare an environmental impact statement (EIS) for a geologic repository at Yucca Mountain, Nye County, Nevada, for the disposal of spent nuclear fuel and high-level radioactive waste, in accordance with the Nuclear Waste Policy Act of 1982, as amended (NWSA) (42 U.S.C. § 10101 *et seq.*), the National Environmental Policy Act

Written Comments and Toll-Free Facsimile Number

Written comments and requests may be mailed or sent by facsimile to Wendy R. Dixon at the address or toll-free facsimile number listed above

Toll-Free Telephone Line

All interested parties are invited to record their comments or request information on the scope of the EIS by calling a toll-free telephone number, 1-800-967-3477. Throughout the public scoping period, this number will be staffed between the hours of 9 a.m. to 9 p.m. Eastern Standard Time, Monday through Friday. During other hours, calls will be forwarded to an answering machine.

Electronic Mail

Comments and information requests may be submitted by electronic mail to the following Internet electronic mail address: ymp—eistr@notes.ymp.gov.

Internet

The public may access the Notice of Intent, request information, and provide comments via the World Wide Web at the following Uniform Resource Locator address: <http://www.ymp.gov>, under the listing *Environmental Impact Statement (EIS)* on the Yucca Mountain Project Home Page. When available, the EIS and other selected technical documents may also be accessed at this Uniform Resource Locator address.

Scoping Meetings

DOE will hold 15 public scoping meetings in cities throughout the United States to provide and discuss information and to receive comments on the scope of this EIS. Table 1 at the end of this Notice lists the specific locations, dates, and times for each scoping meeting. Persons wishing to speak at any of these meetings can pre-register up to two days before the meeting by: (1) Calling the toll-free telephone number 1-800-967-3477, (2) writing to Wendy R. Dixon at the address listed above, or (3) sending their request to pre-register by facsimile or electronic mail, as identified above.

Persons wishing to speak who have not registered in advance can register at each meeting. These "walk-in registrants" will be accommodated to the extent practicable, following those persons who have pre-registered. Only one spokesperson per organization, group, or agency may present comments on its behalf. Oral statements will be limited to ten minutes; however, written comments can be of any length and submitted any time during the scoping period.

Each of the 15 public scoping meetings will have either a morning or afternoon session, and an evening session. Morning sessions will begin at 8:30 a.m. and end at 12:30 p.m., and afternoon sessions will begin at 12:00 p.m. and end at 4:00 p.m. Evening sessions will begin at 6:00 p.m. and end about 10:00 p.m. If additional time is required in order to accommodate all speakers wishing to present oral comments, the meeting facilitator will consult with the audience and DOE staff and determine whether to continue the meeting past the scheduled ending time. A court reporter will record all portions of the scoping meetings, and transcripts will be prepared and made a part of the official record of the scoping process.

Each session will have an introductory presentation, a question and answer period, and a public comment segment. A facilitator will begin the introductory presentation of each session by explaining the scoping meeting format. DOE staff will provide a brief description (lasting approximately 30-45 minutes) of the repository program, the EIS, and the scoping process. The question and answer period (lasting approximately 45 minutes) will provide members of the public an opportunity to ask questions and discuss various aspects of the repository and to obtain additional information that may be useful in formulating opinions and comments. Each member of the public will be allowed five minutes to ask questions. The meeting facilitator may allow extra time for additional questions depending on the number of people present who have indicated their desire to participate during the question and answer period. The meeting facilitator will begin the public comment portion of the scoping meeting after the question and answer period. At this time, members of the public will provide their comments on the scope of the EIS.

Each public scoping meeting also will have a separate information room containing exhibits and informational handouts about the repository program and the EIS. DOE and contractor staff will be available throughout the day to answer questions in an informal setting. A table with blank comment cards will also be available for people to privately prepare and submit written comments on the scope of the EIS. These comment cards will be included in the formal record of each scoping meeting.

Subsequent Document Preparation

Results of scoping, including the transcripts from the question and answer periods and public comment segments, and all other oral and written

comments received by DOE, will be summarized in the EIS Implementation Plan. This Plan will guide the preparation of the EIS, and will describe the planned scope and content of the EIS, record the results of the scoping process, and contain EIS activity schedules. As a "living document," the Implementation Plan may be amended as needed to incorporate changes in schedules, alternatives, or EIS content.

The Implementation Plan will be available to the public for information purposes as soon as possible after the close of the public scoping process, and before issuing the Draft EIS. The Implementation Plan and the transcripts from the public scoping meetings will be available for inspection at major DOE facilities and public reading rooms in Nevada and across the country, as identified at the end of this Notice. Copies of the Implementation Plan, as well as the Draft and Final EIS and related comments, will be provided to anyone requesting copies of these documents.

Availability of the Draft EIS for public review, and the locations and times of public hearings on the Draft EIS, will be announced in the *Federal Register* and through local media (approximately in the Fall of 1998). After considering public comments received on the Draft EIS, DOE will prepare and issue a Final EIS, followed thereafter by a Record of Decision (approximately in the Fall of 2000).

Background

Spent nuclear fuel¹ has been and is being generated and stored in the United States as part of commercial power generation. The accumulation of spent nuclear fuel from commercial power reactor operations in the United States probably will continue for several decades. There are 109 operating commercial facilities at 75 sites in 34 States where spent nuclear fuel is stored. By the year 2035, total spent nuclear fuel from power reactors will amount to about 85,000 metric tons of heavy metal (i.e., metric tons of heavy metal, typically uranium, without materials such as cladding, alloy and structural materials) (MTHM).

Spent nuclear fuel and high-level radioactive waste², generated from

¹ Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing.

² High-level radioactive waste is the highly radioactive material resulting from reprocessing of spent nuclear fuel. It includes liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient

DOE's national atomic energy defense and research activities, are primarily located at DOE's Hanford Reservation, the Savannah River Site, and the Idaho National Engineering Laboratory. Other spent nuclear fuel, either currently in DOE possession or which may come under DOE possession, includes material from foreign research reactors, approximately 29 domestic university reactors, 5 non-DOE research reactors, and 4 "special case" reactors at non-DOE locations.

In 1982, in response to the continued accumulation of spent nuclear fuel and high-level radioactive waste, Congress passed the NWSA. The purpose of the NWSA was to establish geologic repositories that would provide reasonable assurance that the public and the environment would be adequately protected from the hazards posed by these materials. In 1987, Congress amended the NWSA and directed DOE to evaluate the suitability of only the Yucca Mountain site in southern Nevada as a potential site for the first repository. If, based on this evaluation, the Secretary of Energy determines that the Yucca Mountain site is suitable, the Secretary may then recommend that the President approve the site for development of a repository.

Under the NWSA, DOE is prohibited from emplacing more than 70,000 MTHM of spent nuclear fuel and high-level radioactive waste in the first repository until such time as a second repository is in operation. The current planning basis calls for 63,000 MTHM of commercial spent nuclear fuel to be disposed of in the first repository, proposed to be located at the Yucca Mountain site. The planning basis also calls for the disposal of 7,000 MTHM equivalent of DOE-owned spent nuclear fuel and high-level radioactive waste in this first repository.

Proposed Action

If the site were found to be suitable, the proposed action would be to construct, operate, and eventually close a repository at Yucca Mountain for the geologic disposal of up to 70,000 MTHM of commercial and DOE-owned spent nuclear fuel and high-level radioactive waste. Spent nuclear fuel and high-level radioactive waste would be disposed of in the repository in a subsurface configuration that would ensure its long-term isolation from the human environment. Repository construction, operation, and closure would be

concentrations and other highly radioactive material that the Nuclear Regulatory Commission, consistent with existing law, determines by rule requires permanent isolation.

governed by the Nuclear Regulatory Commission's licensing process.

Construction would begin if the Nuclear Regulatory Commission authorizes construction of the repository. Surface facilities would be designed and constructed to receive, and prepare for disposal, spent nuclear fuel and high-level radioactive waste that would arrive in transportation casks by highway and by rail. Capability to treat or package the secondary wastes generated during disposal operations would also be provided. Subsurface facilities would be designed and constructed for emplacement of spent nuclear fuel and high-level radioactive waste in disposal drifts. Subsurface facilities would primarily include access ramps, ventilation systems, disposal drifts, and equipment alcoves.

Disposal operations would begin once the Nuclear Regulatory Commission issues a license allowing receipt of spent nuclear fuel and high-level radioactive waste. Disposal operations would be expected to last up to 40 years, depending on shipment schedules. Disposal drifts would continue to be constructed during this time period as necessary. Spent nuclear fuel assemblies,³ and canisters containing assemblies⁴ or vitrified (i.e., solidified) high-level radioactive waste⁵ would be shipped to the repository in transportation casks that meet the Nuclear Regulatory Commission and U.S. Department of Transportation requirements for shipping by truck or rail⁶. The assemblies would be removed from the transportation casks, which would be placed back into service after decontamination and maintenance or after necessary repairs were completed. Canisters and assemblies would be transferred to a "hot" cell—a room where remotely-controlled equipment would be used to place the material in disposal containers. These "waste packages" (i.e., assemblies and canisters

³ A fuel assembly is made up of fuel elements held together by plates and separated by spacers attached to the fuel cladding.

⁴ Under one scenario, spent nuclear fuel assemblies would be sealed in a multi-purpose canister that would then be inserted into separate casks/containers for storage, transportation, and disposal. Other canisters are available and include single-purpose systems, which require transferring of individual assemblies from one cask/container to another for storage, transport, and disposal. Another alternative would be dual-purpose systems which require storing and transporting individual assemblies in one cask and disposing of them in another container.

⁵ Vitrified high-level radioactive waste would be sealed in canisters suitable for transport in a truck or train cask.

⁶ Barges may also be used for intermodal shipments of spent nuclear fuel and high-level radioactive waste from generator sites to nearby locations for transfer to truck and rail.

in disposal containers) would be transported underground in a transportation vehicle having radiation shielding for worker protection. Monitoring equipment, which would either be placed in selected drifts or would be mobile remote-sensing devices, would monitor performance of waste packages and aspects of the local repository geology.

The closure/post-closure period would begin after the Nuclear Regulatory Commission amends the license to authorize permanent closure. Underground equipment would be removed, repository openings would be backfilled and sealed, and the surface facilities would be decontaminated, decommissioned, and dismantled or converted to other uses. Institutional controls, such as permanent markers and monuments, would be designed and constructed to last thousands of years and discourage human activities that could compromise the waste isolation capabilities of the repository.

The disposal and closure/post-closure activities would be designed and implemented so that the combination of engineered (i.e., waste package and any backfill) and natural (geologic system) barriers would isolate the spent nuclear fuel and high-level radioactive waste. The combination of barriers would meet a standard to be specified by the Environmental Protection Agency, which has been entrusted to develop a radiation release standard pursuant to Section 801 of the Energy Policy Act of 1992 (42 U.S.C. § 10141 note); individual barriers would perform according to Nuclear Regulatory Commission requirements, including its performance objectives at 10 CFR 60.113. The engineered barrier must provide substantially complete containment of spent nuclear fuel and high-level radioactive waste for between 300 and 1,000 years by using corrosion resistant materials in the waste package.

Beyond 1,000 years, continued isolation would be assisted by features that would limit the rate at which radioactive components of the waste would be released. The rate of release would be substantially affected by natural conditions, the heat generation rate of spent nuclear fuel and high-level radioactive waste (i.e., thermal load), and its rate of heat dissipation. First, different thermal loads would affect directly the internal and external waste package temperatures, thereby affecting the corrosion rate and integrity of the waste package. Second, the heat would affect the geochemistry, hydrology, and mechanical stability of the disposal drifts, which in turn would influence the flow of groundwater and the

transport of radionuclides from the engineered and natural barrier systems to the environment. Therefore, the long-term performance of the repository would be managed by appropriately spacing the waste packages within disposal drifts and the distances between disposal drifts, and by selectively placing spent nuclear fuel and high-level radioactive waste packages to account for their individual heat generation rates.

Alternatives

DOE has preliminarily identified for analysis in the EIS a full range of reasonable implementation alternatives for the construction, operation, and closure/post-closure of a repository at Yucca Mountain. These implementation alternatives are based on thermal load objectives and include High Thermal Load, Intermediate Thermal Load, and Low Thermal Load alternatives.

Under each implementation alternative, DOE will evaluate different spent nuclear fuel and high-level radioactive waste packaging and transportation options. DOE anticipates that these options would produce the broadest range of potential configurations for both surface facilities and possible operational and disposal conditions at the repository. Evaluation of these options will identify the full range of reasonably foreseeable impacts to human health and the environment associated with each implementation alternative.

High Thermal Load Alternative

Under the High Thermal Load implementation alternative, spent nuclear fuel and high-level radioactive waste would be disposed in an underground configuration that would generate the upper range of repository temperatures while meeting performance objectives to isolate the material in compliance with Environmental Protection Agency standards and Nuclear Regulatory Commission requirements. Under this alternative, the emplacement density would likely be greater than 80 MTHM per acre. This alternative would represent the highest repository thermal loading based on available information and expected test results.

Intermediate Thermal Load Alternative

Under the Intermediate Thermal Load implementation alternative, spent nuclear fuel and high-level radioactive waste would be disposed in an underground configuration that would generate an intermediate range of repository temperatures (compared to the High and Low Thermal Load

alternatives) while meeting performance objectives to isolate the material in compliance with Environmental Protection Agency standards and Nuclear Regulatory Commission requirements. Under this alternative, the disposal density would likely range between 40 to 80 MTHM per acre.

Low Thermal Load Alternative

Under the Low Thermal Load implementation alternative, spent nuclear fuel and high-level radioactive waste would be disposed in an underground configuration that would provide the lowest potential repository thermal loading (based on available information and expected test results) while meeting performance objectives to isolate the material in compliance with Environmental Protection Agency standards and Nuclear Regulatory Commission requirements. Under this alternative, the disposal density would likely be less than 40 MTHM per acre.

Packaging Options

As part of each implementation alternative, two packaging options would be evaluated. Under Option 1, spent nuclear fuel assemblies would be packaged and sealed in multi-purpose canisters at the generator sites prior to being transported to the repository in Nuclear Regulatory Commission-certified casks. High-level radioactive waste also would be packaged and sealed in canisters prior to shipment in similar casks. Under Option 2, spent nuclear fuel assemblies (without canisters) and sealed canisters of high-level radioactive waste would be transported to the repository in Nuclear Regulatory Commission-certified casks. Under both options, assemblies and canisters with intact seals would be removed from the casks and placed in disposal containers at the repository.

DOE recognizes that it is likely that a mix of spent nuclear fuel assemblies and canisters (and canister systems) of spent nuclear fuel and vitrified high-level radioactive waste would arrive at the repository during disposal operations. However, since the specific mix is speculative, the above packaging options were chosen to produce the broadest range of potential configurations for both surface facilities and possible operational and disposal conditions at the repository. These options were also selected to reflect the potential range of exposures to workers and the public at the generator sites, along transportation routes, and at the repository from the packaging, transport, and disposal of spent nuclear fuel and high-level radioactive waste.

Transportation

As part of each implementation alternative, two national transportation options and three regional (i.e., within the State of Nevada) transportation options would be evaluated. These options would be expected to result in the broadest range of operating conditions relevant to potential impacts to human health and the environment.

In a national context, the first option would consist of shipping all spent nuclear fuel and high-level radioactive waste by truck, from the generator site to the repository.

The second national option would consist of shipment by rail, except from those generator sites (as many as 19) that may not have existing capabilities to load and ship rail casks. For such sites, the spent nuclear fuel would be transported by truck to the repository, or to a facility near the nuclear power plant where it would be transferred to rail cars for shipment to the repository.

In a regional context, there are three transportation options: two of these options apply to shipments that would arrive in Nevada by rail, and the third applies to shipments that would arrive in Nevada by legal weight truck.⁷

The first regional transportation option would consist of several rail corridors to the repository. The rail corridor option would involve identifying and applying siting criteria, based on engineering considerations (e.g., topography and soils), potential land use restrictions (e.g., wilderness areas and existing conflicting uses), and any other factors identified from the scoping process.

The second regional transportation option would involve the use of heavy haul truck⁸ routes to the repository. The heavy haul option would include the construction and use of an intermodal transfer facility to receive shipments that would arrive in Nevada by rail; the intermodal transfer facility would be located at the beginning of the heavy haul route. The heavy haul option would include any need to improve the local transportation infrastructure.

The third regional transportation option would involve legal weight truck shipments directly to the repository. Under this option, a transfer facility would not be required.

No Action

The No Action alternative would evaluate termination of site

⁷ A legal weight truck consists of a tractor, semi-trailer, and loaded cask, with a maximum gross weight of 80,000 pounds.

⁸ A heavy haul truck consists of a tractor, semi-trailer, and loaded cask, with a gross weight in excess of 129,000 pounds.

characterization activities at Yucca Mountain and the continued accumulation of spent nuclear fuel and high-level radioactive waste at commercial storage sites and DOE facilities. Spent nuclear fuel and high-level radioactive waste would continue to be managed for the foreseeable future at existing commercial storage sites and DOE facilities located in 34 States. The No Action alternative, although contrary to the Congressional desire to provide a permanent solution for isolation of the Nation's spent nuclear fuel and high-level radioactive waste, provides a baseline against which the implementation alternatives can be compared.

At the Yucca Mountain site, the surface facilities, excavation equipment, and other support facilities would be dismantled and removed for reuse or recycling, or would be disposed of in solid waste landfills. Disturbed surface areas would be reclaimed and excavated openings to the subsurface would be sealed and backfilled.

At commercial reactors, spent nuclear fuel would continue to be generated and stored in either water pools or in canisters, until storage space at individual reactors becomes inadequate, which time reactor operations would cease. DOE-owned spent nuclear fuel and high-level radioactive waste would continue to be managed at three primary sites—the Hanford Reservation, Savannah River Site, and the Idaho National Engineering Laboratory.

Environmental Issues To Be Examined in the EIS

This EIS will examine the site-specific environmental impacts from construction, operation, and eventual closure of a repository for spent nuclear fuel and high-level radioactive waste disposal at Yucca Mountain, Nevada. Transportation-related impacts of the alternatives will also be analyzed. Through internal discussion and outreach programs with the public, DOE is aware of many environmental issues related to the construction, operation, and closure/post-closure phases of such a repository. The issues identified here are intended to facilitate public scoping. The list is not intended to be all-inclusive or to predetermine the scope of the EIS, but should be used as a starting point from which the public can

DOE define the scope of the EIS.

Radiological and non-radiological releases. The potential effects to the public and on-site workers from radiological and nonradiological releases:

- Public and Worker Safety and Health. Potential health and safety

impacts (e.g., injuries) to on-site workers during the unloading, temporary surface storage, and underground emplacement of waste packages at Yucca Mountain;

- Transportation. The potential impacts associated with national and regional shipments of spent nuclear fuel and high-level radioactive waste from reactor sites and DOE facilities to the Yucca Mountain site will be assessed. Regional transportation issues include: (a) technical feasibility, (b) socioeconomic impacts, (c) land use and access impacts, and (d) impacts of constructing and operating a rail spur, a heavy haul route, and/or a transfer facility;

- Accidents. The potential impacts from reasonably foreseeable accidents, including any accidents with low probability but high potential consequences;

- Criticality. The likelihood that a self-sustaining nuclear chain reaction could occur and its potential consequences;

- Waste Isolation. Potential impacts associated with the long-term performance of the repository;

- Socioeconomic Conditions. Potential regional (i.e., in Nevada) socioeconomic impacts to the surrounding communities, including impacts on employment, tax base, and public services;

- Environmental Justice. Potential for disproportionately high and adverse impacts on minority or low-income populations;

- Pollution Prevention. Appropriate and innovative pollution prevention, waste minimization, and energy and water use reduction technologies to eliminate or significantly reduce use of energy, water, hazardous substances, and to minimize environmental impacts;

- Soil, Water, and Air Resources. Potential impacts to soil, water quality, and air quality;

- Biological Resources. Potential impacts to plants, animals, and habitat, including impacts to wetlands, and threatened and endangered species;

- Cultural Resources. Potential impacts to archaeological/historical sites, Native American resources, and other cultural resources;

- Cumulative impacts from the proposed action and implementing alternatives and other past, present, and reasonably foreseeable future actions;

- Potential irreversible and irretrievable commitment of resources.

Under the No Action alternative, potential environmental effects associated with the shutdown of site characterization activities at Yucca Mountain will be estimated. Potential

environmental effects from the continued accumulation of spent nuclear fuel and high-level radioactive waste at commercial reactors and DOE sites will be addressed by summarizing previous relevant environmental analyses and by performing new analyses of representative sites, as appropriate. At the Yucca Mountain site, the potential environmental consequences from the reclamation of disturbed surface areas, and the sealing of excavated openings following the dismantlement and removal of facilities and equipment, will be quantified. These analyses would be similar in level of detail to the analyses of the implementing alternatives. At the commercial reactor and DOE sites, the potential environmental consequences will be addressed in terms of risk to the environment and the public from long-term management of spent nuclear fuel and high-level radioactive waste. In addition, the loss of storage capacity, the need for additional capacity, and their potential consequences to continued reactor operations, will be described.

Consultations With Other Agencies

The NWPA requires DOE to solicit comments on the EIS from the Department of the Interior, the Council on Environmental Quality, the Environmental Protection Agency, and the Nuclear Regulatory Commission (42 U.S.C. § 10134(a)(1)(D)). DOE also intends to consult with the Departments of the Navy and Air Force and will solicit comments from other agencies, the State of Nevada, affected units of local government, and Native American tribal organizations, regarding the environmental issues to be addressed by the EIS.

Relationship to Other DOE NEPA Reviews

DOE is preparing or has completed other NEPA documents that may be relevant to the Office of Civilian Radioactive Waste Management Program and this EIS. If appropriate, this EIS will incorporate by reference and update information taken from these other NEPA documents. These documents (described below) are available for inspection by the public at the DOE Freedom of Information Reading Room (1E-190), Forrestal Building, 1000 Independence Ave., S.W., Washington, D.C. and will be made available in Nevada at locations to be announced at the public scoping meetings. These documents include the following:

- *Environmental Assessment, Yucca Mountain Site*. Nevada Research and

Development Area, Nevada, DOE/RW-0073, 1986.

- *Environmental Assessment for a Monitored Retrievable Storage Facility*, DOE/RW-0035, 1986.

- *Environmental Impact Statement for a Multi-Purpose Canister System for the Management of Civilian and Naval Spent Nuclear Fuel*. The Notice of Intent was published on October 24, 1994 (59 FR 53442). The scoping process for this EIS has been completed and an Implementation Plan is being prepared. The Draft EIS is scheduled to be issued for public review in late 1995.

- *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* [Final EIS issued April 1995 (DOE/EIS-0203-F); Record of Decision (60 FR 28680-96, June 1, 1995)]. This EIS analyzes the potential environmental consequences of managing DOE's inventory of spent nuclear fuel over the next 40 years. The Nevada Test Site was considered but was not selected as a DOE spent nuclear fuel management site.

- *Waste Management Programmatic Environmental Impact Statement* (formerly Environmental Management Programmatic EIS). A revised Notice of Intent was published January 24, 1995 (60 FR 4607). This Programmatic EIS will address impacts of potential DOE waste management actions for the treatment, storage, and disposal of waste. The Draft EIS is scheduled to be issued for public review in September 1995.

- *Environmental Impact Statement for a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* [Notice of Intent published October 21, 1993 (58 FR 54336)]. The draft EIS was issued for public review in March 1995 (DOE/EIS-0218D). This EIS addresses the potential environmental impacts of the proposed policy's implementation. Under the proposed policy, the United States could accept up to 22,700 foreign research reactor spent nuclear fuel elements over a 10-15 year period.

- *Environmental Impact Statement on the Transfer and Disposition of Surplus Highly Enriched Uranium* (formerly part of the Programmatic Environmental Impact Statement for Long-Term Storage and Disposition of Weapons-Usable Fissile Materials). The Notice of Intent was issued April 5, 1995 (60 FR 17344). This EIS will address disposition of DOE's surplus highly enriched uranium to support the President's Nonproliferation Policy. The

Draft EIS is scheduled to be issued in September 1995.

- *Programmatic Environmental Impact Statement for Storage and Disposition of Weapons-Usable Fissile Materials* [Notice of Intent published June 21, 1994 (59 FR 31985)]. This Programmatic EIS will evaluate alternatives for long-term storage of all weapons-usable fissile materials (primarily plutonium and highly enriched uranium retained for strategic purposes—not surplus) and disposition of surplus weapons-usable fissile materials (excluding highly enriched uranium), so that risk of proliferation is minimized. The Nevada Test Site is a candidate storage site.

- *Tritium Supply and Recycling Programmatic Environmental Impact Statement*. A revised Notice of Intent was published October 28, 1994 (59 FR 54175), and the Draft Programmatic EIS was issued in March 1995 (60 FR 14433, March 17, 1995). Public hearings on the Draft Programmatic EIS were held in April 1995, and a Final Programmatic EIS is scheduled for October 1995. This EIS addresses how to best assure an adequate tritium supply and recycling capability. The Nevada Test Site is an alternative site for new tritium supply and recycling facilities.

- *Stockpile Stewardship and Management Programmatic Environmental Impact Statement*. A Notice of Intent was published June 14, 1995 (60 FR 31291). A prescoping workshop was held on May 19, 1995, and scoping meetings are scheduled to be held during July and August 1995. This Programmatic EIS will evaluate proposed future missions of the Stockpile Stewardship and Management Program and potential configuration (facility locations) of the nuclear weapons complex to accomplish the Stockpile Stewardship and Management Program missions. The Nevada Test Site is an alternative site for potential location of new or upgraded Stockpile Stewardship and Management Program facilities.

- *Site-Wide Environmental Impact Statement for the Nevada Test Site* [Notice of Intent published August 10, 1994 (59 FR 40897)]. This EIS will address resource management alternatives for the Nevada Test Site to support current and potential future missions involving defense programs, research and development, waste management, environmental restoration, infrastructure maintenance, transportation of wastes, and facility upgrades and alternative uses. The public scoping process has been completed, and the Implementation Plan was issued in July 1995. The Draft

EIS is scheduled to be issued for public review in September 1995.

- *Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapon Components* [Notice of Intent published May 23, 1994 (59 FR 26635); an amended Notice of Intent published June 23, 1995 (60 FR 32661)]. This EIS will address the potential environmental impacts of the continued operation of the Pantex Plant, which includes near- to mid-term foreseeable activities and the nuclear component storage activities at other DOE sites associated with nuclear weapon disassembly operations at the Pantex Plant. The Nevada Test Site is being considered as an alternative site for relocation of interim plutonium pit storage.

Public Reading Rooms

Copies of the Implementation Plan, and the Draft and Final EISs, will be available for inspection during normal business hours at the following public reading rooms. DOE may establish additional information locations and will provide an updated list at the public scoping meetings.

Albuquerque Operations Office, National Atomic Museum, Bldg. 20358, Wyoming Blvd., S.E., Kirtland Air Force Base, Albuquerque, NM 87117. Attn: Diane Leute (505) 845-4378

Atlanta Support Office, U.S. Dept. of Energy, Public Reading Room, 730 Peachtree Street, Suite 876, Atlanta, GA 30308-1212. Attn: Nancy Mays/Laura Nicholas (404) 347-2420

Bartlesville Project Office/National Institute for Petroleum and Energy Research, Library, U.S. Dept. of Energy, 220 Virginia Avenue, Bartlesville, OK 74003. Attn: Josh Stroman (918) 337-4371

Bonneville Power Administration, U.S. Dept. of Energy, BPA-C-KPS-1, 905 N.E. 11th Street, Portland, OR 97208. Attn: Sue Ludeman (503) 230-7334

Chicago Operations Office, Document Dept., University of Illinois at Chicago, 801 South Morgan Street, Chicago, IL 60607. Attn: Seth Nasatir (312) 996-2738

Dallas Support Office, U.S. Dept. of Energy, Public Reading Room, 1420 Mockingbird Lane, Suite 400, Dallas, TX 75247. Attn: Gailene Reinhold (214) 767-7040

Fernald Area Office, U.S. Dept. of Energy, Public Information Room, FERMCO, 7400 Willey Road, Cincinnati, OH 45239. Attn: Gary Stegner (513) 648-3153

Headquarters Office, U.S. Dept. of Energy, Room 1E-190, Forrestal Bldg.,

- '000 Independence Avenue, S.W., Washington, D.C. 20585. Attn: Gayla Messoms (202) 586-5955
- Idaho Operations Office, Idaho Public Reading Room, 1776 Science Center Dr., Idaho Falls, ID 83402. Attn: Brent Jacobson (208) 526-1144
- Kansas City Support Office, U.S. Dept. of Energy, Public Reading Room, 911 Walnut Street, 14th Floor, Kansas City, MO 64106. Attn: Anne Scheer (816) 426-4777
- Office of Civilian Radioactive Waste Management National Information Center, 600 Maryland Avenue, S.W., Suite 760, Washington, D.C. 20024. Attn: Paul D'Anjou (202) 488-6720
- Oak Ridge Operations Office, U.S. Dept. of Energy, Public Reading Room, 55 South Jefferson Circle, Room 112, Oak Ridge, TN 37831-8510. Attn: Amy Rothrock (615) 576-1216
- Oakland Operations Office, U.S. Dept. of Energy, Public Reading Room, EIC, 8th Floor, 1301 Clay Street, Room 700N, Oakland, CA 94612-5208. Attn: Laura Noble (510) 637-1762
- Pittsburgh Energy Technology Center, U.S. Dept. of Energy, Bldg. 922/M210, Receiving Department, Building 166, Cochrans Mill Road, Pittsburgh, PA 15236-0940. Attn: Ann C. Dunlap (412) 892-6167
- Richland Operations Office, U.S. Dept. of Energy, Public Reading Room, 100 Sprout Rd., Room 130 West, Mailstop H2-53, Richland, WA 99352. Attn: Terri Traub (509) 376-8583
- Rocky Flats Field Office, Front Range Community College Library, 3645 West 112th Avenue, Westminster, CO 80030. Attn: Nancy Ben (303) 469-4435
- Savannah River Operations Office, Gregg-Graniteville Library, University of S. Carolina-Aiken, 171 University Parkway, Aiken, SC 29801. Attn: James M. Gaver (803) 725-2889
- Southeastern Power Administration, U.S. Dept. of Energy, Legal Library, Samuel Elbert Bldg., 2 South Public Square, Elberton, GA 30635-2496.
- Attn: Joel W. Seymour/Carol M. Franklin (706) 213-3800
- Southwestern Power Administration, U.S. Dept. of Energy, Public Reading Room, 1 West 3rd, Suite 1600, Tulsa, OK 74103. Attn: Marti Ayers (918) 581-7426
- Strategic Petroleum Reserve Project Management Office, U.S. Dept. of Energy, SPRPMO/SEB Reading Room, 900 Commerce Road East, New Orleans, LA 70123. Attn: Ulysess Washington (504) 734-4243
- Yucca Mountain Science Centers
Yucca Mountain Science Center, U.S. 95—Star Route 374, Beatty, NV 89003. Attn: Marina Anderson (702) 553-2130
- Yucca Mountain Science Center, 4101-B Meadows Lane, Las Vegas, NV 89107. Attn: Melinda D'ouville (702) 295-1312
- Yucca Mountain Science Center, 1141 South Hwy. 160, Pahrump, NV 89041. Attn: Lee Krumm (702) 727-0896

TABLE 1.—SCOPING MEETINGS

Location of scoping meeting	Dates/times ¹
Pahrump Community Center, 400 N. Hwy. 160, Pahrump, NV 89048	Tuesday, August 29, 1995, morning/evening sessions.
Centre on the Grove, 850 W. Front St., Boise, ID 83702	Wednesday, September 6, 1995, morning/evening sessions.
Events Center, University of Nevada-Reno Campus, Reno, NV 89567.	Friday, September 8, 1995, morning/evening sessions.
University of Chicago, Downtown MBA Center, 450 N. Cityfront Plaza Drive, Chicago, IL 60611.	Tuesday, September 12, 1995, morning/evening sessions.
Cashman Field, 850 Las Vegas Blvd. North, Las Vegas, NV 89101	Friday, September 15, 1995, morning/evening sessions .
Denver Convention Complex, 700 14th Street, Denver, CO 80202	Tuesday, September 19, 1995, afternoon/evening sessions.
Sacramento Public Library, 828 I Street, Sacramento, CA 95814	Thursday, September 21, 1995, afternoon/evening sessions.
Arlington Community Center, 2800 South Center Street, Dallas, TX 76004.	Tuesday, September 26, 1995, afternoon/evening sessions.
Caliente Youth Center, Highway 93, Caliente, NV 89008	Thursday, September 28, 1995, morning/evening sessions.
Hilton Inn, 150 West 500 South, Salt Lake City, UT 84111	Thursday, October 5, 1995, afternoon/evening sessions.
Maritime Institute of Technology and Graduate Studies, 5700 Hammonds Ferry Rd., Linthicum (near Baltimore), MD 21090.	Wednesday, October 11, 1995, morning/evening sessions.
Russell Sage Conference Center, 45 Ferry St., Troy (Albany), NY 12180.	Friday, October 13, 1995, afternoon/evening sessions.
Georgia International Convention Center, 1902 Sullivan Road, College Park (Atlanta), GA 30337.	Tuesday, October 17, 1995, morning/evening sessions.
Penn Valley Community College, 3201 S.W. Trafficway, Kansas City, MO 64111.	Friday, October 20, 1995, afternoon/evening sessions.
Tonopah Convention Center, 301 Brougner, Tonopah, NV 89049	Tuesday, October 24, 1995, morning/evening sessions.

¹ Session times are as follows: Morning (8:30 a.m.—12:30 p.m.), Afternoon (12:00 a.m.—4:00 p.m.), Evening (6:00 p.m.—10:00 p.m.).

DEPARTMENT OF ENERGY

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

AGENCY: Department of Energy.

ACTION: Notice of floodplain and
wetlands involvement.

SUMMARY: The U.S. Department of Energy (DOE) is proposing to construct, operate and monitor, and eventually close a geologic repository for the disposal of spent nuclear fuel and high-level radioactive waste at Yucca Mountain, Nye County, Nevada. As part of its proposal, DOE is considering shipping spent nuclear fuel and high-level radioactive waste in the State of Nevada over a rail line that would be constructed or over an existing highway route that may need upgrading to accommodate heavy-haul trucks. Portions of the rail corridor or highway route would cross perennial and ephemeral streams and their associated floodplains, as well as possible wetlands. Furthermore, portions of the transportation system in the immediate vicinity of the proposed repository would be located within the 100-year floodplains of Midway Valley Wash, Drillhole Wash, Busted Butte Wash and/or Fortymile Wash. No other aspect of repository-related operations or nuclear or nonnuclear repository facilities would be located within the 500-year or 100-year floodplains of these washes. In accordance with DOE regulations for Compliance with Floodplain/Wetlands Environmental Review Requirements (10 CFR Part 1022), DOE will prepare a floodplain and wetlands assessment commensurate with proposed decisions and available information. The assessment will be included in the Environmental Impact Statement (EIS) for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada. A draft of this EIS is scheduled to be published during the summer of 1999.

DATES: The public is invited to comment on this notice on or before July 1, 1999. Comments received after this date will be considered to the extent practicable.

ADDRESSES: Comments on this notice should be addressed to Ms. Wendy Dixon, EIS Project Manager, Yucca Mountain Site Characterization Office.

U.S. Department of Energy, P.O. Box 30307, M/S 010, Las Vegas, Nevada 89036-0307. Comments also can be submitted via electronic mail to: eirs@notes.ymp.gov.

FOR FURTHER INFORMATION CONTACT :

Proposed Action: Ms. Wendy Dixon, EIS Project Manager, at the above address, or by calling (800)-881-7292.

Floodplain and Wetlands

Environmental Review Requirements: Ms. Carol Borgstrom, Office of NEPA Policy and Assistance (EH-42), U.S. Department of Energy, 1000 Independence Avenue, S.W., Washington, D.C. 20585, (202)-586-4600 or leave a message at (800) 472-2756.

SUPPLEMENTARY INFORMATION : In accordance with the Nuclear Waste Policy Act, as amended, DOE is studying Yucca Mountain in Nye County, Nevada, to determine its suitability for the deep geologic disposal of commercial and DOE spent nuclear fuel and high-level radioactive waste. In 1989, DOE published a Notice of Floodplain/Wetlands Involvement (54 FR 6318, February 9, 1989) for site characterization at Yucca Mountain, and in 1992 published a Floodplain Statement of Findings (57 FR 48363, October 23, 1992).

DOE is now preparing an EIS (DOE-EIS-0250) to assess the potential environmental impacts from the construction, operation and monitoring, and eventual closure of the proposed geologic repository. DOE issued a Notice of Intent to prepare the EIS on August 7, 1995 (60 FR 40164). As part of its proposal, DOE is considering shipping spent nuclear fuel and high-level radioactive waste in the State of Nevada over a rail line that would be constructed or over an existing highway route that may need upgrading to accommodate heavy-haul trucks. For the rail mode, DOE is evaluating five potential corridors (Figure 1). For the heavy-haul truck mode, DOE is evaluating three potential locations for an intermodal transfer station associated with five potential highway routes (Figure 2; an intermodal transfer station is a facility at which shipping casks containing spent nuclear fuel and high-level radioactive waste would be transferred from trains to trucks, and empty shipping casks would be transferred from trucks to trains). The rail corridors would be about 400 meters (0.25 mile) wide. The Carlin Corridor would be the longest at 520 kilometers (323 miles) followed by the Caliente (513 kilometers, 319 miles), Caliente-Chalk Mountain (345 kilometers, 214 miles), Jean (181 kilometers, 112 miles),

and Valley Modified (159 kilometers, 98 miles) corridors. The heavy-haul routes would utilize existing roads and rights-of-ways which typically would be less than 400 meters (0.25 miles) in width. The Caliente Route would be the longest at 533 kilometers (331 miles) followed by the Caliente-Las Vegas (377 kilometers, 234 miles), Caliente-Chalk Mountain (282 kilometers, 175 miles), Sloan/Jean (190 kilometers, 118 miles) and Apex/Dry Lake (183 kilometers, 114 miles) routes.

Portions of the transportation system in the immediate vicinity of the proposed repository are likely to be located within the 100-year floodplains of Midway Valley Wash, Drillhole Wash, Busted Butte Wash and/or Fortymile Wash (Figure 3). Fortymile Wash, a major wash that flows to the Amargosa River, drains the eastern side of Yucca Mountain. Midway Valley Wash, Drillhole Wash and Busted Butte Wash are tributaries to Fortymile Wash. Although water flow in Fortymile Wash and its tributaries is rare, the area is subject to flash flooding from thunderstorms and occasional sustained precipitation. There are no naturally occurring wetlands near the proposed repository facilities, although there are two man-made well ponds in Fortymile Wash that support riparian vegetation.

If the Proposed Action were implemented, DOE would use an existing road during construction of the repository that crosses the 100-year floodplain of Fortymile Wash (Figure 3). This road and other features of site characterization that involve floodplains have previously been examined by DOE and a Statement of Findings was issued in 1992 (57 FR 48363, October 23, 1992). It is uncertain at this time whether this existing road would require upgrading to accommodate the volume and type of construction vehicles.

In addition, transportation infrastructure would be constructed either in Midway Valley Wash, Drillhole Wash and Busted Butte Wash, or in Midway Valley Wash, Drillhole Wash and Fortymile Wash. The decision on which washes would be involved is dependent on future decisions regarding the mode of transport (rail or truck) which, in turn, would require the selection of one rail corridor or the selection of one site for an intermodal transfer station and its associated heavy-haul route. Structures that might be constructed in a floodplain could include one or more bridges to span the washes, one or more roads that could pass through the washes, or a combination of roads and culverts in the washes. No other aspect of repository-

related operation of nuclear or nonnuclear facilities would be located within 500-year or 100-year floodplains.

Outside of the immediate vicinity of the proposed repository, the five rail corridors, and the three sites for an intermodal transfer station and associated five heavy-haul routes, would cross perennial and ephemeral streams, and possibly wetlands. It is likely that a combination of bridges, roads and culverts, or other engineered features, would be needed to span or otherwise cross the washes and possible wetlands, although the location of such structures is uncertain at this time.

DOE will prepare an initial floodplain and wetlands assessment commensurate with the proposed decisions and available information. This assessment will be included in the Draft EIS that is scheduled to be issued for public comment later this summer. If, after a possible recommendation by the Secretary of Energy, the President considers the site qualified for an application to the U.S. Nuclear Regulatory Commission for a construction authorization, the President will submit a recommendation of the site to Congress. If the site designation becomes effective, the Secretary of Energy will submit to the Nuclear Regulatory Commission a License Application for a construction authorization. DOE would then probably select a rail corridor or a site for an intermodal transfer station among those considered in the EIS. Following such a decision, additional field surveys, environmental and engineering analyses, and National Environmental Policy Act reviews would likely be needed regarding a specific rail alignment for the selected corridor or the site for the intermodal transfer station and its associated heavy-haul truck route. When more specific information becomes available about activities proposed to take place within floodplains and wetlands, DOE will conduct further environmental review in accordance with 10 CFR Part 1022. Information that would be considered in a subsequent assessment includes, for example, the identification of 500-year and 100-year floodplains among feasible alignments of the selected rail corridor or the site of the intermodal transfer station and its associated heavy-haul route, identification of individual wetlands, and whether the floodplains and wetlands could be avoided. If the floodplains and wetlands could not be avoided, information on specific engineering designs and associated construction activities in the floodplains and wetlands also would be needed to permit a more detailed assessment and

to ensure that DOE minimizes potential harm to or within any affected floodplains or wetlands.

Issued in Las Vegas, Nevada, on the 4th day of June 1999.

Wendy Dixon,
EIS Project Manager.

BILLING CODE 6450-01-P

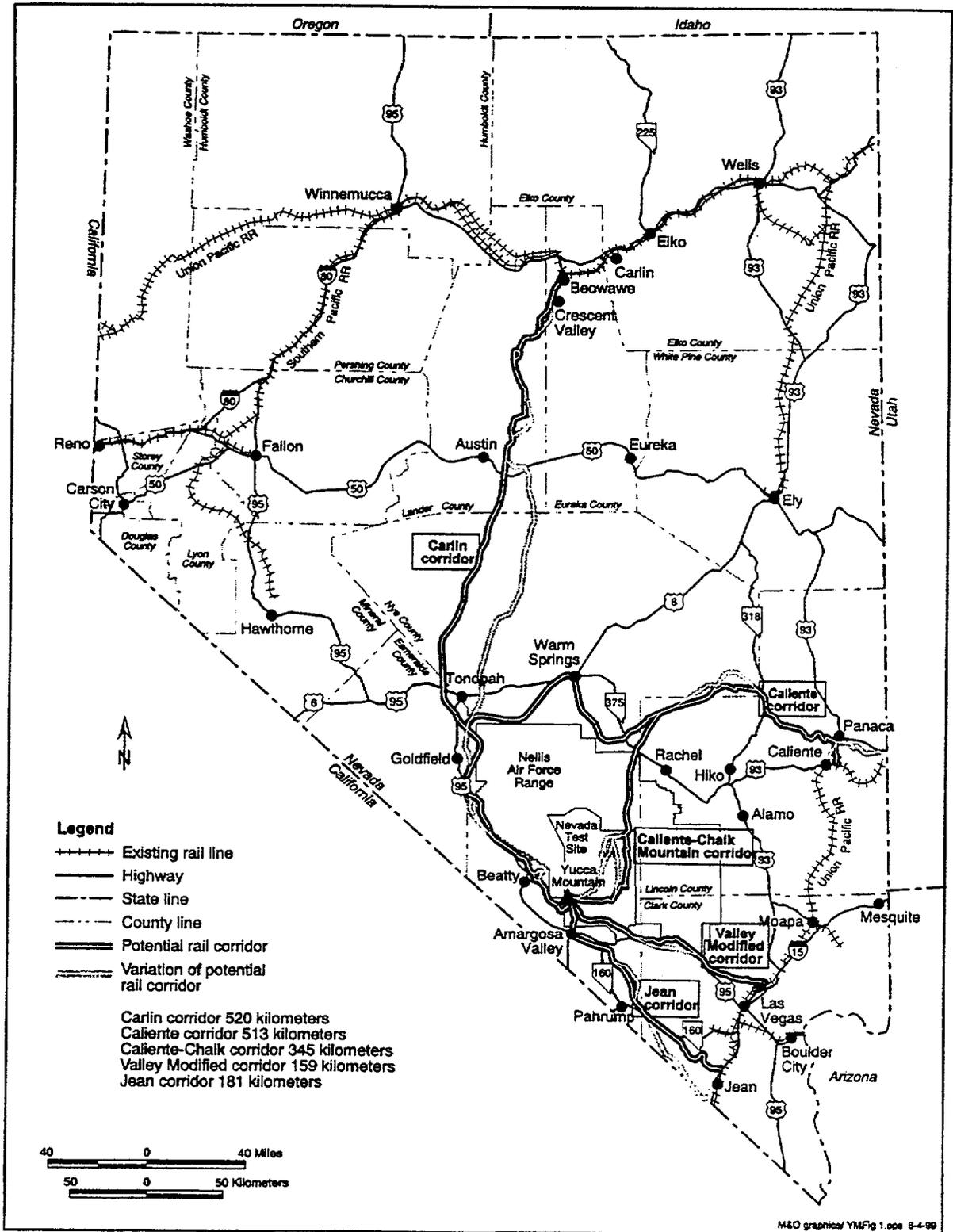


Figure 1. Potential Nevada rail corridors to Yucca Mountain.

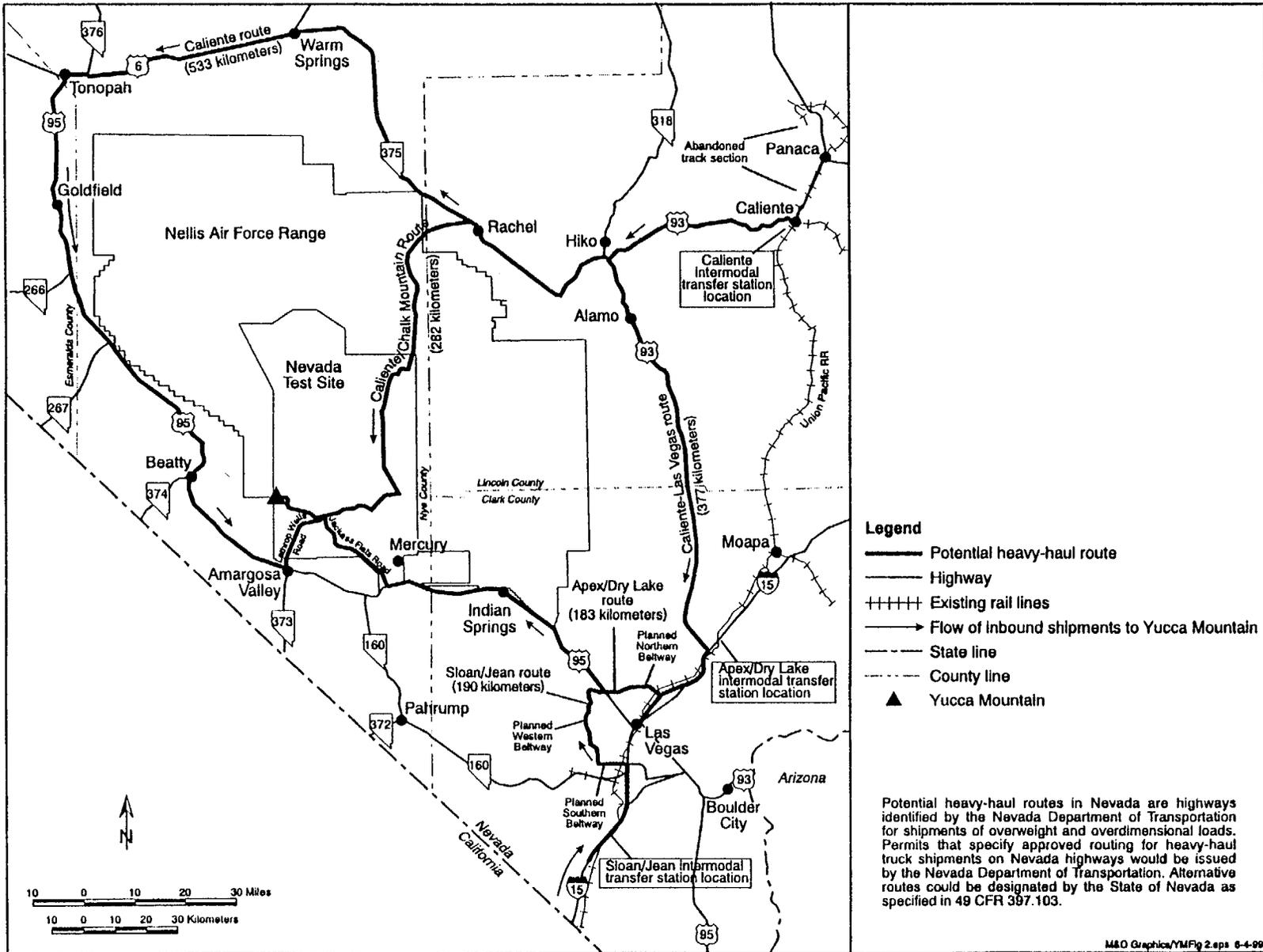


Figure 2. Potential routes in Nevada for heavy-haul trucks.

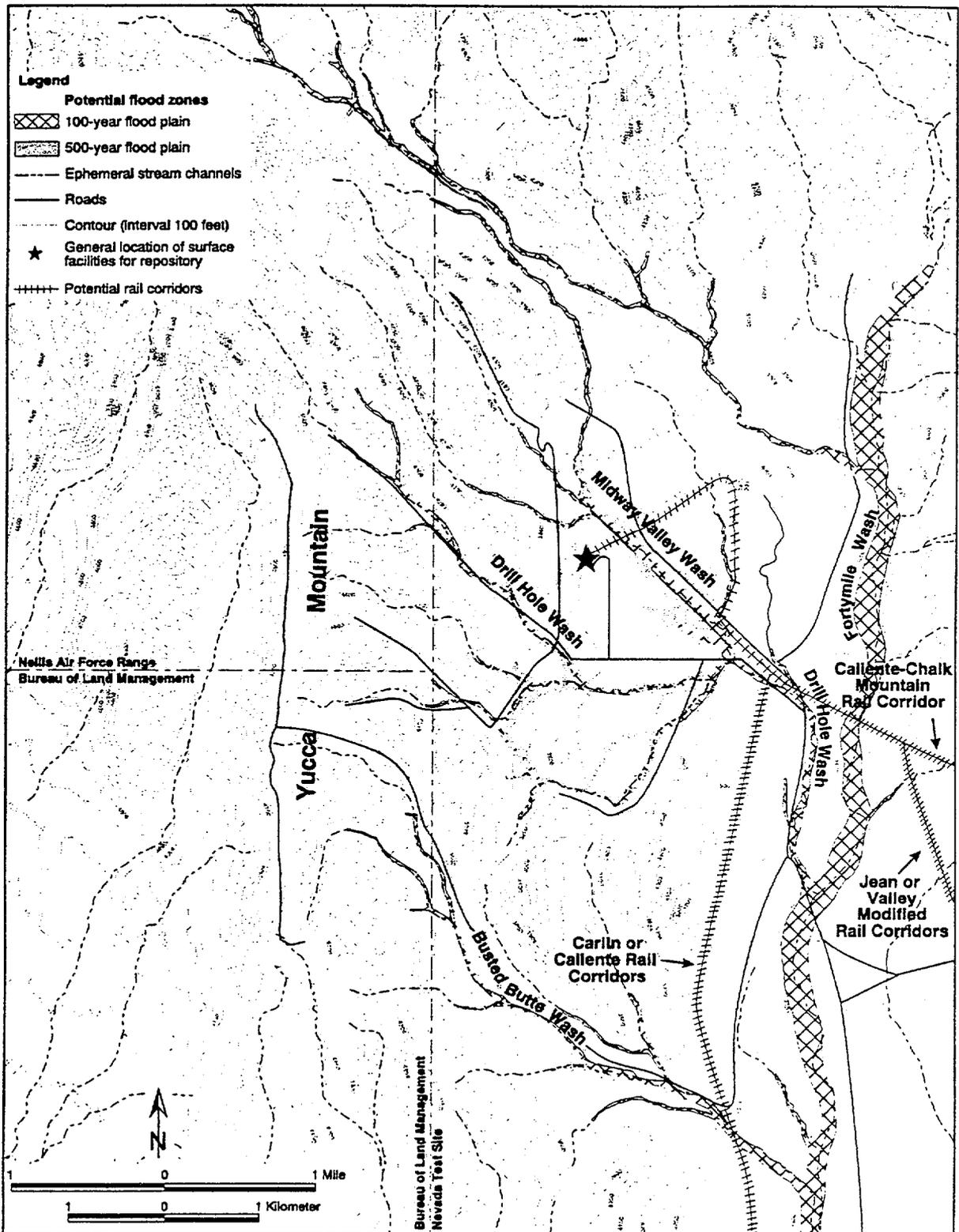


Figure 3. Yucca Mountain site topography, plains, and potential rail corridors.



Appendix C

Interagency and
Intergovernmental
Interactions

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APPENDIX C. INTERAGENCY AND INTERGOVERNMENTAL INTERACTIONS

In the course of producing this environmental impact statement (EIS), the U. S. Department of Energy (DOE) has interacted with a number of governmental agencies and other organizations. These interaction efforts have several purposes, as follows:

- Discuss issues of concern with organizations having an interest in or authority over land that the Proposed Action (to construct, operate and monitor, and eventually close a geologic repository at Yucca Mountain) would affect directly, or organizations having other interests that some aspect of the Proposed Action could affect.
- Obtain information pertinent to the environmental impact analysis of the Proposed Action.
- Initiate consultations or permit processes, including providing data to agencies with oversight, review, or approval authority over some aspect of the Proposed Action.

Section C.1 summarizes the interactions. DOE has completed several efforts and will complete all required consultations before publishing the Final EIS. Section C.2 describes interests held by agencies and organizations involved in consultations and other interactions.

C.1 Summary of Activity

Table C-1 lists organizations with which DOE has initiated interaction processes concerning the proposed Yucca Mountain Repository and the status of those interactions.

C.2 Interests of Selected Agencies and Organizations in the Yucca Mountain Repository Proposal

Regulations that establish a framework for interactions include 40 CFR 1502.25, which provides for consultations with agencies having authority to issue applicable licenses, permits, or approvals, or to protect significant resources, and 10 CFR 1021.341(b), which provides for interagency consultations as necessary or appropriate.

C.2.1 BUREAU OF LAND MANAGEMENT

The Bureau of Land Management has a range of interests potentially affected by the Proposed Action. The Bureau, as a part of the U.S. Department of the Interior:

- Controls a portion of the land that would need to be withdrawn by Congress to accommodate the proposed repository
- Controls portions of land in Nevada in the five corridors for a potential branch rail line and along the five potential routes for heavy-haul trucks
- Has responsibility for wild horse and wild burro management areas (Public Law 92-195, as amended, Section 3; 43 CFR Part 2800) and wildlife management areas (43 CFR 24.4) in Nevada that alternative rail corridors and routes for heavy-haul trucks cross
- Has power to grant rights-of-way and easements for transportation routes across lands it controls

Table C-1. Organizations with which DOE has initiated interactions (page 1 of 2).

Organization	Authority/interest	Interactions
Bureau of Land Management	Controls part of land required for repository. Controls portions of lands in Nevada that transportation corridors cross. Has responsibility for management and use of lands it controls, including management of habitat and species. Has data on topography, habitat, species, and other topics on land it controls.	DOE provided a briefing on the EIS during a meeting on September 15, 1998.
U.S. Air Force	Controls part of land being considered for withdrawal for repository (on the Nellis Air Force Range) and for one Nevada rail implementing alternative and one heavy-haul truck implementing alternative. Has identified security concerns over potential development of the Nevada rail and heavy-haul truck implementing alternatives that would pass through land it controls.	DOE has provided a briefing for USAF personnel on the process DOE is following for this EIS and on the range of issues being analyzed. DOE and Air Force personnel have held informal meetings to discuss specific issues and update EIS status. The Air Force has provided a statement of its concerns regarding certain transportation alternatives DOE is considering.
Naval Nuclear Propulsion Program	The Naval Nuclear Propulsion Program is a joint U.S. Navy and DOE organization responsible for management of naval spent nuclear fuel.	Ongoing dialogue and information exchange.
Fish and Wildlife Service	Oversees compliance with the Endangered Species Act for some species and compliance with the Fish and Wildlife Coordination Act.	Discussions have been held and species list information has been obtained. Interaction activities under the Endangered Species Act are ongoing.
National Marine Fisheries Service	Oversees compliance with Marine Protection Research and Sanctuaries Act and, for some species, with the Endangered Species Act.	Discussions have been held and information has been obtained. Interaction activities under the Endangered Species Act are ongoing.
U.S. Department of Transportation	Has regulatory authority over transportation of nuclear and hazardous waste materials, including packaging design, manufacture and use, pickup, carriage, and receipt, and highway route selection.	EIS status briefing has been provided. DOE and DOT have held informal discussions concerning modeling techniques and analytical methods DOE is using in its evaluation of transportation issues.
U.S. Environmental Protection Agency	Has regulatory authority over radiological standards and groundwater protection standards. Mandatory role in review of EIS adequacy.	DOE and EPA have held a meeting at which DOE provided a briefing on its approach to the EIS and on scope and content. At this meeting, EPA described its EIS rating process and personnel from the two agencies discussed methods for addressing any EIS comments that EPA may submit.
U.S. Nuclear Regulatory Commission	Required by NWRPA to adopt Yucca Mountain Repository EIS to the extent practicable with the issuance by the Commission of any construction authorization and license for a repository. Has licensing authority over spent nuclear fuel and high-level radioactive waste geologic repositories. Has licensing authority over spent nuclear fuel and high-level radioactive waste geologic repositories. Has regulatory authority over commercial nuclear power plants, storage of spent nuclear fuel at commercial sites, and packaging for transportation of spent nuclear fuel and high-level radioactive waste. Has general authority over possession and transfer of radioactive material.	Discussions have been held on the purpose and need for the action and on the status of the EIS. Numerous interactions related to the potential repository program in general.

Table C-1. Organizations with which DOE has initiated interactions (page 2 of 2).

Organization	Authority/interest	Interactions
U.S. Army Corps of Engineers	Has authority over activities that discharge dredge or fill material into waters of the United States.	Discussed strategies for minimizing impacts and obtaining permits for waters of the United States.
U.S. Department of Agriculture	Responsible for protection of prime farm lands for agriculture in areas potentially affected by the Proposed Action.	Letter exchange has resolved issues regarding repository's potential effect on farmlands. Need for additional interaction is uncertain.
Native American Tribes	Have concern for potential consequences of repository development and transportation activities on cultural resources, traditions, and spiritual integrity of the land. Have governmental status. All interactions required for the American Indian Religious Freedom Act, the Native American Graves Protection and Repatriation Act, and the National Historic Preservation Act are being accomplished.	Ongoing discussions on a range of topics at least twice per year. Tribal representatives have prepared and submitted the <i>American Indian Perspectives on the Yucca Mountain Site Characterization Project and the Repository Environmental Impact Statement</i> (AIWS 1998, all).
Affected units of local government	Local governments with general jurisdiction over regions or communities that could be affected by implementation of the Proposed Action.	Meetings that include discussions, information exchange, and status briefings.
National Park Service	Potential for proposal to affect water supply in Death Valley region. Effect of any water appropriation required for repository, EIS status, and approach to EIS development.	Discussion completed. National Park Service concerns in regard to use of water for repository construction and operation were addressed.
Advisory Council on Historic Preservation and Nevada State Historic Preservation Officer	Protection and preservation of historic properties and cultural resources of importance to Native Americans and others. Administration of the National Historic Preservation Act and of regulatory requirements supporting that act.	Following discussions among DOE, the Advisory Council on Historic Preservation, and the Nevada State Historic Preservation Officer, DOE and the Advisory Council on Historic Preservation have entered into a programmatic agreement (DOE 1988, all) establishing procedures DOE is to follow during site characterization and during the Secretary of Energy's development of a repository site recommendation. The Advisory Council on Historic Preservation indicated that it would be available to assist DOE in complying with environmental review requirements for historic properties.
State of Nevada Department of Transportation	Has authority over transportation and highways in Nevada.	DOE and Nevada Department of Transportation personnel have had informal discussions on Nevada transportation issues. The State of Nevada has requested a formal briefing on this draft EIS after DOE publishes the document. DOE has agreed to provide a briefing to the state.

The Bureau of Land Management would have a continuing interest in the development of a repository at Yucca Mountain and associated transportation routes in the State of Nevada. Any comments from the Secretary of the Interior on the EIS must be included in the Secretary of Energy's recommendations to the President on the Yucca Mountain site.

Interaction

DOE held a meeting with the Bureau of Land Management on September 15, 1998.

C.2.2 U.S. AIR FORCE

The U.S. Air Force operates Nellis Air Force Base northeast of Las Vegas, and the Nellis Air Force Range, which occupies much of south-central Nevada. The Nellis Range is an important facility for training American and Allied combat pilots and crews (USAF 1999, pages 1-1 and 1-3).

A portion of the land being considered for withdrawal for the proposed repository is on the Nellis Range. If the land were withdrawn and development of the proposed repository proceeded, the Air Force would hold a continuing interest in the potential for construction, operation and monitoring, and closure activities at the repository to have consequences for Air Force operations on the adjoining land.

One Nevada rail implementing alternative and one Nevada heavy-haul truck implementing alternative that DOE is evaluating for the transportation of spent nuclear fuel and high-level radioactive waste would pass through a portion of the Nellis Range, for which the Air Force has national security concerns.

Interaction

DOE provided a briefing for USAF personnel on the process DOE is following for this EIS and on the range of issues being analyzed. DOE and Air Force personnel have held informal meetings to discuss specific issues. The Air Force has provided a statement of concerns about certain transportation alternatives DOE considered in the EIS.

C.2.3 NAVAL NUCLEAR PROPULSION PROGRAM

The Naval Nuclear Propulsion Program is a joint U.S. Navy and DOE program responsible for all matters pertaining to naval nuclear propulsion (USN 1996, page 2-2). This program is responsible for the nuclear propulsion plants aboard more than 93 nuclear-powered warships with more than 108 reactors and for nuclear propulsion work performed at four naval shipyards and two private shipyards. It is also responsible for two government-owned, contractor-operated laboratories, two moored training ships, two land-based prototype reactors, and the Expended Core Facility at the Naval Reactors Facility at the Idaho National Engineering and Environmental Laboratory.

The Naval Nuclear Propulsion Program manages naval spent fuel after its withdrawal from nuclear-powered warships and prototype reactors at the Expended Core Facility. The program has conducted studies and performed environmental impact analyses on the management and containerization of naval spent nuclear fuel to prepare it for shipment to the proposed repository or other spent fuel management system (USN 1996, all). Information from these studies is relevant to the containerization of other spent nuclear fuel that could be shipped to the proposed repository.

Interaction

Since the beginning of preparations for this EIS, the Naval Nuclear Propulsion Program has participated in quarterly meetings with DOE to discuss information relevant to the emplacement of naval spent nuclear fuel in a monitored geologic repository. Detailed information about naval spent nuclear fuel is classified; therefore, the Naval Nuclear Propulsion Program performed a parallel set of thermal, nuclear, and dose calculations and provided unclassified results to DOE for inclusion in this EIS. In some cases DOE used those results as input parameters for additional analyses. Representatives of the program participated throughout the review process to ensure the accurate presentation of information on naval spent nuclear fuel.

C.2.4 FISH AND WILDLIFE SERVICE

The Fish and Wildlife Service, a bureau of the U.S. Department of the Interior, has a role in the overall evaluation of the impacts from the Proposed Action under consideration in the repository EIS. Under the Endangered Species Act of 1973, as amended, the Fish and Wildlife Service has responsibility to determine if projects such as the proposed Yucca Mountain Repository would have an adverse impact on endangered or threatened species or on species proposed for listing. Any comments from the Secretary of the Interior on the EIS must accompany the Secretary of Energy's recommendation to the President on the Yucca Mountain site.

No endangered or proposed species occur on lands that would be needed for the repository. The desert tortoise is the only threatened species known to exist on this land, which lies at the northern edge of the range for desert tortoises (Buchanan 1997, pages 1 to 4). The repository would not need or impact any critical habitat.

To evaluate the potential for the proposed repository to affect the desert tortoise, DOE and the Fish and Wildlife Service are following a process that, in summary, includes three steps:

1. DOE submits a study (biological assessment) containing information on desert tortoise activities and habitat in the vicinity of the proposed project, a description of project activities that could affect the desert tortoise, and the potential for adverse impacts to desert tortoises or habitat. Based on this information, DOE will determine if the project would result in adverse impacts to the species.
2. DOE and the Fish and Wildlife Service will meet as necessary to discuss details of the potential for interaction between desert tortoises and project activities, and to consider appropriate protective measures DOE could take to reduce the potential for project impact to desert tortoises.
3. The Fish and Wildlife Service will issue a biological opinion that states its opinion on whether the proposed project may proceed without causing adverse impacts to the desert tortoise, jeopardizing the continued existence of the species, or resulting in harassment, harm, or death of individual animals. The biological opinion may contain protective measures and conditions that DOE would have to implement during construction, operation and monitoring, and closure of the proposed repository to minimize adverse impacts and the potential for tortoise deaths.

DOE, which has conducted site characterizations at Yucca Mountain since 1986, and the Fish and Wildlife Service have conducted previous consultation processes that addressed the potential for site characterization activities to affect the desert tortoise. These processes resulted in biological opinions, published in 1990 and 1997, that determined that site characterization activities could proceed without unacceptable harm to the desert tortoise and that the protective measures and conditions stated in the biological opinions should apply to DOE activities. None of the proposed repository land is critical habitat for tortoises. The current consultation process on the desert tortoise will build on the information gathered and the practices developed in the previous consultations, and on the positive results obtained.

Interaction

DOE is currently preparing a Biological Assessment to be submitted to the Fish and Wildlife Service.

C.2.5 NATIONAL MARINE FISHERIES SERVICE

The National Marine Fisheries Service exercises protective jurisdiction over aspects of the marine environment, including research activities, marine sanctuaries, and certain species protected by the Endangered Species Act. Potential DOE actions associated with transportation to the repository (for

example, barging and construction or modification of bridges and docking facilities) could require interaction with the National Marine Fisheries Service.

Interaction

DOE participated in an informal discussion that identified National Marine Fisheries Service jurisdiction relevant to the Yucca Mountain Project and potential project activities of jurisdictional interest to the National Marine Fisheries Service in fulfilling its responsibilities.

C.2.6 U.S. DEPARTMENT OF TRANSPORTATION

The U.S. Department of Transportation has the authority to regulate several aspects of the transportation of spent nuclear fuel and high-level radioactive waste to the proposed Yucca Mountain Repository. The general authority of the Department of Transportation to regulate carriers and shippers of hazardous materials includes packaging procedures and practices, shipping of hazardous materials, routing, carrier operations, equipment, shipping container construction, and receipt of hazardous materials (49 USC 1801; 49 CFR Parts 171 through 180).

Interaction

DOE and the Department of Transportation have exchanged letters and informal communications on topics pertaining to the proposed Yucca Mountain Project that are within the Department of Transportation's regulatory interest. DOE and the Department of Transportation have held informal discussions on the modeling techniques and analytical methods DOE used in its evaluation of transportation issues.

C.2.7 U.S. ENVIRONMENTAL PROTECTION AGENCY

The U.S. Environmental Protection Agency has two primary responsibilities in relation to the proposed Yucca Mountain Repository. It is responsible for promulgating regulations that set radiological protection standards for media that would be affected if radionuclides were to escape the confinement of the repository. In addition, the Agency oversees the National Environmental Policy Act process for Federal EISs. Council on Environmental Quality regulations implementing the National Environmental Policy Act specify procedures that agencies must follow and actions that agencies must take in preparing EISs. Depending on the level of concern that the Agency might have with environmental aspects of the Yucca Mountain Project Draft EIS, it can initiate a consultation between DOE and the Council on Environmental Quality. The Secretary of Energy's recommendation to the President must include both the Final EIS and the Environmental Protection Agency's comments on the EIS.

Interaction

DOE and the Environmental Protection Agency held a meeting at which DOE provided a briefing on its approach to the EIS and its scope and content. At that meeting, the Environmental Protection Agency described its EIS rating process, and personnel from the two agencies discussed methods for addressing EIS comments that the Agency might submit.

C.2.8 U.S. NUCLEAR REGULATORY COMMISSION

The Nuclear Waste Policy Act (42 USC 10101 *et seq.*) establishes a multistep procedure for reviews and decisions on the proposal to construct, operate and monitor, and close a geologic repository at Yucca Mountain. The final steps in this procedure require DOE to make an application to the U.S. Nuclear Regulatory Commission for authorization to construct a repository at Yucca Mountain and the Commission to consider this information and make a final decision within 3 years on whether to approve the application. The Nuclear Waste Policy Act directs the Commission to adopt this EIS to the

extent practicable in support of its decisionmaking process. Any Nuclear Regulatory Commission comment on this EIS must accompany the Secretary of Energy's recommendation to the President.

The Nuclear Regulatory Commission also has authority under the Atomic Energy Act of 1954, as amended, to regulate persons authorized to own, possess, or transfer radiological materials. In addition, the Commission regulates transportation packaging, transportation operations, and the design, manufacture, and use of shipping containers for radiological materials with levels of radioactivity greater than Department of Transportation Type A materials. Determination as to whether radiological materials are Type A or greater are made in accordance with a procedure set forth in 49 CFR 173.431.

Interaction

Discussions have been held on the purpose and need for the Proposed Action and on the status of the EIS. Interactions with the Nuclear Regulatory Commission will include those necessary to process any application to construct a repository at Yucca Mountain.

C.2.9 U.S. ARMY CORPS OF ENGINEERS

The Clean Water Act of 1977 (42 USC 1251 *et seq.*) gives the U.S. Army Corps of Engineers permitting authority over activities that discharge dredge or fill material into waters of the United States. If DOE activities associated with a repository at Yucca Mountain discharged dredge or fill into any such waters, DOE could need to obtain a permit from the Corps. The construction or modification of rail lines or highways to the repository would also require Section 404 permits if those actions included dredge and fill activities or other activities that would discharge dredge or fill into waters of the United States. DOE has obtained a Section 404 permit for site characterization-related construction activities it might conduct in Coyote Wash or its tributaries or in Fortymile Wash.

Interaction

Strategies for minimizing any impacts and obtaining permits have been discussed.

C.2.10 U.S. DEPARTMENT OF AGRICULTURE

The U.S. Department of Agriculture has the responsibility to ensure that the potential for Federal programs to contribute to unnecessary and irreversible conversion of farmlands to nonagricultural uses is kept to a minimum. Proposed Federal projects must obtain concurrence from the Natural Resource Conservation Service of the Department of Agriculture that potential activities would not have unacceptable effects on farmlands (7 USC 4201 *et seq.*).

Interaction

DOE has had written communication with the Department of Agriculture. The process has resulted in a concurrence that a repository at Yucca Mountain would not affect farmlands.

C.2.11 NATIVE AMERICAN TRIBES

Many tribes have historically used the area being considered for the proposed Yucca Mountain Repository, as well as nearby lands (AIWS 1998, page 2-1). The region around the site holds a range of cultural resources and animal and plant resources. Native American tribes have concerns about the protection of cultural resources and traditions and the spiritual integrity of the land. Tribal concerns extend to the propriety of the Proposed Action, the scope of the EIS, and opportunities to participate in the EIS process, as well as issues of environmental justice and the potential for transportation impacts (AIWS 1998, pages 2-2 to 2-26, and 4-1 to 4-12). Potential rail and legal-weight truck routes would follow existing rail lines and highways, respectively. The legal-weight truck route would pass through

the Moapa Indian Reservation and the potential rail line would pass near the Reservation. Potential routes for legal-weight and heavy-haul trucks would follow existing highways, and would pass through the Las Vegas Paiute Indian Reservation.

DOE Order 1230.2 recognizes that Native American tribal governments have a special and unique legal and political relationship with the Government of the United States, as defined by history, treaties, statutes, court decisions, and the U.S. Constitution. DOE recognizes and commits to a government-to-government relationship with Native American tribal governments. DOE recognizes tribal governments as sovereign entities with, in most cases, primary authority and responsibility for Native American territory. DOE recognizes that a trust relationship derives from the historic relationship between the Federal Government and Native American tribes as expressed in certain treaties and Federal law. DOE has and will consult with tribal governments to ensure that tribal rights and concerns are considered before taking actions, making decisions, or implementing programs that could affect tribes. These interactions ensure compliance with provisions of the American Indian Religious Freedom Act (42 USC 1996 *et seq.*), the Native American Graves Protection and Repatriation Act (25 USC 3001 *et seq.*), DOE Order 1230.2 (*American Indian Tribal Government Policy*), Executive Order 13007 (*Sacred Sites*), Executive Order 13084 (*Consultation and Coordination with Indian Tribal Governments*), and the National Historic Preservation Act (16 USC 470f).

Interaction

The Native American Interaction Program was formally begun in 1987. Representatives from the Consolidated Group of Tribes and Organizations have met in large group meetings twice yearly with DOE on a range of cultural and other technical concerns. Additionally, specialized Native American subgroups have been periodically convened to interact with DOE on specific tasks including ethnobotany, review of artifact collections, field archaeological site monitoring, and the EIS process.

The Consolidated Group of Tribes and Organizations consists of the following:

- ***Southern Paiute***
 - Kaibab Paiute Tribe, Arizona
 - Paiute Indian Tribes of Utah
 - Moapa Band of Paiutes, Nevada
 - Las Vegas Paiute Tribe, Nevada
 - Pahrump Paiute Tribe, Nevada
 - Chemehuevi Paiute Tribe, California
 - Colorado River Indian Tribes, Arizona

- ***Western Shoshone***
 - Duckwater Shoshone Tribe, Nevada
 - Ely Shoshone Tribe, Nevada
 - Yomba Shoshone Tribe, Nevada
 - Timbisha Shoshone Tribe, California

- ***Owens Valley Paiute and Shoshone***
 - Benton Paiute Tribe, California
 - Bishop Paiute Tribe, California
 - Big Pine Paiute Tribe, California
 - Lone Pine Paiute Tribe, California
 - Fort Independence Paiute Tribe, California

- ***Other Official Native American Organizations***
 - Las Vegas Indian Center, Nevada

Tribal representatives have prepared and submitted the *American Indian Perspectives on the Yucca Mountain Site Characterization Project and the Repository Environmental Impact Statement* (AIWS 1998, all). This document discusses site characterization at Yucca Mountain and the Proposed Action in the context of Native American culture, concerns, and views and beliefs concerning the surrounding region. It has been used as a resource in the preparation of the EIS; excerpts are presented in Chapter 4, Section 4.1.13.4, to reflect a Native American point of view. The issues discussed ranged from traditional resources to concerns related to the potential repository.

C.2.12 AFFECTED UNITS OF LOCAL GOVERNMENT

As defined by the NWPA, the affected units of local government are local governments (counties) with jurisdiction over the site of a repository. Concerns of the affected units of local government range from socioeconomic impacts to potential consequences of transportation activities. Nye County, Nevada, in which DOE would build the repository, is one of the affected units of local government. Others include Clark, Lincoln, Esmeralda, Mineral, Churchill, Lander, Eureka, White Pine, and Elko Counties in Nevada and Inyo County in California.

DOE has offered local governments the opportunity to submit documents providing perspectives of issues associated with the EIS. At Draft EIS publication, Nye County had prepared such a document. In addition, other documents related to the Yucca Mountain region have been prepared in the past by several local government units including Clark, Lincoln, and White Pine Counties.

Interaction

DOE has held formal meetings twice a year with the affected units of local government. These meetings have included discussions and status briefings on a range of issues of interest to local governments. DOE has also held numerous informal meetings with representatives. Documents have been received from units of local government.

C.2.13 NATIONAL PARK SERVICE

The National Park Service, which is a bureau of the U.S. Department of the Interior, is responsible for the management and maintenance of the Nation's national parks and monuments. The implementation of the Proposed Action could potentially affect the water supply in Death Valley National Park, which is downgradient from Yucca Mountain. The National Park Service, therefore, would have an interest in any water appropriation granted to DOE for the repository. In addition, the Park Service has expressed its interest in this EIS, its status, and the approach DOE has followed in developing the EIS.

Interaction

DOE and National Park Service representatives held a discussion during which they addressed Park Service concerns about water use for repository construction and operation.

C.2.14 STATE OF NEVADA

If DOE receives authorization to construct, operate and monitor, and eventually close a geologic repository at Yucca Mountain, DOE would need to obtain a range of permits and approvals from the State of Nevada. DOE would need to coordinate application processing activities with the State to complete the permitting processes. DOE could require permits or approvals such as the following:

- An operating permit for control of gaseous, liquid, and particulate emissions associated with construction and operation
- A public water system permit and a water system operating permit for provision of potable water

- A general permit for storm-water discharge
- A National Pollutant Discharge Elimination System permit for point source discharges to waters of the State
- A hazardous materials storage permit to store, dispense, use, or handle hazardous materials
- A permit for a sanitary and sewage collection system
- A solid waste disposal permit
- Other miscellaneous permits and approvals

DOE required similar permits and approvals from the State of Nevada to conduct site characterization activities at Yucca Mountain. DOE and the State coordinated on a range of activities, including an operating permit for surface disturbances and point source emissions, an Underground Injection Control Permit and a Public Water System Permit, a general discharge permit for effluent discharges to the ground surface, a permit for the use of groundwater, a permit from the State Fire Marshal for the storage of flammable materials, and a permit for operation of a septic system. DOE could apply for additional or expanded authority under the existing permits, where needed, if provisions for expansion became applicable. DOE or its contractors could also need to coordinate transportation activities, highway uses, and transportation facility construction and maintenance activities with the Nevada Department of Transportation.

Interaction

The State of Nevada has requested a formal briefing on this Draft EIS after its publication, and DOE has agreed to provide the briefing. DOE and the Nevada Department of Transportation personnel have had information discussions on Nevada transportation issues.

C.2.15 ADVISORY COUNCIL ON HISTORIC PRESERVATION AND NEVADA STATE HISTORIC PRESERVATION OFFICER

In the mid- to late-1980s, DOE, the Nevada State Historic Preservation Officer and the Advisory Council on Historic Preservation discussed the development of a Programmatic Agreement to address DOE responsibilities under Sections 106 and 110 of the National Historic Preservation Act and the Council's implementing regulations. These discussions led to a Programmatic Agreement between DOE and the Advisory Council on Historic Preservation (DOE 1988, all) that records stipulations and terms to resolve potential adverse effects of DOE activities on historic properties at Yucca Mountain. The activities covered by the Agreement include site characterization of the Yucca Mountain site under the NWPA and the DOE recommendation to the President on whether or not to develop a repository, informed by a final EIS prepared pursuant to the National Environmental Policy Act and the NWPA.

Although not a formal signatory, the Nevada State Historic Preservation Officer has the right at any time, on request, to participate in monitoring DOE compliance with the Programmatic Agreement. In addition, DOE must provide opportunities for consultations with the Advisory Council on Historic Preservation, the Nevada State Historic Preservation Officer, and Native American tribes as appropriate throughout the process of implementing the Agreement. DOE submits an annual report to the Advisory Council and the Nevada State Historic Preservation Officer describing the activities it conducts each year to implement the stipulations of the Programmatic Agreement. This report includes a description of DOE coordinations and consultations with Federal and State agencies and Native American Tribes on historic and culturally significant properties at Yucca Mountain.

DOE will continue to seek input from the Nevada State Historic Preservation Officer and the Advisory Council on Historic Preservation, and will interact appropriately to meet the reporting and other stipulations of the Programmatic Agreement.

Interaction

DOE has submitted annual reports to the Nevada State Historic Preservation Officer and the Advisory Council on Historic Preservation and has provided opportunities for consultations with agencies and Native American Tribes as appropriate in accordance with the terms of the Programmatic Agreement.

C.3 Requests for Cooperating Agency Status

This EIS addresses a range of potential activities that are of potential concern to other agencies and to Native Americans. Governmental agencies and Native American tribes participated in the EIS process by submitting scoping comments and may submit comments on this Draft EIS. Representatives of Native American tribes have submitted a document that provides their perspective on the Proposed Action. Moreover, DOE has invited local governments in Nevada to submit reference documents providing information on issues of concern.

DOE is the lead agency for this EIS. Regulations of the Council on Environmental Quality allow the lead agency to request any other Federal agency that has jurisdiction by law or special expertise regarding any environmental impact involved in a proposal (or a reasonable alternative) to be a cooperating agency for an EIS (40 CFR 1501.6 and 1508.5). The regulations also allow another Federal agency to request that the lead agency designate it as a cooperating agency. Finally, the regulations allow state or local agencies of similar qualifications or, when the effects are on a reservation, a Native American Tribe, by agreement with the lead agency to become a cooperating agency (40 CFR 1508.5). Table C-2 lists requests for cooperating agency status and other proposals.

If the lead agency designates a cooperating agency, the lead agency's duties toward the cooperating agency include the following:

- Requesting early participation in the National Environmental Policy Act (that is, EIS) process
- Using any environmental analysis or proposal provided by a cooperating agency with legal jurisdiction or special expertise to the greatest extent possible consistent with its responsibilities as a lead agency
- Meeting with a cooperating agency when the cooperating agency requests

A cooperating agency's duties include the following:

- Participating early in the National Environmental Policy Act process
- Participating in the scoping process
- If requested by the lead agency, assuming responsibility for developing information and preparing environmental analyses including portions of the EIS for which the cooperating agency has special expertise
- If the lead agency requests, making staff support available
- Using its own funds, except the lead agency is to fund major activities or analyses it requests to the extent available

Table C-2. History of requests for cooperating status and similar proposals (page 1 of 4).

Agency	Request/statement/offer	Date	DOE response	Date
U.S. Department of the Navy	Request for cooperating agency status (Guida 1995, all)	May 23, 1995	DOE can draw on existing information from Navy participation in other EISs. DOE will conduct close consultations to ensure accuracy of information used. DOE declines cooperating agency status (Dixon 1995a, all).	July 10, 1995
U.S. Department of the Interior, National Park Service	Request for cooperating agency status (Martin 1995, all)	September 21, 1995	DOE prefers to address NPS comments or issues related to the Death Valley National Park through close consultations between the two agencies. DOE declines cooperating agency status (Dixon 1995b, all).	November 11, 1995
Nye County	Request for cooperating agency status (McRae 1995, all) (Bradshaw 1995, all) (DOE 1997, all) (Bradshaw 1998, all)	August 15, 1995 October 4, 1995 December 5, 1995 July 30, 1998	DOE expresses appreciation for the County's interest and desire to participate, commits to active consultations with Nye County and other entities on selected issues during EIS development, outlines general elements of consultation and coordination contemplated by DOE. DOE declines cooperating agency status (Barnes 1995a, all) (Barnes 1995b, all) (Barrett 1998, all).	November 21, 1995 December 1, 1995 September 24, 1998
Churchill County	Request for cooperating agency status (Regan 1995, all)	May 30, 1995	DOE does not foresee the need to establish formal MOUs to govern Churchill County's or other parties' participation in the NEPA process for the Repository EIS. CEQ and DOE regulations provide sufficient guidance for participation of all affected units of local government and members of the public. DOE describes steps being taken to ensure all interested and potentially affected organizations and individuals have early and equal opportunity to participate in EIS development. DOE declines cooperating agency status (Barnes 1995c, all).	July 21, 1995

Table C-2. History of requests for cooperating status and similar proposals (page 2 of 4).

Agency	Request/statement/offer	Date	DOE response	Date
Lincoln County	Proposal for a cooperative agreement with DOE in assessing the continued development of rail and highway route options to the Yucca Mountain site (Wright 1996, all).	April 22, 1996	DOE expresses appreciation for the County's desire to participate in DOE transportation planning activities, but indicates that, because much of the planning will be done to support the EIS, a cooperative agreement would be unnecessary. DOE identifies active consultation and coordination as an objective of the EIS process (Benson 1996, all).	August 2, 1996
Nuclear Regulatory Commission	NRC does not intend to participate as a cooperating agency (Holonich 1995, all)	March 1, 1995	DOE sent no response to this letter.	NA
Nuclear Regulatory Commission	NRC sent a letter (July 7, 1997) to the Navy. The NRC letter responded to a Navy transmission to the NRC of information on naval spent nuclear fuel. The information had been prepared for EIS use. In its letter, the NRC indicated that it would evaluate the information as part of prelicensing consultations with DOE on waste form issues but that, because NRC is required to review and adopt any EIS submitted as part of a DOE License Application, including information on naval SNF, NRC staff does not intend to formally review and comment on the Navy data. NRC sent DOE a copy of its response to the Navy (Stablein 1997, all).	August 22, 1996	NA	NA

Table C-2. History of requests for cooperating status and similar proposals (page 3 of 4).

Agency	Request/statement/offer	Date	DOE response	Date
U.S. Department of Air Force	Letter from USAF to the State of Nevada, stating that DOE has no obligation to consult with USAF regarding the transportation options DOE elects to evaluate as a result of NEPA public scoping comments, including the Caliente-Chalk Mountain heavy-haul route through Nellis Air Force Range. USAF acknowledged its close interaction with YMP and its intent to "continue this close relationship" (Esmond 1997, all).	September 4, 1997	NA	NA
Council of Energy Resources Tribes	Concept paper for Native American participation in the production of the YMP EIS (Burnell 1996, all).	June 19, 1996	DOE expressed thanks for the concept paper, described the status of the EIS (deferred during Fiscal Year 1996), committed to consideration of comments expressed in the concept paper along with all other comments received during the public scoping process. DOE stated that it would prepare a scoping comment summary and make the summary publicly available, indicated its active consideration of various approaches to consultations with other agencies and Native American tribes, including possible preparation of an EIS-referenceable document (Dixon 1996, all).	July 26, 1995
Advisory Council on Historic Preservation	Expressed thanks for DOE invitation to participate in the EIS process. Indicated desire to assist with development of the EIS and availability to assist DOE in complying with environmental review requirements; expressed intent to provide comments on the draft EIS (Nissley 1995, all).	October 12, 1995	DOE did not prepare a response to this formal scoping comment.	NA

Table C-2. History of requests for cooperating status and similar proposals (page 4 of 4).

Agency	Request/statement/offer	Date	DOE response	Date
Timbisha Shoshone Tribe of Death Valley, California	Letter to President Clinton expressing opposition to YMP; enclosed a Tribal Resolution condemning the siting of YMP; requested active involvement/consultation at a government-to-government level (Boland 1996, all).	August 14, 1996	DOE acknowledged expressed concerns and Tribal Resolution; identified ongoing Native American Interaction Program as vehicle to promote consultations and protection of cultural resources in YMP area; stated that comments from tribal governments were actively solicited during scoping period and Timbisha Shoshone will be afforded opportunity to comment on Draft EIS following its publication (Barnes 1996, all).	11/12/96
National Congress of American Indians	Letter expressed thanks to DOE (Secretary O'Leary) for invitation to meeting of public and private officials to exchange views on DOE management of SNF and radioactive waste, described NCAI as an organization, described Federal Government's fiduciary duty to tribes as sovereign nations, discussed lack of "affected status" for tribes under the NWPA, state Secretary O'Leary's three commitments to Federally recognized tribes in the Yucca Mountain area during the last year, including inclusion in future Yucca Mountain consultations, requested that DOE and Congress mandate a participatory role for tribal governments as part of any proposals to change the NWPA (Gaiashkibos 1995, all).	March 1, 1995	NA	NA

- a. Abbreviations: CEQ = Council on Environmental Quality; MOU = Memorandum of Understanding; NA = not applicable; NCAI = National Congress of American Indians; NEPA = National Environmental Policy Act; NPS = National Park Service; NRC = U.S. Nuclear Regulatory Commission; NWPA = Nuclear Waste Policy Act; SNF = spent nuclear fuel; USAF = U.S. Air Force; YMP = Yucca Mountain Project.

Several agencies, tribes, or tribal organizations have either requested cooperating agency status for this EIS, made comparable proposals for participation, or stated positions in regard to the extent of their participation. Table C-2 summarizes agency requests, proposals, and position statements together with the DOE responses, if appropriate.

REFERENCES

- AIWS 1998 AIWS (American Indian Writers Subgroup), 1998, *American Indian Perspectives on the Yucca Mountain Site Characterization Project and the Repository Environmental Impact Statement*, American Indian Resource Document, Consolidated Group of Tribes and Organizations, Las Vegas, Nevada. [MOL.19980420.0041]
- Barnes 1995a Barnes, W. E., 1995a, "Nye County's Request for Cooperating Agency Designation," letter to The Honorable Cameron McRae (Nye County Commissioners, Tonopah, Nevada), November 21, Office of Civilian and Radioactive Waste Management, U.S. Department of Energy, Las Vegas, Nevada. [MOL.19960424.0182]
- Barnes 1995b Barnes, W. E., 1995b, letter to L. Bradshaw (Nuclear Waste Repository Project Office, Tonopah, Nevada), December 1, Office of Civilian and Radioactive Waste Management, U.S. Department of Energy, Las Vegas, Nevada. [MOL.19960425.0310]
- Barnes 1995c Barnes, W. E., 1995c, "Proposed Memorandum Of Understanding (MOU) Regarding The U.S. Department Of Energy's (DOE) Preparation Of An Environmental Impact Statement (EIS) For A Potential Repository At Yucca Mountain, Nevada," letter to J. Regan (Office of the Churchill County Commissioners, Fallon Nevada), July 21, Office of Civilian and Radioactive Waste Management, U.S. Department of Energy, Las Vegas, Nevada. [MOL.19951220.0136]
- Barnes 1996 Barnes, W. E., 1996, letter to R. F. Boland (The Timbisha Shoshone - Death Valley Land Restoration Project, Death Valley, California), November 12, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, Las Vegas, Nevada. [MOL.19970210.0099]
- Barrett 1998 Barrett, L. H., 1998, letter to L. W. Bradshaw (Department of Natural Resources and Federal Facilities, Nuclear Waste Repository Project Office, Pahrump, Nevada), September 24, Office of Civilian and Radioactive Waste Management, U.S. Department of Energy, Washington, D.C. [MOL.19990610.0300]
- Benson 1996 Benson, A. B., 1996, letter to The Honorable Edward E. Wright (Lincoln County Commissioner), August 2, Office of Public Affairs, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, Las Vegas, Nevada. [MOL.19961115.0045]
- Boland 1996 Boland, R. F., 1996, "Yucca Mountain High Level Nuclear Waste Depository Siting In Nevada Threatens Native American Cultural Resources And Adversely Affects Public Health and Safety," letter to W. J. Clinton (President of the United States), August 14, The Timbisha Shoshone - Death Valley Land Restoration Project, Death Valley, California. [HQO.19961112.0018]

- Bradshaw 1995 Bradshaw, L. W., 1995, letter to Dr. D. Dreyfus (Office of Civilian and Radioactive Waste Management, U.S. Department of Energy), October 4, Nuclear Waste Repository Project Office, Tonopah, Nevada. [MOL.19990319.0217]
- Bradshaw 1998 Bradshaw, L. W., 1998, "Request for Cooperating Agency Status in the Preparation of the Yucca Mountain (YM) Environmental Impact Statement (EIS)," letter to L. Barrett (Office of Civilian and Radioactive Waste Management, U.S. Department of Energy, Washington D.C.), July 30, Department of Natural Resources and Federal Facilities, Nuclear Waste Repository Project Office, Pahrump, Nevada. [MOL.19980903.0847]
- Buchanan 1997 Buchanan, C. C., 1997, "Final Biological Opinion for Reinitiation of Formal Consultation for Yucca Mountain Site Characterization Studies," letter to W. Dixon (U.S. Department of Energy, Yucca Mountain Site Characterization Office), File No. 1-5-96-F-307R, Fish and Wildlife Service, U.S. Department of the Interior, Nevada State Office, Reno, Nevada. [MOL.19980302.0368]
- Burnell 1996 Burnell, J. R., 1996, letter to J. Chirieleison (Office of Civilian Radioactive Waste Management, U.S. Department of Energy), June 19, Council of Energy Resource Tribes, Denver, Colorado. [MOL.19961002.0379, letter; MOL. 19961002.0380, concept paper]
- Dixon 1995a Dixon, W. R., 1995a, "Proposal To Participate as A Cooperating Agency In The Yucca Mountain Site Characterization Office's (YMSCO) Preparation Of An Environmental Impact Statement (EIS) For A Potential Repository At Yucca Mountain, Nevada," interoffice letter to R. Guida, (Office of Naval Reactors), July 10, Office of Civilian and Radioactive Waste Management, U.S. Department of Energy, Las Vegas, Nevada. [MOL.19990610.0298]
- Dixon 1995b Dixon, W. R., 1995b, "Letter Requesting Cooperating Agency Involvement In The Repository Environmental Impact Statement (EIS)," letter to R. Martin, (Death Valley National Park, National Park Service, U.S. Department of the Interior), November 14, Office of Civilian and Radioactive Waste Management, U.S. Department of Energy, Las Vegas, Nevada. [MOL.19960419.0246]
- Dixon 1996 Dixon, W. R., 1996, letter to J. Burnell (Council of Energy Resource Tribes), July 26, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, Las Vegas, Nevada. [MOL.19961015.0306]
- DOE 1988 DOE (U.S. Department of Energy), 1988, *Programmatic Agreement Between the United States Department of Energy and the Advisory Council on Historic Preservation for the Nuclear Waste Deep Geologic Repository Program, Yucca Mountain, Nevada*, Yucca Mountain Site Characterization Office, Nevada Operations Office, North Las Vegas, Nevada. [HQX.19890426.0057]

- DOE 1997 DOE (U.S. Department of Energy), 1997, *Summary of Public Scoping Comments Related to the 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*, Office of Civilian Radioactive Waste Management, Yucca Mountain Site Characterization Office, North Las Vegas, Nevada. [MOL.19970731.0515]
- Esmond 1997 Esmond, M. R., Major General, USAF, 1997, letter to R. Loux (Agency for Nuclear Projects, Nevada Nuclear Waste Project Office), September 4, Department of the Air Force, Nellis Airforce Base, Nevada. [MOL.19971124.0417]
- Gaiashkibos 1995 Gaiashkibos, 1995, letter to H. O'Leary (U.S. Department of Energy), March 1, National Congress of American Indians, Washington, D.C. [MOL.19990610.0304]
- Guida 1995 Guida, R. A., 1995, "Comments On Notice Of Intent For Repository EIS," interoffice letter to L. Barrett (Office of Civilian and Radioactive Waste Management), May 23, Office of Naval Reactors, U.S. Department of Energy, Washington, D.C. [HQO.19950712.0020]
- Holonich 1995 Holonich, J. J., 1995, "Identification Of Lead Contact In Nuclear Regulatory Commission's Review And Comment Of U.S. Department Of Energy's Draft Environmental Impact Statement," letter to R. Milner (Office of Civilian Radioactive Waste Management, U.S. Department of Energy), March 1, High-Level Waste and Uranium Recovery Projects Branch, Division of Waste Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. [MOL.19990610.0301]
- Martin 1995 Martin, R. H., 1995, letter to W. Dixon (Office of Civilian and Radioactive Waste Management, U.S. Department of Energy), September 21, Death Valley National Park, National Park Service, U.S. Department of the Interior, Death Valley, California. [MOL.19960312.0266]
- McRae 1995 McRae, C., 1995, "Cooperating Agency Designation for Nye County in the Preparation of the Yucca Mountain Environmental Impact Statement (EIS)," letter to D. Dreyfus (Office of Civilian and Radioactive Waste Management, U.S. Department of Energy), August 15, Nye County Commission, Tonopah, Nevada. [MOL.19960321.0319]
- Nissley 1995 Nissley, C., 1995, letter to W. Dixon (Office of Civilian Radioactive Waste Management, U.S. Department of Energy), October 12, Advisory Council on Historic Preservation, Washington, D.C. [MOL.19990319.0206]
- Regan 1995 Regan, J., 1995, letter to M. Powell (U.S. Department of Energy), May 30, Office of the Churchill County Commissioners, Fallon, Nevada. [MOL.19990610.0299]

- Stablein 1997 Stablein, N. K., 1997, "Information On Naval Spent Fuel Request," letter to R. Guida, (Naval Nuclear Propulsion Program, U.S. Department of the Navy), August 22, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. [MOL.19990610.0302]
- USAF 1999 USAF (U.S. Air Force), 1999, *Renewal of the Nellis Air Force Range Land Withdrawal: Legislative Environmental Impact Statement*, Air Combat Command, U.S. Department of the Air Force, U. S. Department of Defense, Nellis Air Force Base, Nevada. [243264]
- USN 1996 USN (U.S. Navy), 1996, *Department of the Navy Final Environmental Impact Statement for a Container System for the Management of Naval Spent Nuclear Fuel*, DOE/EIS-0251, in cooperation with the U.S. Department of Energy, Naval Nuclear Propulsion Program, U.S. Department of the Navy, U.S. Department of Defense, Arlington, Virginia. [227671]
- Wright 1996 Wright, E. E., 1996, "Proposal for Lincoln County to Provide input into DOE's Preliminary Transportation Strategies," letter to W. Barnes (Office of Civilian Radioactive Waste Management, U.S. Department of Energy), April 22, Lincoln County Board of County Commissioners, Pioche, Nevada. [MOL.19960905.0149]



Appendix D

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Appendix E

Environmental Considerations for
Alternative Design Concepts and
Design Features for the Proposed
Monitored Geologic Repository
at Yucca Mountain, Nevada

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APPENDIX E. ENVIRONMENTAL CONSIDERATIONS FOR ALTERNATIVE DESIGN CONCEPTS AND DESIGN FEATURES FOR THE PROPOSED MONITORED GEOLOGIC REPOSITORY AT YUCCA MOUNTAIN, NEVADA

E.1 Introduction

E.1.1 OBJECTIVE

This appendix discusses design features and alternatives for a repository at Yucca Mountain in Nevada that were under consideration by the U.S. Department of Energy (DOE) in the winter of 1998 and early 1999. It represents a forward look at how the repository design might evolve to incorporate these and/or other features into a reference design that could be submitted in a repository license application. This appendix also addresses how this design evolution might affect parameters important to the assessment of environmental impacts. The design features and alternatives analyzed as part of the Yucca Mountain Site Characterization Project were conceptual in nature (that is, not developed or analyzed in detail). This appendix presents a qualitative description of the design features and alternatives and a brief assessment of factors associated with each that could cause changes to the environmental impacts analyzed in this environmental impact statement (EIS). This assessment generally indicates that the EIS reasonably represents the foreseeable evolutions in repository design related to environmental impact considerations and bounds potential impacts. Possible design evolutions that occur after DOE issues this Draft EIS will be factored into the Final EIS, as appropriate, and any such refined design concepts will be carried forward to license application if Yucca Mountain is determined to be a suitable site for a repository.

E.1.2 BACKGROUND

DOE has completed the *Viability Assessment of a Repository at Yucca Mountain* (DOE 1998, all). The Viability Assessment included a preliminary design concept (referred to as the *Viability Assessment reference design* throughout this appendix), which presented preliminary design concepts for the repository surface facilities, underground facilities, and waste packages. The Viability Assessment reference design is the same as the high thermal load implementing alternative in the EIS.

Technical work associated with the Viability Assessment and the Viability Assessment reference design was not intended to support the selection of a repository design concept or specific alternative for licensing. Rather, the Viability Assessment identified areas requiring further study to determine site suitability to support a Site Recommendation and a License Application for a repository at Yucca Mountain. One area of further study and evaluation identified in the Viability Assessment was the assessment of alternative repository design features and concepts. The License Application Design Selection Process was established to study a broad range of alternative design concepts and design features to support the selection of the design to be incorporated into a license application.

The License Application Design Selection Process used a multistep approach for evaluating a selected set of features and alternatives against several criteria, including postclosure waste isolation performance, preclosure performance, assurance of safety, engineering acceptance, operations and maintenance, schedule, cost, and environmental considerations. In the first step, features and alternatives are evaluated against these criteria. Following this initial evaluation, enhanced design alternatives (which provide a unique approach to repository design and rely on the attributes of selected design features) were developed. In the development of enhanced design alternatives, there were no limitations placed on the development team to restrict consideration of features and alternatives to those on the initially selected list. From the inception of the License Application Design Selection Process, additional or evolved

alternatives were expected to result. The process called for ranking of the enhanced design alternatives against a selected set of criteria using decision analysis methods. At the time of development of this appendix, enhanced design alternatives that were not part of the Viability Assessment had been developed, but documentation of that development and ranking had not been completed. Therefore, the information presented in this appendix is preliminary and based on both observations of the process and informal discussions with License Application Design Selection Process participants. This appendix will be revised as necessary to incorporate the final results of the License Application Design Selection Process. For the purposes of the License Application Design Selection Process, the following terms were defined:

- **Design Feature.** A design feature is a particular element or attribute of the repository design for which postclosure performance could be evaluated independently of a specific repository design alternative (fully developed design concept) or other design features. An individual design feature could encompass separate discrete concepts or a continuous range of parametric values. Design features can be added singularly or in combination to a design alternative. A design feature could theoretically be applied to any design alternative, although logical compatibility and expected postclosure waste isolation performance enhancement might be evident only when applied to particular design alternatives. Section E.2 of this appendix discusses the design features that were considered in the License Application Design Selection Process.
- **Design Alternative.** Each design alternative represents a fundamentally different conceptual design for the repository, which could potentially stand alone as the license application repository design concept. A design alternative can define major sections or the entire repository design. Design alternatives are distinguished from design features by their complexity and their inclusion of several features. Furthermore, a number of attributes are required to distinguish one design alternative from another. While not mutually exclusive, design alternatives represent diverse and independent methods of accomplishing the repository mission. Section E.2 discusses the design alternatives that were considered in the License Application Design Selection Process.
- **Enhanced Design Alternative.** Enhanced design alternatives are combinations (and/or variations) of one or more design alternative and design feature. While an enhanced design alternative could be made up of any conceivable combination of design alternatives and design features, enhanced design alternatives selected for further evaluation are those combinations that include mutually compatible attributes and expected postclosure waste isolation performance characteristics that exceed those of the basic design alternatives. In other words, the enhanced design alternatives are all improvements to the design alternatives in the first phase of the License Application Design Selection Process, including the Viability Assessment reference design. Other considerations in developing the enhanced design alternatives include the compatibility of the features and alternatives; the developmental, operational, and maintenance simplicity of the resulting combination; and the ability of the set of enhanced design alternatives to address the entire set of design features and alternatives under consideration.

Recommendations for the repository design concept that resulted from the License Application Design Selection Process will be part of a technical report scheduled for completion after this appendix was prepared. The design concept to be carried forward is expected to be one of the five enhanced design alternatives currently identified or minor variations of one of those enhanced design alternatives. Section E.3 of this appendix discusses the enhanced design alternatives that are the subject of consideration in the License Application Design Selection Process.

E.1.3 SCOPE

This appendix discusses the evolution of the EIS repository design concept to the concept that will ultimately be submitted as part of the license application for the Yucca Mountain repository, should the site be approved. The discussion is broken down into three basic categories that reflect the potential types of benefits from the design features and alternatives under consideration. The benefits that could be derived from each of the features and alternatives are not necessarily limited to the categorization presented, and some features and alternatives could fit into more than one category. However, the three categories were chosen to facilitate an understanding of the design evolution process that is presented in the main body of the EIS. Section E.2 discusses the set of selected design features and alternatives.

The categories, as presented in Sections E.2.1 through E.2.3, are Barriers to Limit Release and Transport of Radionuclides; Repository Designs to Control Thermal/Moisture Environment; and Repository Designs to Support Operational and Cost Considerations. Within each category, the text includes descriptions of the features and alternatives, explanations of why each feature/alternative was considered, and discussions of the potential for environmental impacts associated with each feature/alternative.

Section E.3 presents the five enhanced design alternatives that were considered in the first phase of the License Application Design Selection Process to develop a design concept for the proposed Yucca Mountain Repository that was an improvement over the Viability Assessment reference design. This improvement could take many forms, including enhanced licensibility, reduced uncertainty, and ease of construction and operation. The five enhanced design alternatives represent five complete basic design concepts that evolved from consideration of the features and alternatives discussed in Section E.2. The enhanced design alternatives were selected to represent the potential differences in waste isolation performance among differing repository designs. The participants in the License Application Design Selection Process determined that a major factor in selecting the final design for the Yucca Mountain Repository would be the thermal loading of the repository. As such, the five enhanced design alternatives represent a range of thermal loads from 40 metric tons of heavy metal (MTHM) per acre to 150 MTHM per acre. Important differences between the enhanced design alternatives and the Viability Assessment reference design include differences in waste package materials and the addition of a drip shield to each of the enhanced design alternatives. Each of the enhanced design alternatives was selected to improve on the Viability Assessment reference design from a waste isolation performance perspective. As was the case with the basic design features and alternatives discussed in Section E.2, there is the potential for environmental impacts associated with the enhanced design alternatives.

E.2 Design Features and Alternatives

E.2.1 BARRIERS TO LIMIT RELEASE AND TRANSPORT OF RADIONUCLIDES

E.2.1.1 Ceramic Coatings

A thin coating [1.5 millimeters (0.06 inch) or more] of a ceramic oxide on the outer surface of the waste package could increase the life of the waste package by slowing the rate at which the waste package will corrode. Candidate materials for the ceramic coating are magnesium aluminate spinel, aluminum oxide, titanium oxide, and zirconia-yttria. Spinel is the leading alternative.

E.2.1.1.1 Potential Benefits

The ceramic coating could increase waste package life and repository waste isolation performance by reducing corrosion of the waste package surface and, therefore, delaying the release of radionuclides.

E.2.1.1.2 Potential Environmental Considerations

There are no significant environmental considerations associated with ceramic coatings.

E.2.1.2 Drip Shields

Drip shields would provide a partial barrier by diverting infiltrating water away from waste packages in an emplacement drift. Drip shields could be metal (for example, Alloy-22, a nickel-chromium-molybdenum alloy, or titanium-7, a titanium metal alloyed with 0.15 percent palladium) or ceramic-coated metal. One option is to place drip shields under backfill; another is to place the drip shields over the backfill. Drip shields could be implemented with or without backfill.

If the drip shield was placed under backfill, it would fit over the entire length of each waste package, configured to the outer diameter with an unspecified clearance between drip shield and waste package, and enclosed at each end. Backfill, which would be emplaced during the repository's closure, would be comprised of a heaped, single-layered material that covers the waste package and drip shield to some unspecified depth. Another form of backfill, the Richards Barrier, could also be used. Backfill and Richards Barriers are discussed later in this appendix.

The drip shield, as used in the second option, is formed to the approximate backfill surface profile and placed atop the backfill (or Richards Barrier). With this option, the drip shield is placed in conjunction with the placement of backfill at the closure of the repository.

E.2.1.2.1 Potential Benefits

Drip shields are intended to enhance long-term repository performance by reducing waste package corrosion and extending waste package life.

E.2.1.2.2 Potential Environmental Considerations

Additional labor hours would be required for the generation and placement of backfill material, and industrial accidents could increase proportionately. Although drip shields would be emplaced remotely, there could be some incidental radiological doses to workers.

Drip shields of titanium-7, Alloy-22, or other corrosion-resistant material would increase the demand for such materials. Costs for repository closure would increase due to the cost of procuring and installing the drip shields.

E.2.1.3 Backfill

At repository closure, loose, dry, granular material such as sand or gravel would be placed over the waste packages in a continuous, heaped pile. Other materials for backfill, such as crushed rock and depleted uranium, may be evaluated in the future.

E.2.1.3.1 Potential Benefits

Backfill would provide protection of waste packages and drip shields (if placed over the drip shields) from rockfall. It could protect against corrosion of the waste packages by (1) potentially capturing the corrosive salts of various soluble chemicals that might enter with water intrusion, (2) retarding advective flow, and/or (3) increasing the temperature of the emplacement drift to decrease relative humidity.

E.2.1.3.2 Potential Environmental Considerations

Additional workers would be needed, and there would be a potential increase for industrial accidents because of the additional operations. Although backfill would be placed remotely, there could be some incidental radiological doses to workers.

E.2.1.4 Waste Package Corrosion-Resistant Materials

The Viability Assessment reference design for the waste package uses two concentric barrier layers: an outer 100-millimeter (3.9-inch)-thick A516 carbon steel structural corrosion-allowance material, and an inner 20-millimeter (0.8-inch)-thick nickel-based alloy-22 corrosion-resistant material. These two barriers would be expected to provide substantially complete containment of the waste for the lifetime goals established in the Viability Assessment; however, a waste package with the capability to provide substantially complete containment for a significantly extended lifetime would be more desirable.

A variation of the waste package design would replace the corrosion-allowance barrier with a second corrosion-resistant barrier. This design would provide in-depth defense if the second corrosion-resistant barrier was independent of the first (for example, made of a different metal or ceramic). A number of configurations of waste package containers with two corrosion-resistant materials were analyzed, including designs with an inner layer of titanium and outer layer of nickel-based Alloy-22, with a combined thickness of about 55 millimeters (2.2 inches).

E.2.1.4.1 Potential Benefits

Longer waste package lifetimes would lead to improved long-term waste isolation performance of the repository.

E.2.1.4.2 Potential Environmental Considerations

The addition of a second independent corrosion-resistant layer would prolong waste package lifetimes, resulting in delay and minimization of potential groundwater contamination.

Radiological dose to workers would increase without compensating changes in operating procedures, because the total thickness of the waste package container could be less than the Viability Assessment reference design. Appropriate shielding might have to be provided for the workers engaged in waste package handling and emplacement operations. However, there would be a potential increased occupational dose to the workers because the calculated dose rates at the waste package surface would be higher.

E.2.1.5 Richards Barrier

A Richards Barrier would be formed by placing two layers of backfill over the emplaced waste packages at closure. The barrier would consist of a coarse-grained, sand-sized material underlying a fine-grained, sand-sized material. Both materials would be placed as a continuous, heaped pile extending along the alignment of the waste packages. A variety of materials could be used for both layers, including depleted uranium as a coarse-grained material.

The Richards Barrier would be designed to divert water that might enter the emplacement drifts away from the waste packages by transferring the vertical migration of water seepage laterally along the interface between the two layers. The particle size distribution, shape, and porosity of material in the two

layers would provide a permeability difference and would cause the upper layer to channel water seepage along the boundary of the lower layer.

E.2.1.5.1 Potential Benefits

The Richards Barrier would delay the transport of water to the waste packages, thereby delaying waste package corrosion and improving long-term repository performance.

E.2.1.5.2 Potential Environmental Considerations

Dust and equipment emissions could be a concern during the placement phase of the Richards Barrier.

If the chosen coarse material was depleted uranium, there would be an increase in radon emissions. Uranium might also lead to an increase in the contamination of groundwater because the uranium in the Richards Barrier would not be contained or restricted by other engineered barriers. Radiation exposure would also have to be considered in design and operations of depleted uranium handling.

Additional workers would be needed during closure to implement this design feature, and there would be an increased potential for industrial accidents. Although personnel would not be in the drifts, there might be some incidental radiation dose to workers outside the drifts; therefore, additional shielding might be required for personnel.

E.2.1.6 Diffusive Barrier Under the Waste Package

A diffusive barrier would consist of loose, dry, granular material placed in the space between each waste package and the bottom of the emplacement drift to form a restrictive barrier to seepage. Below a critical seepage flux, water would disperse throughout the porous medium of the diffusive barrier, providing both lateral vertical dispersion and thereby slowing the fluid movement to the natural environment. Radionuclides, which might be released from breached waste packages, could become solubilized or suspended within the seepage flow and be retarded by the porous material forming the barrier.

The diffusive barrier could be anything from common sand to gravel-size material without any special qualifications to mineralogy, grain size distribution, shape, or density. Depleted uranium could also be used. The diffusive barrier would be installed prior to waste emplacement.

E.2.1.6.1 Potential Benefits

Improved waste isolation performance could be achieved by slowing radionuclide movement to the natural environment.

E.2.1.6.2 Potential Environmental Considerations

If the diffusive barrier material were depleted uranium, there would be increased radon emissions and increased radiological dose to workers. There could be an increase in the contamination of groundwater because the uranium would not be contained or restricted by other engineered barriers.

Additional workers would be needed to construct the diffusive barrier; therefore, there would be a proportional increase in the potential for industrial accidents.

E.2.1.7 Getter Under Waste Packages

A getter would be a fine-grained material [either phosphate rock (apatite) or iron oxide (hematite, goethite, etc.)] with an affinity for radionuclides. This material would be placed in the invert recess below the waste packages prior to waste emplacement.

E.2.1.7.1 Potential Benefits

A getter material below the waste packages could improve long-term waste isolation through retardation of radionuclide movement from the repository drifts.

E.2.1.7.2 Potential Environmental Considerations

Additional workers would be needed to place the getter material in the drifts; therefore, there would be a proportional increase in the potential for industrial accidents.

E.2.1.8 Canistered Assemblies

Placing spent fuel assemblies in canisters at the Waste Handling Building before inserting them into waste packages would provide an additional barrier and further limit mobilization of radionuclides if the waste package is breached. The canisters would be fabricated from a corrosion-resistant material (for example, Alloy-22 or a zirconium alloy). There are three general concepts for the placement of fuel assemblies in canisters:

- Rectangular canisters designed to hold individual fuel assemblies: these canisters could be placed into a waste package with a basket containing neutron absorber and aluminum thermal shunts, similar to the current basket designs.
- Rectangular canisters designed to hold a few fuel assemblies: these canisters could have neutron absorber between assemblies and fit into a basket containing neutron absorber and aluminum thermal shunts.
- Large circular canister designed to hold multiple fuel assemblies and fit one per waste package: the canister would have an internal basket with neutron absorber, aluminum thermal shunts, and fuel tubes, similar to previous canistered fuel waste package designs.

E.2.1.8.1 Potential Benefits

Placing spent fuel assemblies in canisters before inserting them into waste packages would provide an additional barrier and limit mobilization of radionuclides in breached waste packages.

E.2.1.8.2 Potential Environmental Considerations

Use of this feature could cause an increase in the size of the Waste Handling Building and require additional workers. There would be an increase in operations and a possible increase in the number of lifts required per fuel assembly. This increase could be as much as one extra lift per assembly (canister), due to the moving of the canister to the waste package, which would lead to the potential for greater exposure to radiation for workers.

Implementation of this feature could increase the amount of rejected materials due to faulty welding, potentially generating more low-level radioactive waste and/or solid waste.

E.2.1.9 Additives and Fillers

Additives and fillers are materials that could be placed into waste packages (in addition to those normally required for the basket material) to fill the basket and waste form void spaces. The additives and fillers would:

- Sorb radionuclides and retard their release from a breached waste package
- Sorb boron neutron absorber that might be released from corrosion of the borated stainless steel absorber plates
- Displace moderator from the interior of the waste package to provide additional defense-in-depth for nuclear criticality control

Potential additives and fillers would be oxides of iron and aluminum. These materials could be placed within the waste package as a powder or as shot following loading of the waste form, or integrated into the basket design.

E.2.1.9.1 Potential Benefits

Additives and fillers could improve long-term repository performance by retardation of release of radionuclides to the groundwater and could also improve long-term criticality control.

E.2.1.9.2 Potential Environmental Considerations

Adding additives and fillers would make it more difficult to remove spent nuclear fuel assemblies from waste packages following retrieval, if necessary. Operations would have to include the additional step of removing this material before removal of the fuel.

E.2.1.10 Ground Support Options

Ground support in the repository ensures drift stability before closure. Selection of ground support options could affect repository waste isolation performance. Considerations of ground support options include functional requirements for ground support, the use of either concrete or steel-lined systems, and the feasibility of using an unlined drift ground support system with grouted rock bolts.

A concrete lining has been studied for its structural/mechanical behavior and subjected to the load conditions expected of emplacement drifts. However, a number of postclosure performance assessment issues related to the presence of concrete within the emplacement drift environment have been identified.

An all-steel ground support system (for example, steel sets with partial or full steel lagging) has been considered to be a viable ground support candidate for emplacement drifts. Use of an all-steel lining system would provide a means of limiting or eliminating the introduction of cementitious materials (that is, concrete, shotcrete, or grout), including organic compounds into the emplacement drift environment. The potential for corrosion of steel subjected to the emplacement drift environment is a concern with this system. Another concern is the interaction of steel ground supports with waste package materials.

For an unlined drift scenario, rockbolts and mesh could be considered as permanently maintainable ground support. Design and performance advantages associated with the use of rockbolts as permanent ground support for emplacement drifts include durability and longevity of this system. A postclosure concern would be the suitability of cementitious grout, which would be used for installing rockbolts.

E.2.1.10.1 Potential Benefits

Safety during emplacement and potential retrieval would be enhanced by use of appropriate ground supports. Long-term repository performance could be improved by reducing or delaying damage to canisters from rockfall, because damaged areas would be locations for enhanced corrosion even if the canister was not breached by the rockfall.

E.2.1.10.2 Potential Environmental Considerations

The choice of ground support options does not significantly impact any environmental consideration except for long-term repository waste isolation performance.

E.2.2 REPOSITORY DESIGNS TO CONTROL HEAT AND MOISTURE

E.2.2.1 Design Alternative 1, Tailored Waste Package Spatial Distribution

Tailored spatial distributions of waste packages within the repository block emplacement drifts could improve the postclosure waste isolation performance of the repository. The EIS design assumes the various waste package types would be emplaced on a random basis, modified only to meet the areal mass loading requirement of 25 to 85 MTHM per acre and the commercial fuel cladding and drift wall thermal goals of 350°C and 200°C (662°F and 392°F), respectively. There are three different methods of spatial distribution under review, including:

- Distribution of waste packages as a function of infiltrating water percolation rate within various regions of the repository block. Higher heat-producing packages would be placed in areas with higher percolation rates.
- Distribution of commercial spent nuclear fuel waste package types as a function of the distance to the water table and/or unsaturated zone zeolite content. Waste packages with radionuclides with the highest tendency to travel would be placed furthest from the water table, and waste packages with radionuclides with a higher tendency to be sorbed would be placed above areas with the highest zeolite content.
- Grouping waste package types into categories of hot, medium, and cold waste packages to even out the temperature differences across the repository.

E.2.2.1.1 Potential Benefits

Tailoring spatial distribution of the waste packages within the repository block might improve the performance of waste packages by delaying and reducing contact of water and/or increasing sorption of released radionuclides by zeolites in the unsaturated zone. This form of distribution has the potential to improve repository waste isolation performance.

E.2.2.1.2 Potential Environmental Considerations

Larger surface storage facilities could be needed to allow appropriate selection of waste packages for the desired spatial distribution. However, if the retrieval pad can be used for this purpose, no additional land would be needed.

E.2.2.2 Design Alternative 2, Low Thermal Load

The low thermal load design alternative would limit the temperature of the drift wall and host rock. It would cause less thermal change in the host rock than the Viability Assessment reference design. Limiting temperature rise would also reduce the uncertainty in predicting several processes, and thermal, chemical, mechanical, and hydrological effects would be easier to describe because coupling of these effects would extend over a smaller region than the Viability Assessment reference design. In this evaluation, a low thermal load refers to 40 MTHM per acre.

- *Option 1.* The waste package spacing would be the same as the spacing of the drifts, creating a square area between waste packages. The spacing of waste packages would be farther apart than in the Viability Assessment reference design. This option is the equivalent of the low thermal load implementing alternative analyzed in the EIS.
- *Option 2.* The spacing of the waste packages within the drifts would be 9 meters (30 feet) as in the Viability Assessment reference design, but drift spacing is increased to about 90 meters (300 feet). This can be compared to 28 meters (92 feet) for the Viability Assessment reference design.
- *Option 3.* This option consists of a greater number of smaller waste packages than in Option 1 or 2, and spacing of waste packages within the drifts is similar to Option 2. Drift spacing and excavated rock volume are about the same as for Option 1.

E.2.2.2.1 Potential Benefits

The primary benefit would be the reduction in uncertainties associated with higher thermal loads and the elevated temperature of the host rock. Lower repository temperatures could also potentially reduce waste package material corrosion rates.

E.2.2.2.2 Potential Environmental Considerations

Options 1 and 3 would result in generation of more excavated rock compared to the Viability Assessment reference design, and therefore requires a larger area for storage/disposal of excavated rock. Subsurface costs would increase. Option 2 would result in less volume of excavated rock than Option 1 or 3.

E.2.2.3 Design Alternative 3, Continuous Postclosure Ventilation

Under this alternative there would be continuous ventilation of the emplacement drifts during the postclosure period. Ventilation would occur by natural ventilation pressure induced by the difference in air density between hot and cool areas. Three primary options were considered:

- Closed loop airways connected underground but sealed to the surface
- Open loop airways where the primary airways stay open and in which the repository drifts are open to exchange air with the atmosphere; two additional ventilation shafts would be needed
- Open/closed loop ventilation where primary airways would be sealed, but drifts would be located very close to a system of tunnels open to the atmosphere

E.2.2.3.1 Potential Benefits

Postclosure ventilation would increase the removal of moisture from air around the waste packages for a period of time (estimated to be 1,000 to 2,000 years for the closed loop system), but moisture would eventually reestablish itself. Reduced moisture could improve performance by retarding waste package corrosion.

E.2.2.3.2 Potential Environmental Considerations

Excavated rock piles would increase in size in proportion to the increase in drift excavation required. Additional shafts would result in additional surface disturbed areas (small, relative to the Viability Assessment reference design). Additional occupational exposure to radon-222 associated with excavation would occur.

Overall, work force would increase by less than 10 percent, as would associated impacts such as industrial accidents.

E.2.2.4 Design Alternative 6, Viability Assessment Reference Design

The Viability Assessment reference design is equivalent to the high thermal load alternative evaluated in the EIS.

E.2.2.5 Design Alternative 7, Viability Assessment Reference Design with Options

The Viability Assessment reference design with options was considered as a design alternative in the License Application Design Selection Process. The Viability Assessment reference and design is analyzed in detail in the EIS. Options considered include ceramic coatings, drip shields, and backfill (see Sections E.2.1.1, E.2.1.2, and E.2.1.3, respectively).

E.2.2.6 Aging and Blending of Waste

Pre-emplacment aging and blending of wastes provides mechanisms for managing the thermal output of a waste package and the total thermal energy that must be accommodated by the repository.

Aging the waste before emplacement results in less variable (over time) thermal output of the waste packages and lower waste package temperatures. Aging could be performed at the repository, at the reactor sites, or at other locations.

Blending would allow a more uniform heat output from the waste packages. Blending would be accomplished by selecting waste forms for insertion in waste packages based on their heat output to minimize the variability in the thermal energy of each waste package.

E.2.2.6.1 Potential Benefits

Aging would reduce the temperature increase expected at the surface above the repository because the total heat load of the repository would be decreased. Lower heat output could also result in a smaller repository footprint by allowing more dense waste emplacement schemes without violating waste package or drift wall temperature goals. Both blending and aging reduce the variability of the temperature distribution in the repository, and drifts might be spaced more closely. Lower and equalized temperatures could improve structural stability of the drifts. Aging and blending would improve waste package

stability (reducing rockfall-induced damage and corrosion) and improve long-term repository performance.

E.2.2.6.2 Potential Environmental Considerations

The blending feature might require a significantly larger storage pool size. This would increase the size of the pool storage building, and result in correspondingly higher costs. The Viability Assessment reference design staging pools have the capacity for about 300 MTHM. This would be reconfigured and expanded to allow for storage of up to 6,500 MTHM. Expanded pool storage would require additional resources (steel, concrete, gravel and asphalt, fuel, electricity and water for construction and operation, but the increases would not be significant (about 10 percent). Waste generation would also increase. During operations, use of well water will increase by about 15 percent. Well water is used to replace evaporative losses in the pools. Land use does not increase. Increases in worker population mean an increase in the potential for industrial accidents. Cumulative annual dose to workers would increase slightly, but the average dose to workers would not increase.

If aging is done at the Yucca Mountain site, a surface storage facility would be required. The effects of the aging feature are identical to the retrieval contingency discussed in the EIS because the same size storage facility/pad would be needed. The retrieval contingency assumes a surface storage facility able to handle the entire repository inventory.

E.2.2.7 Continuous Preclosure Ventilation

Continuous preclosure ventilation would provide increased air flow in the emplacement drifts compared to the reference design preclosure ventilation rate of 0.1 cubic meter (3.5 cubic feet) per second. The system would be shut off at closure.

Additional excavation would be required for an additional exhaust main. The actual number of emplacement drifts would not change, but the layout of drifts would vary slightly to accommodate the additional ventilation shafts. The sizes of the shafts would have to be increased and more would need to be added. Access drifts and additional connections would have to be added between the exhaust mains and the shafts.

E.2.2.7.1 Potential Benefits

Continuous ventilation in the preclosure period could reduce the rock wall and air temperature. It could also remove enough moisture to reduce the length of time the waste packages are exposed to temperature/moisture conditions that could result in higher corrosion rates. The removal of moisture also would increase the stability of the ground-support system. In addition, with lower drift temperatures retrieval would be easier.

E.2.2.7.2 Potential Environmental Considerations

Additional drifts and intake and exhaust shafts would be required to handle the additional airflow quantities, resulting in additional excavated rock. Additional shaft locations would disturb land surface in the limited locations available to place the shafts, and roads would have to be constructed to the shaft sites. Additional shafts and night lighting at the top of the mountain might be visible from off the Yucca Mountain site.

The changes in repository ventilation would increase emissions of naturally occurring radon-222 and its radioactive decay products in the air exhausted from the subsurface. Power requirements could increase substantially during emplacement operations and postclosure monitoring.

The number of workers would increase by less than 10 percent, with an attendant increase in the potential for industrial accidents.

Closure would be more difficult because there would be additional openings to seal.

E.2.2.8 Drift Diameter

The emplacement drift diameter is a secondary design feature because the diameter is determined by a number of primary design features. The size of the emplacement drift could directly affect design considerations such as opening stability (rockfall potential), the extent of the mechanically induced disturbed zone, and the amount and location of seepage into the drifts.

The drift diameter for the Viability Assessment reference design is 5.5 meters (18 feet). A range of drift diameters is being considered [from 3.5 meters (11 feet) to 7.5 meters (25 feet)].

E.2.2.8.1 Potential Benefits

A smaller diameter drift is inherently more stable and could reduce the need for ground-support systems, potentially reducing costs. The smaller drift diameter would also be less susceptible to water seepage. A larger diameter allows for other modes of emplacement, such as horizontal or vertical borehole emplacement. Both of these emplacement modes would reduce the potential for damage to waste packages from rockfall, therefore potentially improving long-term performance of the repository.

E.2.2.8.2 Potential Environmental Considerations

An increase in drift diameter could increase the potential for rockfall (both size and frequency) and decrease the overall opening stability. Rockfall could breach waste packages or cause lesser damage to the packages, providing locations for accelerated corrosion. Also, the larger the drift diameter, the more vulnerable it would be to water entry from seepage flow.

A smaller drift diameter would be inherently more stable in highly jointed rock and a decreased rockfall size would be anticipated. A change to a smaller diameter could allow modification to the ground-support system with possible elimination of a full circle drift liner. Although a smaller drift diameter would be less susceptible to seepage, the smaller diameter drift might result in short-term increases of temperature, which could affect the characteristics of potential groundwater movement.

Increasing the emplacement drift diameter would result in an increase in the quantity of excavated rock and increased use of equipment and materials, higher releases of radon-222, and lower ventilation air velocity. The lower air velocity would result in greater quantities of radon-222 and dust during development, an important consideration for preventing suspension of respirable silica dust.

A smaller drift diameter, although reducing the potential of radon-222 releases, might not be able to provide the quantities of air necessary for ventilation without raising velocities to undesirable levels. Increased drift diameter would require more workers for tunnel boring machine operations, excavated rock handling, ground-support installation and finishing works, surface equipment operators, and maintenance. A decrease in the drift diameter would have an opposite affect on the worker requirements;

that is, with a larger drift diameter, the additional excavation work would produce an increase in worker accidents. Larger tunnel boring machines could require substantially more electrical power.

E.2.2.9 Drift Spacing and Waste Package Spacing

In repository design, thermal load refers to a density at which the waste packages will be emplaced in the repository. The Viability Assessment reference design involves emplacement of waste packages in drifts in a horizontal mode, and thermal load is directly related to the emplacement drift and waste package spacing. The Viability Assessment reference design used a spacing of 28 meters (92 feet) between drifts.

For a given drift spacing, emplacement of waste packages can be arranged by using point load (waste package spacing determined based on individual waste package characteristics, such as mass content or equivalent heat output of each waste package), or line load [waste packages are emplaced nearly end to end that is, with a 0.1-meter (0.3-foot) gap with no considerations of individual waste package characteristics].

The point load approach was used for the Viability Assessment reference design. Waste-package spacing was determined based on mass content of waste packages, to achieve an overall area mass loading of 85 MTHM per acre for commercial spent nuclear fuel.

The line load method would be expected to provide a more intense and uniform heat source along the length of emplacement. An increase in emplacement drift spacing would be required in conjunction with line loading to maintain a constant overall thermal loading density (for example, 85 MTHM per acre).

E.2.2.9.1 Potential Benefits

The line load approach would keep the emplacement drifts hot and dry longer and would decrease the amount of water that could contact waste packages. Consequently, waste package performance could be improved. The line load approach would also reduce the number of emplacement drifts needed for waste emplacement. However, the concentrated heat load in the drifts could require continuous ventilation of emplacement drifts to meet the near-field temperature requirements. Continuous ventilation is discussed in Section E.2.2.7.

E.2.2.9.2 Potential Environmental Considerations

Line loading would require excavation of about 30 fewer emplacement drifts, with correspondingly less excavated rock, dust, and pollutants from diesel- and gasoline-powered equipment and vehicles. Decreased excavation would also reduce radon-222 release in the underground facility. However, decreasing the waste package spacing would result in potentially large increases in the rock temperatures in and near the emplacement drifts. This could create the need for continuous ventilation of emplacement drifts, which could increase emissions of naturally occurring radon-222 and its radioactive decay products in the air exhausted from the subsurface.

The reduction in total work and material requirements would be expected to be linearly proportional to the reduction in required drift length. Fewer work hours would also result in less potential for industrial accidents during construction. Decreased emplacement drift excavation would reduce the demand for electric power, equipment fuel, construction materials, and site services. However, the higher drift temperature associated with the line load option could require continuous ventilation of emplacement drifts.

E.2.2.10 Near-Field Rock Treatment

Near-field rock treatment involves injection of a grout material into the cracks in a portion of the rock above each emplacement drift to reduce the hydraulic conductivity of the treated rock. Injection would start at least 6 meters (20 feet) above the drift crown and would form a zone at least 4 meters (13 feet) thick, extending at least 6 meters on each side of the drift. Injection would be through holes 2.5 to 5 centimeters (1 to 2 inches) in diameter drilled from inside each drift prior to waste emplacement. Injection pressures would not exceed a certain minimum pressure, selected to limit rock fracturing or joint opening.

The candidate materials include Portland cement grout, sodium silicate, bentonite (a clay), and calcite.

E.2.2.10.1 Potential Benefits

Reducing the hydraulic conductivity of the rock would improve long-term repository performance by reducing or retarding postclosure water seepage into the drifts.

E.2.2.10.2 Potential Environmental Considerations

Installation of the grout material would require additional labor hours, with an associated change in the potential for industrial accidents.

E.2.2.11 Surface Modification – Alluvium Addition

Covering the surface of Yucca Mountain above the repository footprint with alluvium could decrease the net infiltration of precipitation water into the repository by increasing evapotranspiration. To cover the mountain with alluvium, the surface of the mountain would be modified to prevent the alluvium from washing away. Ridge tops on the eastern flank of Yucca Mountain would be removed and the excavated rock placed in Solitario Canyon and in Midway Valley or used to fill the alluvium borrow pit. The maximum slope of the ground surface remaining would be approximately 10 percent. Alluvium [approximately 2 meters (7 feet) thick] would be placed on the new surface and vegetation would be established. New haul roads to move the necessary materials would have to be constructed.

E.2.2.11.1 Potential Benefits

Reduced net infiltration would improve long-term repository performance. However, there is uncertainty about the permanence of both the vegetation and the alluvium that would be added to the surface of Yucca Mountain.

E.2.2.11.2 Potential Environmental Considerations

Approximately 8 square kilometers (2,000 acres) on Yucca Mountain would be resloped and covered. The excavated material would cover 4.8 square kilometers (1,200 acres) in the fill area in Solitario Canyon. The borrow pit would be about 5.2 square miles (1,300 acres). Additional access roads would also be needed. Yucca Crest would be lower by approximately 30 to 60 meters (98 to 197 feet) the ridges on the east side of Yucca Crest would be lowered by as much as 80 meters (262 feet). Quantities of material to be moved would include:

- Total rock cut from Yucca Mountain 220 million cubic meters (17,600 acre-feet)
- Total alluvium removed from the alluvium borrow pit (probably in Midway Valley) about 22 million cubic meters (17,600 acre-feet)

The operation would be equivalent to a major, large-scale open pit mining operation. It would likely require a labor force of about 75 people per shift. There would be an increase in the potential for industrial accidents because of the additional work. Generation of particulate emissions (fugitive dust) and gaseous criteria pollutant emissions from vehicles would increase.

There would be alterations to natural drainage; however, the potential for flooding would not increase with proper design.

The view to and from Yucca Mountain would be altered. Mining operations at the top of the mountain would be visible for some distance, and the mountain would be considerably lower. Vegetation would be restored because the design requires vegetation as part of the evapotranspiration process. The operation would be carried out on three shifts, and night lighting on the top of the mountain could be visible to the public.

E.2.2.12 Surface Modification – Drainage

Surface modification could reduce infiltration at the surface of the mountain. Net infiltration into Yucca Mountain could be significantly decreased if the thin alluvium layer over the footprint of the repository were removed to promote rapid runoff of the surface water. It has been shown that where the alluvium is thin, it retains the surface water and allows it to infiltrate into the unsaturated zone. Where bedrock is exposed on slopes, the water runs off rapidly and net infiltration is very small or reduced to zero.

The thin alluvium layer would be stripped from the topographic surface above the repository footprint and a 300-meter (984-foot) buffer surrounding it.

E.2.2.12.1 Potential Benefits

Reduced infiltration would result in improved long-term repository waste isolation. However, there is uncertainty about the permanence of alluvium removal. In addition, while infiltration might be reduced on the top of the mountain, infiltration could increase in other areas because of the higher volumes of surface water runoff.

E.2.2.12.2 Potential Environmental Considerations

The amount of land modified to improve drainage would be approximately 1,100 acres, located mainly on the eastern flank of Yucca Mountain. Additional road construction would also be required. The removed alluvium, about 2.1 million cubic meters (2.7 million cubic yards), would be placed in Midway Valley. There would be alterations to natural drainage, and the increased runoff could increase the potential for flooding. The landforms would be changed only slightly because of the thin [less than 0.5-meter (1.6-foot) thick] alluvium that would be removed. Any existing vegetation on the side of the ridges would be removed during the process of alluvium removal. Bare bedrock would be exposed, which would discourage vegetation from growing except from cracks in the rock.

Additional workers would be required, and there would be an accompanying increase in the potential for industrial accidents.

Night lighting would be needed to support this operation that could be visible from off the site.

E.2.2.13 Higher Thermal Loading

Higher thermal loading would keep the drift temperature above the boiling point longer, thereby minimizing the amount of moisture around the waste package during a longer postclosure period. The higher thermal loading could also have adverse effects on the surrounding rock. This feature could also be combined with aging to achieve greater mass loading per acre of repository area.

Higher thermal loads could be achieved by either decreasing drift spacing, by placing waste packages closer together in the drift, or by a combination of drift spacing and waste package spacing. In all three cases, the increased number of waste packages in a given area would result in a higher thermal load to a given area of the repository.

The benefits and environmental considerations associated with this feature would be similar to those discussed under Drift Spacing and Waste Package Spacing (Section E.2.2.9).

E.2.3 REPOSITORY DESIGNS TO SUPPORT OPERATIONAL AND/OR COST CONSIDERATIONS

E.2.3.1 Design Alternative 4, Enhanced Access

The purpose of the enhanced access design would be to provide additional shielding around the waste package to allow for personnel accessibility during waste package loading, transfer to the drift, emplacement, and performance confirmation. Shielding would lower the dose rate to less than 25 millirem per hour. Enhanced access could be provided by:

- Additional shielding integral to the waste package
- Supplemental (separate from the waste package) shielding in the emplacement drifts only
- Portable shielding for personnel to access the drift

E.2.3.1.1 Potential Benefits

The major benefit of these three options would be to provide access to the emplacement drifts so personnel could carry out performance confirmation activities. Enhanced access designs could also offer increased access for maintenance and ease of operations, and the potential elimination of some remote handling equipment. If shielding were left in place at closure, it could provide additional protection for waste packages from rock falls.

E.2.3.1.2 Potential Environmental Considerations

Increased personnel access would increase occupational exposure, even with the additional shielding. Enhanced access would decrease the number of observation and performance confirmation drifts needed, and slightly decrease the volume of excavated rock piles.

The addition of shielding to waste packages would result in increased materials usage. Shielding materials could be steel, concrete, magnetite concrete (concrete with iron shot included), or Ducrete® (concrete with depleted uranium included).

E.2.3.2 Design Alternative 5, Modified Waste Emplacement Mode

In a modified waste emplacement design, unshielded waste packages would be emplaced in a configuration in which the repository's natural or engineered barriers would provide shielding. Examples

include placing waste packages in boreholes drilled into the floor or wall of emplacement drifts, in alcoves off the emplacement drifts, in trenches at the bottom of the emplacement drifts, or in short cross drifts excavated between pairs of excavated drifts. In each case, some type of cover plug would be used to shield radiation in the emplacement drifts.

Unshielded waste packages, which in some designs would have a smaller capacity than specified in the Viability Assessment reference design, would be used.

E.2.3.2.1 Potential Benefits

Natural or engineered barriers would enhance human access, reduce performance confirmation costs, and facilitate conducting inspections and maintaining ground support. Retrieval operations would also be easier because of easier access.

E.2.3.2.2 Potential Environmental Considerations

The footprint of the repository would not change, but the amount of excavated rock would increase. The vertical borehole emplacement concept would generate the most additional excavated rock. Peak power consumption would increase substantially because of the use of additional boring machines.

E.2.3.3 Design Alternative 8, Modular Design (Phased Construction)

Modular design is an alternative that could reduce annual expenditures during construction if annual funding is constrained below that required for the Viability Assessment reference design. This alternative would include staged modular construction of repository surface and subsurface facilities.

The modularized Waste Handling Building would be designed to handle specific types of waste forms and quantities. The modular concept would include one Waste Handling Building completed in modular phases or two separate buildings constructed in sequence.

E.2.3.3.1 Potential Benefits

The primary benefit would be leveled cash flow during construction.

E.2.3.3.2 Potential Environmental Considerations

The dual buildings would increase the overall size of the Waste Handling Building by an estimated 10 percent. The Radiologically Controlled Area could increase by about 10 percent or less. Operating times (years of operation) would be extended and operations would be at a lower rate.

Some options would involve receipt of spent nuclear fuel from reactor sites prior to the start of emplacement that could increase worker dose because it would have to be handled twice.

E.2.3.4 Rod Consolidation

Both pressurized-water reactor and boiling-water reactor fuel assemblies have fuel rods arranged in regular square arrays with rod-to-rod separation maintained by the fuel assembly hardware. Rod consolidation would involve eliminating this separation and bringing the fuel rods into close contact. Reducing the volume taken up by fuel assemblies would allow the capacity of waste packages to be increased and/or the size of waste packages to be reduced. Consolidation could be done at either the current spent fuel storage locations or at the repository.

Rod consolidation would be accomplished by removing fuel rods from an assembly, repackaging the rods in a denser arrangement in a suitable canister, and loading the new canister into a waste container. This process could occur either in a pool or in a dry (hot cell) environment.

E.2.3.4.1 Potential Benefits

A reduced number or size of waste packages would be possible and could result in reduced emplacement costs. If rod consolidation took place at the reactor sites, waste transportation requirements might be reduced.

E.2.3.4.2 Potential Environmental Considerations

Because of the disassembly operations, the size of the Waste Handling Building would more than double in area if rod consolidation were done at the repository. With the large number of fuel rod handling operations in the hot cells, there would be a greater potential for radiological releases due to fuel handling accidents (such as dropping a fuel rod/assembly).

The number of workers at the repository could increase if rod consolidation were performed at the repository. With an increase in the number of fuel handling operations, the number of fuel handling accidents would increase and result in a small increase in radiological exposure for onsite workers.

Approximately 10 to 40 kilograms (22 to 88 pounds) of leftover, nonfuel components from each as-received fuel assembly would be packaged as Class C or Greater-Than-Class-C low-level wastes. In addition, low-level waste would be generated by decontamination and disposal of equipment. Low-level waste would be transported to the Nevada Test Site or other appropriate facility for disposal. Greater-than-Class-C wastes could be disposed of offsite or in the repository with approval of the U.S. Nuclear Regulatory Commission.

Waste packages containing consolidated fuel rods might result in higher cladding temperatures, which could damage the cladding and have negative impacts on waste isolation performance.

E.2.3.5 Timing of Repository Closure

The first option assumes that the subsurface facilities would be fully maintained to the same level of readiness during the 300-year period as planned for the 100-year period assumed for the Viability Assessment reference design. There would be continuous ventilation during the entire 300-year period. The second option assumes the Nuclear Regulatory Commission would have approved completion of the Performance Confirmation Program at the end of the first 100 years, and that continued access to the emplacement drifts would no longer be required. The second option considers that ventilation, maintenance, and repairs would be reduced to a minimum for cost considerations, but that temperatures would be maintained at 50°C (122°F) or less for human access to the subsurface (nonemplacement) facilities.

E.2.3.5.1 Potential Benefits

Extending the period before final closure would allow for reduction of waste package heat output, extended monitoring, and extended retrieval period for the waste.

E.2.3.5.2 Potential Environmental Considerations

Delayed closure of the repository would lengthen the time that land would remain disturbed through the occupation of surface facilities necessary to support extended operations from 100 to 300 years. It would delay the reclamation of surface stockpiles retained for filling the mains, ramps, and shafts.

The release of radon-222 from excavations is proportional to time. Delayed closure from 100 to 300 years would increase the emissions of radon-222 by a factor of approximately 3.6.

The number of workers required for monitoring would not change. However, the number of labor hours required, compared to the Viability Assessment reference design monitoring period, would be 3.6 times the number required for closure at 100 years. The base case scenario requires the periodic retrieval of waste packages for performance confirmation testing. An increase in the monitoring period from 76 to 276 years would increase radiation exposure due to increased waste package handling. More frequent inspections would be likely during this extended period due to aging. Additionally, emplacement drifts maintenance would require removal and re-emplacment of waste packages. An increased monitoring period would increase the potential for industrial accidents and radiological exposure.

E.2.3.6 Maintenance of Underground Features and Ground Support

A maintenance program in the emplacement drifts would be needed to accommodate an extended long-term repository service life and to reduce the risk of keeping the repository open for an additional 200 years. Repository emplacement drift ground support components would have to be designed and maintained for a service life of greater than 300 years, including closure and retrieval times.

E.2.3.6.1 Potential Benefits

The benefits are the same as those listed in Section E.2.3.5.1

E.2.3.6.2 Potential Environmental Considerations

Some types of maintenance in the emplacement drifts would require retrieval of waste packages for maintenance access. Blast cooling would be needed to lower the temperature to below 50°C for worker access. There could be additional radiological exposure to workers.

E.2.3.7 Waste Package Self-Shielding

In the Viability Assessment reference design, handling of waste packages in the emplacement drifts would be performed remotely, and human access to the emplacement drifts would be precluded when waste packages are present. Waste package self-shielding would reduce the radiation in the drifts to levels such that personnel access would be possible. This would allow direct access to the performance confirmation instrumentation, and maintenance and repair in the drifts.

Self-shielding would be accomplished by adding a shielding material around the waste packages. Candidate materials include A516 carbon steel, concrete with depleted uranium (Ducrete®), magnetite concrete, and a composite material of boron-polyethylene and carbon steel.

The amount of shielding would depend on the target radiation dose level in the drift environment. For a 25-millirem-per-hour waste package contact dose, the estimated thickness of the concrete would be about 0.6 meter (2 feet). For higher contact doses, less shielding material would be required.

E.2.3.7.1 Potential Benefits

Monitoring, maintenance, and retrieval would be easier with contact handling of the waste packages.

E.2.3.7.2 Potential Environmental Considerations

Self-shielding could not be used with high thermal loading because the shielding would provide a thermal barrier that would result in excessive fuel cladding temperature. Smaller waste packages would maintain a constant outside diameter but would also require about four times as many waste packages and more drifts. Radon-222 emissions would increase in proportion to the additional excavation.

Concrete shielding would be applied at the repository, and the number of workers would slightly increase, as would the number of industrial accidents. There could be a reduction in radiological exposure to workers during emplacement operations. The concrete shielding could degrade the long-term performance of the waste packages.

E.2.3.8 Repository Horizon Elevation

This feature considers a two-level repository to increase repository capacity without moving out of the characterized area.

One two-level concept would divide the Viability Assessment reference design layout along a north-south axis and would relocate the western half above the eastern half. A second two-level concept would duplicate the Viability Assessment reference design layout 50 meters (164 feet) above the current footprint. The thermal loading of each level could be adjusted to increase the capacity.

E.2.3.8.1 Potential Benefits

There would be two potential advantages to repository long-term performance. Increased thermal load would potentially enhance the umbrella effect (this could reduce the amount of water that could come in contact with the waste package). There would also be added flexibility in emplacing waste packages on the lower level, which could be shielded from moisture infiltration by the upper level horizon.

Retrieval could be accomplished more quickly due to the ability to operate two independent retrieval operations at the same time.

E.2.3.8.2 Potential Environmental Considerations

The first two-level concept could use slightly less land area to store excavated rock because less material would be excavated. The second two-level concept could double the excavation and double the excavated rock volume that would require storage.

Surface soil temperatures could increase due to locating waste closer to the surface and/or increasing thermal loading per acre.

Construction of the full size footprint two-tier repository would require slightly less than double the number of workers and a longer construction period, with associated changes in the potential for industrial accidents. Power consumption would approximately double.

E.3 Enhanced Design Alternatives

Enhanced Design Alternatives are combinations of the alternatives and design features described in preceding sections. These concepts were developed to cover a range of potential repository designs as part of the License Application Design Selection Process described in Section E.1.2. Enhanced Design Alternatives are intended to be improvements to the basic design alternatives discussed in Section E.2. Five Enhanced Design Alternatives are described below, along with the design concepts that led to their development. Potential benefits and environmental considerations are discussed in the sections above dealing with the design alternative and design features incorporated into each Enhanced Design Alternative.

At the time of development of this appendix, the Enhanced Design Alternatives discussed below had been developed, but documentation of the Enhanced Design Alternative development process was forthcoming. That documentation was scheduled to be complete in May 1999. The Enhanced Design Alternatives described in the following sections are preliminary and based on observations of the License Application Design Selection Process and informal discussions with process participants.

E.3.1 ENHANCED DESIGN ALTERNATIVE I

Enhanced Design Alternative I is a low-temperature design intended to remove uncertainties and modeling difficulties associated with above-boiling temperatures. Lower temperatures would mean less disturbance of the subsurface and limit the combined effects of thermal, hydrological, and geochemical processes that are more pronounced in above-boiling-temperature environments.

The goals of Enhanced Design Alternative I are to keep the drift wall temperature below the boiling point of water and the commercial fuel cladding temperature below 350°C (662°F). This would be achieved for the Enhanced Design Alternative I design by limiting areal mass loading to 45 MTHM per acre, increasing the size of the repository to 6 square kilometers (1,500 acres), and using smaller waste packages. Drift spacing would be 43 meters (141 feet) between drift centerlines, with an average end-to-end waste package spacing of 3 meters (10 feet). Preclosure ventilation would use two intake and three exhaust shafts.

The waste package design for this Enhanced Design Alternative would consist of two layers, with Alloy-22 on the outside and 316L stainless steel (nuclear grade) on the inside. Flexible waste package spacing would be used to control the drift temperature. Blending would be used to reduce the maximum thermal output of a waste package to 6.7 kilowatts. To optimize selection of waste for emplacement, additional surface storage capacity above and beyond that in the Viability Assessment reference design would be necessary. A 2-centimeter (0.8-inch)-thick titanium-7 drip shield, to be placed over the waste package just prior to closure, is included in this design to provide defense in depth.

This design allows human access using blast cooling and portable shielding [15 centimeters (6 inches) stainless steel and 7.5 centimeters (3 inches) borated polyethylene].

The major disadvantage of this design is that it uses all of the available space in the upper repository block. Another disadvantage is that it uses smaller waste packages, requiring about 6,000 more waste packages than other Enhanced Design Alternatives.

E.3.2 ENHANCED DESIGN ALTERNATIVE II

Enhanced Design Alternative II is a moderate temperature design intended to keep commercial fuel cladding temperature below 350°C (662°F) and to keep the boiling fronts from merging in the rock walls

between the drifts. Keeping a non-boiling area between the drifts ensures that there would be sufficient area between the drifts that would be below the boiling point to allow water to drain. The areal mass loading could be up to 60 MTHM per acre and still achieve these goals.

The waste package design would consist of two layers with Alloy-22 on the outside and 316L stainless steel on the inside. Blending would be used to reduce the maximum heat output of a waste package to 9 kilowatts. The emplacement area would be 4.3 square kilometers (1,064 acres), and the waste package design would be the same as for Enhanced Design Alternative I. The Enhanced Design Alternative II design would use closely spaced waste packages, line loading, and a drift spacing of 81 meters (266 feet). To optimize selection of waste for emplacement, additional surface storage capacity above and beyond that in the Viability Assessment reference design would be necessary. This design also includes backfill, a 2-centimeter (0.8-inch)-thick titanium-7 drip shield placed just prior to closure, as in Enhanced Design Alternative I. Continuous ventilation would be used for the 50-year preclosure period.

An advantage of this design is that it would reduce or avoid uncertainties associated with the thermal period or thermal pulse where large quantities of water could pool above the repository area. The cooler pillars between the drifts would allow for drainage of waters. However, an uncertainty is that the drainage of water has not been demonstrated. Another advantage is that the design provides flexibility for modification to either a hotter or cooler design.

E.3.3 ENHANCED DESIGN ALTERNATIVE III

Enhanced Design Alternative III is a high thermal load design. The goals are to keep the drift wall temperatures below 200°C (329°F), the commercial fuel cladding temperature below 350°C (662°F), and to ensure that the waste package surface temperature cools to below 80°C (176°F) before the relative humidity at the waste package surface rises above 90 percent. These goals would be met with an 85 MTHM per acre loading, close [0.1 meter, (0.3 foot)] spacing of line-loaded waste packages, and a drift spacing of 56 meters (184 feet).

Two different waste packages are considered (Enhanced Design Alternatives IIIa and IIIb). The Enhanced Design Alternative IIIa waste package would use a two-layer design with 2 centimeters (0.8-inch) of Alloy-22 over 5 centimeters (2 inches) of 316L stainless steel (as in Enhanced Design Alternatives I, II, and V). The Enhanced Design Alternative IIIb waste package design would use a waste package with an outer layer of 2.2 centimeters (0.9 inch) of Alloy-22 over 1.5 centimeter (0.6 inch) of titanium-7 that have been shrink-fitted together, and a 4-centimeter (1.6-inch) inner layer of 316L stainless steel that would fit loosely (gap of 4 millimeters or less) inside the Alloy-22/titanium-7 shell.

Blending would not be used in Enhanced Design Alternative III. However, preclosure ventilation of at least 5 cubic meters (177 cubic feet) per second would be needed for a minimum of 50 years to achieve the temperature goals of this Enhanced Design Alternative. This would require two intake and three exhaust shafts in addition to the access tunnels. Enhanced Design Alternative III also includes a titanium-7 drip shield.

The advantage of Enhanced Design Alternative III is that the surface of the waste package is predicted to cool below 80°C (176°F) before the relative humidity exceeds 90 percent, thus avoiding the worst of the corrosive, warm-moist environment after closure. The disadvantages are the uncertainties connected with temperatures over 100°C (212°F).

E.3.4 ENHANCED DESIGN ALTERNATIVE IV

Enhanced Design Alternative IV is a shielded waste package design located entirely in the upper block with a high thermal load (85 MTHM per acre). The goals of this Enhanced Design Alternative are to keep the gamma radiation dose at the surface of the waste package below 200 millirem per hour, keep the fuel cladding below 350°C (662°F), and keep the emplacement drifts dry for thousands of years.

The waste package would be 30-centimeter (12-inch)-thick A516 steel, and it would have an integral filler that acted as a sponge for oxygen. Waste packages would be line-loaded with a separation of 0.1 meter (0.3 feet). Continuous ventilation at 2 to 5 cubic meters (71 to 177 cubic feet) per second would be required for the 50-year preclosure period. Two intake and three exhaust shafts would be required in addition to the access tunnels. Human access would require blast cooling to reduce temperatures in the drift using a portable 5-centimeter (2-inch)-thick borated polyethylene neutron shielding over the waste packages. Backfill material and drip shields are used in this Enhanced Design Alternative.

The Enhanced Design Alternative IV waste packages would weigh 18,140 metric tons (20 tons) more than those used with other Enhanced Design Alternatives. Since this Enhanced Design Alternative requires a hot postclosure environment to be successful, it would be necessary to manage the waste stream to ensure uniform heat in the repository. Backfill would be placed at closure.

If this design concept does not properly control temperature and relative humidity to protect the drip shield, the carbon steel waste packages would be expected to fail much earlier than the waste packages in the other Enhanced Design Alternatives.

E.3.5 ENHANCED DESIGN ALTERNATIVE V

Enhanced Design Alternative V is a very high thermal load alternative (150 MTHM per acre) and covers the smallest area [168 square kilometers (420 acres)] of the five Enhanced Design Alternatives. The purpose of the very high thermal load is to provide a hot, dry drift environment for thousands of years and avoid extended periods of warm, moist conditions. The goals of this Enhanced Design Alternative were to have drift wall temperatures less than 225°C (437°F) to maintain stability, commercial fuel cladding temperature less than 350°C, and to keep the drift dry for several thousand years.

Waste blending would be required so that waste temperatures were all within 20 percent of the average. Waste packages would be 2-centimeter (0.8-inch) Alloy-22 over 5-centimeter (2-inch) 316L stainless steel, and they would be line loaded with a 0.1-meter (0.3-foot) spacing between waste packages. To optimize selection of waste for emplacement, additional surface storage capacity above and beyond that in the Viability Assessment reference design would be necessary. Drift spacing would be 32.4 meters (106 feet). Preclosure ventilation would reduce air and drift temperatures and remove moisture from the drifts. Four air shafts as well as three access tunnels would be needed. Titanium-7 drip shields would be placed at the time of closure.

The advantage of this design is that it would be located entirely in the lower block of the repository, where the percolation rate is less than half that in the upper block. However, access to the lower block would require a third tunnel. In addition, postclosure conditions could lead to localized corrosion and early failure of waste packages. The high temperatures also could create the possibility that the cladding temperature goal would be exceeded for some waste packages.

REFERENCE

DOE 1998

DOE (U.S. Department of Energy), 1998, *Viability Assessment of a Repository at Yucca Mountain*, DOE/RW-0508, Office of Civilian Radioactive Waste Management, Washington, D.C. [U.S. Government Printing Office, MOL.19981007.0027, Overview; MOL.19981007.0028, Volume 1; MOL.19981007.0029, Volume 2; MOL.19981007.0030, Volume 3; MOL.19981007.0031, Volume 4; MOL.19981007.0032, Volume 5]



Appendix F

Human Health Impacts Primer
and Details for Estimating Health
Impacts to Workers from Yucca
Mountain Repository Operations

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APPENDIX F. HUMAN HEALTH IMPACTS PRIMER AND DETAILS FOR ESTIMATING HEALTH IMPACTS TO WORKERS FROM YUCCA MOUNTAIN REPOSITORY OPERATIONS

Section F.1 of this appendix contains information that supports the estimates of human health and safety impacts in this environmental impact statement (EIS). Specifically, Section F.1 is a primer that explains the natures of radiation and toxic materials, where radiation comes from in the context of the radiological impacts discussed in this EIS, how radiation interacts with the human body to produce health impacts, and how toxic materials interact with the body to produce health impacts. The remainder of the appendix discusses the methodology that was used to estimate worker health impacts and the input data to the analysis, and presents the detailed results of the analysis of worker health impacts.

Section F.2 discusses the methodology and data that the U.S. Department of Energy (DOE) used to estimate worker health and safety impacts for the Proposed Action. It also discusses the detailed results of the impact analysis.

Section F.3 discusses the methodologies and data that DOE used to estimate worker health and safety impacts for Inventory Modules 1 and 2. It also discusses the detailed results of the impact analysis.

Section F.4 discusses the methodology and data that DOE used to estimate worker health and safety impacts for retrieval, should such action become necessary. In addition, it discusses the detailed results from the impact analysis.

Radiological impacts to the public from operations at the Yucca Mountain site could result from release of naturally occurring radon-222 and its decay products in the ventilation exhaust from the subsurface repository operations. The methodology and input data used in the estimates of radiological dose to the public are presented in Appendix G, Air Quality. Outside of the radiation primer, health impacts to the public are not treated in this appendix.

F.1 Human Health Impacts from Exposure to Radioactive and Toxic Materials

This section introduces the concepts of human health impacts as a result of exposure to radiation and potentially toxic materials.

F.1.1 RADIATION AND HUMAN HEALTH

F.1.1.1 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*,

RADIATION

Radiation occurs on Earth in many forms, either naturally or as the result of human activities. Natural forms include light, heat from the sun, and the decay of unstable radioactive elements in the Earth and the environment. Some elements that exist naturally in the human body are radioactive and emit ionizing radiation. They include an isotope of potassium that is an essential element for health and the elements of the uranium and thorium naturally occurring decay series. Human activities have also led to sources of ionizing radiation for various uses, such as diagnostic and therapeutic medicine and nondestructive testing of pipes and welds. Nuclear power generation produces ionizing radiation as well as radioactive materials, which undergo radioactive decay and can continue to emit ionizing radiation for long periods of time.

which is transmitted as photons. Electromagnetic radiation is emitted over a range of wavelengths and energies. We are most commonly aware of visible light, which is part of the spectrum of electromagnetic radiation. Radiation of longer wavelengths and lower energy includes infrared radiation, which heats material when the material and the radiation interact, and radio waves. Electromagnetic radiation of shorter wavelengths and higher energy (which are more penetrating) includes ultraviolet radiation, which causes sunburn, 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 and beta 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.

F.1.1.2 Radioactivity, Ionizing Radiation, Radioactive Decay, and Fission

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. Usually the emitted radiation is ionizing radiation. The result of the process, called *radioactive decay*, is the transformation of an unstable atom (a *radionuclide*) into a different atom, accompanied by the release of energy (as radiation) as the atom 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—but our senses cannot detect them. These types of ionizing radiation can have different characteristics and levels of energy and, thus, varying 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 (shield) the radiation. Alpha particles are the least penetrating and can be stopped by a thin layer of material such as a single sheet of paper. However, if radioactive atoms (called radionuclides) emit alpha particles in the body when they decay, there is a concentrated deposition of energy near the point where the radioactive decay occurs. Shielding for beta particles requires thicker layers of material such as several reams of paper or several inches of wood or water. Shielding from gamma rays, which are highly penetrating, requires very thick material such as several inches to several feet of heavy material (for example, concrete or lead). Deposition of the energy by gamma rays is dispersed across the body in contrast to the local energy deposition by an alpha particle. In fact, some gamma radiation will pass through the body without interacting with it.

FISSION

Fission is the process whereby a large nucleus (for example, uranium-235) absorbs a neutron, becomes unstable, and splits into two fragments, resulting in the release of large amounts of energy per unit of mass. Each fission releases an average of two or three neutrons that can go on to produce fissions in nearby nuclei. If one or more of the released neutrons on the average causes additional fissions, the process keeps repeating. The result is a self-sustaining chain reaction and a condition called criticality. When the energy released in fission is controlled (as in a nuclear reactor), it can be used for various benefits such as to propel submarines or to provide electricity that can light and heat homes.

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. The new lighter atoms are called fission products. The fission products are usually unstable and undergo radioactive decay to reach a more stable state.

Some of the heavy atoms might not fission after absorbing a subatomic particle. Rather, a new nucleus is formed that tends to be unstable (like fission products) and undergo radioactive decay.

The radioactive 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.

F.1.1.3 Exposure to Radiation and Radiation Dose

Radiation that originates outside an individual's body is called *external* or *direct radiation*. Such radiation can come from an X-ray machine or from *radioactive materials* (materials or substances that contain radionuclides), such as radioactive waste or radionuclides in soil. *Internal radiation* originates inside a person's body following intake of radioactive material or radionuclides through ingestion or inhalation. Once in the body, the fate of a radioactive material is determined by its chemical behavior and how it is metabolized. If the material is soluble, it might be dissolved in bodily fluids and be transported to and deposited in various body organs; if it is insoluble, it might move rapidly through the gastrointestinal tract or be deposited in the lungs.

Exposure to ionizing radiation is expressed in terms of *absorbed dose*, which is the amount of energy imparted to matter per unit mass. Often simply called *dose*, it is a fundamental concept in measuring and quantifying the effects of exposure to radiation. The unit of absorbed dose is the *rad*. The different types of radiation mentioned above have different effects in damaging the cells of biological systems. *Dose equivalent* is a concept that considers (1) the absorbed dose and (2) the relative effectiveness of the type of ionizing radiation in damaging biological systems, using a radiation-specific quality factor. The unit of dose equivalent is the *rem*. In quantifying the effects of radiation on humans, other types of concepts are also used. The concept of *effective dose equivalent* is used to quantify effects of radionuclides in the body. It involves estimating the susceptibility of the different tissue in the body to radiation to produce a tissue-specific weighting factor. The weighting factor is based on the susceptibility of that tissue to cancer. The sum of the products of each affected tissue's estimated dose equivalent multiplied by its specific weighting factor is the *effective dose equivalent*. 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 time in the body. The result is called the *committed effective dose equivalent*. The unit of effective dose equivalent is also the *rem*. *Total effective dose equivalent* is the sum of the committed effective dose equivalent from radionuclides in the body plus the dose equivalent from radiation sources external to the body (also in *rem*). All estimates of dose presented in this environmental impact statement, unless specifically noted as something else, are total effective dose equivalents, which are quantified in terms of *rem* or millirem (which is one one-thousandth of a *rem*).

More detailed information on the concepts of radiation dose and dose equivalent are presented in publications of the National Council on Radiation Protection and Measurements (NCRP 1993, page 16-25) and the International Commission on Radiological Protection (ICRP 1991, page 4-11). The DOE implementation guide for occupational exposure assessment (DOE 1998a, pages 3 to 11) also provides additional information.

The factors used to convert estimates of radionuclide intake (by inhalation or ingestion) to dose are called *dose conversion factors*. The National Council on Radiation Protection and Measurements and Federal agencies such as the U.S. Environmental Protection Agency publish these factors (NCRP 1996, all; Eckerman and Ryman 1993, all; Eckerman, Wolbarst, and Richardson 1988, all). They are based on original recommendations of the International Commission on Radiological Protection (ICRP 1977, all).

The radiation dose to an individual or to a group of people can be expressed as the total dose received or as a dose rate, which is dose per unit time (usually an hour or a year).

Collective dose is the total dose to an exposed population. *Person-rem* is the unit of collective dose. Collective dose is calculated by summing the individual dose to each member of a population. For example, if 100 workers each received 0.1 rem, then the collective dose would be 10 person-rem (100×0.1 rem).

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

F.1.1.4 Background Radiation from Natural Sources

Nationwide, on average, members of the public are exposed to approximately 360 millirem per year from natural and manmade sources (Gotchy 1987, page 53). Figure F-1 shows the relative contributions by radiation sources to people living in the United States (Gotchy 1987, page 55).

The estimated average annual dose rate from natural sources is only about 300 millirem per year. This represents about 80 percent of the annual dose received by an average member of the U.S. public. 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 filtered through the atmosphere. With respect to exposures resulting from human activities, medical exposure accounts for 15 percent of the annual dose, and the combined doses from weapons testing fallout, consumer and industrial products, and air travel (cosmic radiation) account for the remaining 3 percent of the total annual dose. Nuclear fuel cycle facilities contribute less than 0.1 percent (0.005 millirem per year per person) of the total dose (Gotchy 1987, pages 53 to 55).

F.1.1.5 Impacts to Human Health from Exposure to Radiation

Chronic Exposure

Cancer is the principal potential risk to human health from exposure to low or chronic levels of radiation. This EIS expresses radiological health impacts as the incremental changes in the number of expected fatal cancers (latent cancer fatalities) for populations and as the incremental increases in lifetime probabilities of contracting a fatal cancer for an individual. The estimates are based on the dose received and on dose-to-health effect conversion factors recommended by the International Commission on Radiological Protection (ICRP 1991, page 22). The Commission estimated that, for the general population, a collective dose of 1 person-rem will yield 0.0005 excess latent cancer fatality. For radiation workers, a collective dose of 1 person-rem will yield an estimated 0.0004 excess latent cancer fatality. The higher risk factor for the general population is primarily due to the inclusion of children in the population group, while the radiation worker population includes only people older than 18. These risk coefficients were adopted by the National Council on Radiation Protection and Measurements in 1993 (NCRP 1993, page 3).

Other health effects such as nonfatal cancers and genetic effects can occur as a result of chronic exposure to radiation. Inclusion of the incidence of nonfatal cancers and severe genetic effects from radiation exposure increases the total change by a factor of 1.5 to 5, compared to the change for latent cancer fatalities (ICRP 1991, page 22). As is the general practice for any DOE EIS, estimates of the total change were not included in the Yucca Mountain EIS.

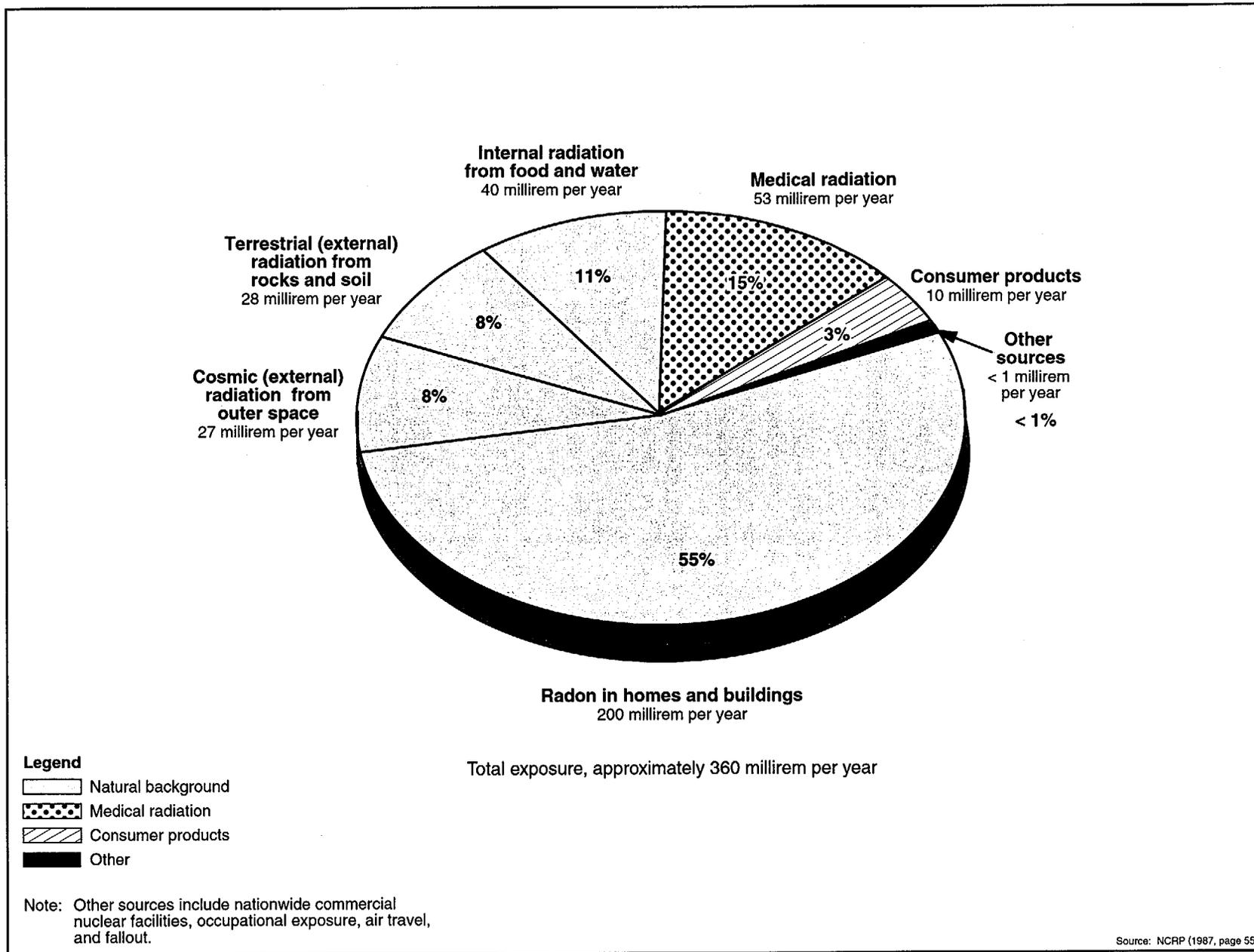


Figure F-1. Sources of radiation exposure.

Acute Exposure

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 be noted at doses 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 received. Information on the effects of acute exposures on humans was obtained from studies of the survivors of the Hiroshima and Nagasaki bombings and from studies following a multitude of acute accidental exposures (Mettler and Upton 1995, pages 276 to 280).

Factors to relate the level of acute exposure to health effects exist but are not applied in this EIS because expected exposures during normal operations for the Proposed Action (including transportation), and for accident scenarios during the Proposed Action and the associated transportation activities, would be well below 50 rem. See Appendix J for exposures from accident scenarios during transportation activities.

F.1.1.6 Exposures from Naturally Occurring Radionuclides in the Subsurface Environment

The estimates of worker doses from inhalation of radon-222 and its decay products while in the subsurface environment and from the ambient radiation fields in the subsurface environment were based on measurements taken in the existing Exploratory Studies Facility drifts. The measurements and the annual dose rates derived from them are discussed below.

Annual Dose Rate for Subsurface Facility Worker from Inhalation of Radon-222

The annual dose rate for a subsurface worker from inhalation of radon-222 and radon decay products was estimated using site-specific measurements of the concentrations of radon-222 and its decay products in the Yucca Mountain Exploratory Studies Facility drifts. Measurements were made at a number of locations in the drifts (TRW 1999a, page 12). After examination of the data from various locations, the measurements taken at the 5,035-meter (about 16,500-foot) station in the main drift, with the ventilation system operating, were determined to provide the best basis for estimating the concentration of radon-222 in the subsurface atmosphere during the various Yucca Mountain Repository phases (TRW 1999a, page 12). The measured concentrations ranged from 0.22 to 72 picocuries per liter, with a median value of 6.5 picocuries per liter.

For each project phase, the measured average value (6.5 picocuries per liter) was adjusted to take into account the difference between the average air residence time in the repository at the time of measurement of radon-222 concentration and the average air residence time for a specific project phase. The average air residence time is the average volume being ventilated divided by the average ventilation rate for a project phase. For example, an increased repository volume would result in an increased average residence time as would a decrease in the ventilation flow rate.

Also considered were (1) the distribution of the measured values of the equilibrium fraction between radon-222 and the decay products in the underground facility; this value ranged from 0.0022 to 0.44, with a median of 0.14 (TRW 1999a, page 12); and (2) the number of hours an involved worker would be underground, exposed to airborne radon. Based on a typical amount of time spent underground (about 6.5 hours per workday) (Jessen 1999, all), the yearly exposure time for involved workers would range from 1,500 to 1,700 hours per year. The dose conversion factor for radon was taken from Publication 65 of the International Commission on Radiological Protection (ICRP 1994, page 24). This dose conversion factor, which is 0.5 rem per working-level month for inhalation of radon decay products by workers, corresponds to 0.029 millirem per picocurie per liter per hour for radon decay products in 100-percent equilibrium (equilibrium factor of 1.0) with the radon-222 parent (ICRP 1994, page 5). For radon

products with a 0.14 equilibrium factor, the dose conversion factor would be 0.0041 millirem per picocurie per liter per hour.

The estimated baseline median dose to an involved worker in the Exploratory Studies Facility from inhalation of radon and radon decay products was estimated to be approximately 60 millirem per year. This estimate was used in calculating the worker dose estimates in this appendix. The estimated 5th-percentile dose is 2 millirem per year, and the 95th-percentile dose is 580 millirem per year. These estimates were made using a Monte Carlo uncertainty analysis.

Annual Dose for Subsurface Facility Worker from Ambient External Radiation in Drifts

Workers in the underground facility would also be exposed to external radiation from naturally occurring primordial radionuclides in the rock. Measured exposure rates for the underground facility ranged from 0.014 to 0.038 millirem per hour (TRW 1999a, page 12). As for inhalation dose estimates, an underground exposure time of 1,500 to 1,700 hours per year was considered. The estimated baseline median dose to an involved worker in the Exploratory Studies Facility from ambient external radiation would be approximately 40 millirem per year. This estimate was used in this appendix for calculating the worker dose estimates from ambient external radiation. The estimated 5th-percentile dose is 23 millirem per year, and the 95th-percentile dose is 56 millirem per year. Like the radon dose estimates, these estimates were made using a Monte Carlo uncertainty analysis.

F.1.2 EXPOSURE TO TOXIC OR HAZARDOUS MATERIALS

When certain natural or manmade materials or substances have harmful effects that are not random or do not occur solely at the site of contact, the materials or substances are described as toxic. Toxicology is the branch of science dealing with the toxic effects that chemicals or other substances might have on living organisms.

Chemicals can be toxic for many reasons, including their ability to cause cancer, to harm or destroy tissue or organs, or to harm body systems such as the reproductive, immune, blood-forming, or nervous systems. The following list provides examples of substances that can be toxic:

- Carcinogens, which are substances known to cause cancer in humans or in animals. If cancers have been observed in animals, they could occur in humans. Examples of generally accepted human carcinogens include asbestos, benzene, and vinyl chloride (Kamrin 1988, pages 37 and 38 and Chapter 6).
- Chemicals that controlled studies have shown to cause a harmful or fatal effect. Examples include metals such as cadmium, lead, and mercury; strong acids such as nitric acid and sulfuric acid; some welding fumes; coal dust; sulfur dioxide; and some solvents.
- Some biological materials, including various body fluids and tissues and infectious agents, are toxic.

Even though chemicals might be toxic, many factors influence whether or not a particular substance has a toxic effect on humans. These factors include (1) the amount of the substance with which the person comes in contact, (2) whether the person inhales or ingests a relatively large amount of the substance in a short time (acute exposure) or repeatedly ingests or inhales a relatively small amount over a longer time (chronic exposure), and (3) the period of time over which the exposure occurs.

Scientists determine a substance's toxic effect (or toxicity) by performing controlled tests on animals. In addition to environmental and physical factors, these tests help establish three other important factors for

measuring toxicity—dose-response relationship, threshold concept, and margin of safety. The dose-response relationship relates the percentage of test animals that experience observable toxic effects to the doses administered. After the administration of an initial dose, the dose is increased or decreased until, at the upper end, all animals are affected and, at the lower end, no animals are affected. Thus, there is a threshold concentration below which there is no effect. The margin of safety is an arbitrary separation between the highest concentration or exposure level that produces no adverse effect in a test animal species and the concentration or exposure level designated safe for humans. There is no universal margin of safety. For some chemicals, a small margin of safety is sufficient; others require a larger margin.

Two substances in the rock at Yucca Mountain, crystalline silica and erionite, are of potential concern as toxic or hazardous materials. Both of these naturally occurring compounds occur in the parent rock at the repository site, and excavation activities could encounter them. The following paragraphs contain additional information on these.

Crystalline Silica

Crystalline silica is a naturally occurring, highly structured form of silica (silicon dioxide, SiO₂). Because it can occur in several different forms, including quartz, cristobalite, and tridymite, it is called a *polymorph*. These three forms occur in the welded tuff parent rock at Yucca Mountain (DOE 1998b, page 25). Crystalline silica is a known causative agent for *silicosis*, a destructive lung condition caused by deposition of particulate matter in the lungs and characterized by scarring of lung tissue. It is contracted by prolonged exposure to high levels of respirable silica dust or an acute exposure to even higher levels of respirable silica dust (EPA 1996, Chapter 8). Accordingly, DOE considers worker inhalation of respirable crystalline silica dust particles to be hazardous to worker health. Current standards for crystalline silica have been established to prevent silicosis in workers.

Cristobalite has a lower exposure limit than does quartz. The limits for these forms of silica include the Permissible Exposure Limits established by the Occupational Safety and Health Administration and the Threshold Limit Value defined by the American Conference of Governmental Industrial Hygienists. The Occupational Safety and Health Administration Permissible Exposure Limit is 50 micrograms per cubic meter averaged over a 10-hour work shift. The American Conference of Governmental Industrial Hygienists Threshold Limit Value is also 50 micrograms per cubic meter, but it is averaged over an 8-hour work shift (NJDHSS 1996, all). Thus, the two limits are essentially the same. In accordance with DOE Order 440.1A (DOE 1998a, page 5), the more restrictive value provided by the American Conference of Governmental Industrial Hygienists will be applied. In addition, the National Institute for Occupational Safety and Health has established Immediately-Dangerous-to-Life-and-Health concentration limits at levels of 50,000 and 25,000 micrograms per cubic meter for quartz and cristobalite, respectively (NIOSH 1996, page 2). These limits are based on the maximum airborne concentrations an individual could tolerate for 30 minutes without suffering symptoms that could impair escape from the contaminated area or irreversible acute health effects.

There is also evidence that silica may be a carcinogen. The International Agency for Research on Cancer has classified crystalline silica and cristobalite as a Class I (known) carcinogen (IARC 1997, pages 205 to 210). The National Institute for Occupational Safety and Health considers crystalline silica to be a potential carcinogen, as defined by the Occupational Safety and Health Administration's carcinogen policy (29 CFR Part 1990). The National Institute for Occupational Safety and Health is reviewing data on carcinogenicity, which could result in a revised limit for crystalline silica. The Environmental Protection Agency has noted an increase in cancer risk to humans who have already developed the adverse noncancer effects of silicosis, but the cancer risk to otherwise healthy individuals is not clear (EPA 1996, pages 1 to 5).

Because there are no specific limits for exposure of members of the public to crystalline silica, this analysis used a comparative benchmark of 10 micrograms per cubic meter, based on a cumulative lifetime exposure limit of 1,000 micrograms per (cubic meter multiplied by years). At this level, an Environmental Protection Agency health assessment has stated that there is a less than 1 percent chance of silicosis (EPA 1996, Chapter 1, page 5, and Chapter 7, page 5). 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 was rounded down to 10 micrograms per cubic meter to establish the benchmark. Appendix G, Section G.1 contains additional information on public exposure to crystalline silica.

Samples of the welded tuff parent rock from four boreholes at Yucca Mountain have an average quartz content of 15.7 percent, an average cristobalite content of 16.3 percent, and an average tridymite content of 3.5 percent (DOE 1998b, page I-1). Worker protection during excavation in the subsurface would be based on the more restrictive Threshold Limit Value for cristobalite. The analysis assumed that the parent rock and dust would have a cristobalite content of 28 percent, which is the higher end of the concentration range reported in TRW (1999b, page 4-81). Thus, the assumed percentage of cristobalite in dust probably will overestimate the airborne cristobalite concentration. Also, studies of both ambient and occupational airborne crystalline silica have shown that most of the airborne crystalline silica is coarse and not respirable (greater than 5 micrometers aerodynamic diameter), and the larger particles will deposit rapidly on the surface (EPA 1996, page 3-26).

Erionite

Erionite is a natural fibrous zeolite that occurs in the rock layers below the proposed repository level in the hollows of rhyolitic tuffs and in basalts. It might also occur in rock layers above the repository level but has not been found in those layers. Erionite is a rare tectosilicate zeolite with hexagonal symmetry that forms wool-like fibrous masses (with a maximum fiber length of about 50 microns, which is generally shorter than asbestos fibers). Erionite particles (ground to powder) resemble amphibole asbestos fibers. Erionite fibers have been detected in samples of road dust in Nevada, and residents of the Intermountain West could be exposed to fibrous erionite in ambient air (Technical Resources 1994, page 134).

There are no specific limits for exposure to erionite. Descriptive studies have shown very high mortality from cancer [malignant mesothelioma, mainly of the pleura (a lung membrane)] in the population of three Turkish villages in Cappadocia where erionite is mined. The International Agency for Research on Cancer has indicated that these studies demonstrate the carcinogenicity of erionite to humans. The Agency classifies erionite as a Group 1 (known) carcinogen (IARC 1987, all).

Erionite could become a potential hazard during excavation of access tunnels to the lower block and to offset Area 5 for the low and intermediate thermal load cases or during vertical boring operations necessary to excavate ventilation shafts. DOE does not expect to encounter erionite layers during the vertical boring operations, which would be through rock layers above known erionite layers, or during excavation of access tunnels to the lower block or offset Area 5, where any identified layers of erionite would likely be avoided (McKenzie 1998, all). In accordance with the Erionite Protocol (DOE 1995, all), a task-specific health and safety plan would be prepared before the start of boring operations to identify this material and prevent worker inhalation exposures from unconfined material.

The Los Alamos National Laboratory is studying the mineralogy and geochemistry of the deposition of erionite under authorization from the DOE Office of Energy Research. Laboratory researchers are applying geochemical modeling so they can understand the factors responsible for the formation of zeolite assemblages in volcanic tuffs. The results of this modeling will be used to predict the distribution of

erionite at Yucca Mountain and to assist in the planning of excavation operations so erionite layers are avoided.

F.1.3 EXPOSURE PATHWAYS

Four conditions must exist for there to be a pathway from the source of released radiological or toxic material to a person or population (Maheras and Thorne 1993, page 1):

- A source term: The material released to the environment, including the amount of radioactivity (if any) or mass of material, the physical form (solid, liquid, gas), particle size distribution, and chemical form
- An environmental transport medium: Air, surface water, groundwater, or a food chain
- An exposure route: The method by which a person can come in contact with the material (for example, external exposure from contaminated ground, immersion in contaminated air or internal exposure from inhalation or ingestion of radioactive or toxic material)
- A human receptor: The person or persons potentially exposed; the level of exposure depends on such factors as location, duration of exposure, time spent outdoors, and dietary intake

These four elements define an exposure pathway. For example, one exposure scenario might involve release of contaminated gas from a stack (source term); transport via the airborne pathway (transport medium); external gamma exposure from the passing cloud (exposure route); and an onsite worker (human receptor). Another exposure scenario might involve a volatile organic compound as the source term, release to groundwater as the transport medium, ingestion of contaminated drinking water as the exposure route, and offsite members of the public as the human receptors. No matter which pathway the scenario involves, local factors such as water sources, agriculture, and weather patterns play roles in determining the importance of the pathway when assessing potential human health effects.

Worker exposure to crystalline silica (and possibly erionite) in the subsurface could occur from a rather unique exposure pathway. Mechanical drift excavation, shaft boring, and broken rock management activities could create airborne dust comprising a range of particles sizes. Dust particles smaller than 10 micrometers have little mass and inertia in comparison to their surface area; therefore, these small particles could remain suspended in dry air for long periods. Airborne dust concentrations could increase if the ventilation system recirculated the air or if airflow velocity in the subsurface facilities became high enough to entrain dust previously deposited on drift or equipment surfaces. As tunnel boring machines or road headers break the rock from the working face, water would be applied to wet both the working face and the broken rock to minimize airborne dust levels. Wet or dry dust scrubbers would capture dust that was not suppressed by the water sprays. To prevent air recirculation, which would lead to an increase of airborne dust loads, the fresh air intake and the exhaust air streams would be separated. Finally, the subsurface ventilation system would be designed and operated to control ambient air velocities to minimize dust reentrainment. If these engineering controls did not maintain dust concentrations below the Threshold Limit Value concentration, workers would have to wear respirators until engineering controls established habitable conditions.

F.2 Human Health and Safety Impact Analysis for the Proposed Action Inventory

This section discusses the methodologies and data used to estimate industrial and radiological health and safety impacts to workers that would result from the construction, operation and monitoring, and closure of the Yucca Mountain Repository, as well as the detailed results from the impact calculations. Section F.2.1 describes the methods used to estimate impacts, Section F.2.2 contains tabulations of the detailed data used in the impact calculations and references to the data sources, and Section F.2.3 contains a detailed tabulation of results.

For members of the public, the EIS uses the analysis methods in Appendix K, Section K.2, to estimate radiation dose from radon-222 and crystalline silica released in the subsurface ventilation system exhaust. The radiation dose estimates were converted to estimates of human health impacts using the dose conversion factors discussed in Section F.1.1.5. These impacts are expressed as the probability of a latent cancer fatality for a maximally exposed individual and as the number of latent cancer fatalities among members of the public within about 80 kilometers (50 miles) for the Proposed Action, the retrieval contingency, and the inventory modules. The results are listed in Chapter 4, Section 4.1.7.

Health and safety impacts to workers have been estimated for two worker groups: involved workers and noninvolved workers. Involved workers are craft and operations personnel who would be directly involved in activities related to facility construction and operations, including excavation activities; receipt, handling, packaging, and emplacement of spent nuclear fuel and high-level radioactive waste material; monitoring of conditions and performance of the waste packages; and those directly involved in closure activities. Noninvolved workers are managerial, technical, supervisory, and administrative personnel who would not be directly involved in construction, excavation, operations, monitoring, and closure activities. The analysis did not consider project workers who would not be located at the repository site.

F.2.1 METHODOLOGY FOR CALCULATING OCCUPATIONAL HEALTH AND SAFETY IMPACTS

To estimate the impacts to workers from industrial hazards common to the workplace, values for the full-time equivalent work years for each phase of the project were multiplied by the statistic (occurrence per 10,000 full-time equivalent work years) for the impact being considered. Values for the number of full-time equivalent workers for each phase of the project are listed in Section F.2.2.1. The statistics for industrial impacts for each of the phases are listed in Section F.2.2.2 for involved and noninvolved workers.

Two kinds of radiological health impacts to workers are provided in this EIS. The first is an estimate of the latent cancer fatalities to the worker group involved in a particular project phase. The second is the incremental increase in latent cancer fatalities attributable to occupational radiation for a maximally exposed individual in the worker population for each project phase.

To calculate the expected number of worker latent cancer fatalities during a phase of the project, the collective dose to the worker group, in person-rem, was multiplied by a standard factor for converting the collective worker dose to projected latent cancer fatalities (see Section F.1.1.5). As discussed in Section F.1.1.5, the value of this factor for radiation workers is 0.0004 excess latent cancer fatality per person-rem of dose.

The collective dose for a particular phase of the operation is calculated as the product of the number of full-time equivalent workers for the project phase (see Section F.2.2.1), the average dose over the exposure period, and the fraction of the working time that a worker is in an environment where there is a

source of radiation exposure. Values for exposure rates for both involved and noninvolved workers are presented in Section F.2.2.3 as are the fractional occupancy factors. The calculation of collective dose to subsurface workers from exposure to the radiation emanating from the loaded waste packages is an exception. Collective worker doses from this source of exposure were calculated using the methodology described in TRW (1999b, Tables G-1 and G-2). For the calculation of exposures, the estimated annual radiation doses listed in TRW (1999b, Tables G-3, G-3a, G-4, and G-4a) for the various classes of involved subsurface workers were used. The exposure values were multiplied by the craft manpower distribution listed in TRW (1999b, Tables G-5, G-5a, G-5b, G-7, G-7a, and G-7b) for each of the involved labor classes for a project phase to obtain an overall annual exposure. The annual exposures for the labor classes were then summed to obtain the collective annual dose in person-rem to the involved subsurface workers for each of the subsurface operational phases. The total collective dose was then obtained by multiplying the annual collective dose by the length of the project phase.

To estimate the incremental increase in the likelihood of death from a latent cancer for the maximally exposed individual, the estimated dose to the maximally exposed worker was multiplied by the factor for converting radiation dose to latent cancers. The factor applied for workers was 0.0004 latent cancer fatality per rem, as discussed above and in Section F.1.1.5. Thus, if a person were to receive a dose of 1 rem, the incremental increase in the probability that person would suffer a latent cancer fatality is 1 in 2,500 or 0.0004.

To estimate the dose for a hypothetical maximally exposed individual, the analysis generally assumed that this individual would be exposed to the radiation fields (see Section F.2.2.3) over the entire duration of a project phase or for 50 years, whichever would be shorter. Other sources of exposure while working underground would be ambient radiation coming from the radionuclides in the drift walls and from inhalation of radon-222 and its decay products. The radiation from the waste package is usually the dominant component when these three dose contributors are added. Doses for the maximally exposed subsurface worker were estimated by adding the three dose components because they would occur simultaneously.

F.2.2 DATA SOURCES AND TABULATIONS

F.2.2.1 Work Hours for the Repository Phases

Table F-1 lists the number of workers involved in the various repository phases in terms of full-time equivalent work years. Each full-time equivalent work year represents 2,000 work hours (the number of hours assumed for a normal work year). The values were obtained from TRW (1999c, Section 6) and from TRW (1999b, Section 6) for surface and subsurface workers, respectively.

F.2.2.2 Workplace Health and Safety Statistics

The analysis selected health and safety statistics for three impact categories—total recordable cases, lost workday cases, and fatalities. Total recordable cases are occupational injuries or illnesses that result in:

- Fatalities, regardless of the time between the injury and death, or the length of the illness
- Lost workday cases, other than fatalities, that result in lost workdays
- Nonfatal cases without lost workdays that result in transfer to another job, termination of employment, medical treatment (other than first aid), loss of consciousness, or restriction of work or motion
- Diagnosed occupational illness cases that are reported to the employer but are not classified as fatalities or lost workday cases

Table F-1. Estimated full-time equivalent worker years for repository phases.

Phase	Subphase or worker group	Source ^a	Length of phase	High thermal load			Intermediate thermal load			Low thermal load		
				UC ^b	DISP ^c	DPC ^d	UC	DISP	DPC	UC	DISP	DPC
<i>Construction</i>	Surface	(1)	44 months									
	Involved			2,380	1,650	1,760	2,380	1,650	1,760	2,380	1,650	1,760
	Noninvolved			900	630	670	900	630	670	900	630	670
	Subsurface	(2)	5 years									
	Involved				2,300	2,300	2,300	2,460	2,460	2,460	2,460	2,460
	Noninvolved			600	600	600	600	600	600	600	600	600
	<i>Construction subtotal</i>			<i>6,180</i>	<i>5,180</i>	<i>5,330</i>	<i>6,340</i>	<i>5,340</i>	<i>5,490</i>	<i>6,340</i>	<i>5,340</i>	<i>5,490</i>
<i>Operation and monitoring</i>	Operations	(3)	24 years									
	Surface handling											
	Involved			17,500	11,470	11,810	17,500	11,470	11,810	17,500	11,470	11,810
	Noninvolved			13,150	11,620	11,760	13,150	11,620	11,760	13,150	11,620	11,760
	Subsurface emplacement	(4)	24 years									
	Involved			1,780	1,780	1,780	1,780	1,780	1,780	1,780	1,780	1,780
	Noninvolved			380	380	380	380	380	380	380	380	380
	Subsurface development	(5)	(e)									
	Involved			6,230	6,230	6,230	6,230	6,230	6,230	6,530	6,530	6,530
	Noninvolved		22 years	1,670	1,670	1,670	1,670	1,670	1,670	1,670	1,670	1,670
	<i>Operations subtotal</i>			<i>40,710</i>	<i>33,150</i>	<i>33,630</i>	<i>40,710</i>	<i>33,150</i>	<i>33,630</i>	<i>41,010</i>	<i>33,450</i>	<i>33,930</i>
	Monitoring	(6)	76 years									
	Surface											
Involved				2,260	2,260	2,260	2,260	2,260	2,260	2,260	2,260	2,260
Noninvolved				NA ^f	NA	NA	NA	NA	NA	NA	NA	NA
Surface decontamination	(7)	3 years										
Involved				4,060	2,950	3,070	4,060	2,950	3,070	4,060	2,950	3,070
Noninvolved				NA	NA	NA	NA	NA	NA	NA	NA	NA
Subsurface	(8)	76 years										
Involved				5,240	5,240	5,240	5,240	5,240	5,240	5,780	5,780	5,780
Noninvolved				990	990	990	990	990	990	990	990	990
<i>Monitoring subtotal</i>				<i>12,550</i>	<i>11,440</i>	<i>11,560</i>	<i>12,550</i>	<i>11,440</i>	<i>11,560</i>	<i>13,090</i>	<i>11,980</i>	<i>12,100</i>
<i>Operation and monitoring subtotal</i>				<i>53,260</i>	<i>44,590</i>	<i>45,190</i>	<i>53,260</i>	<i>44,590</i>	<i>45,190</i>	<i>54,500</i>	<i>45,430</i>	<i>46,030</i>
<i>Closure</i>	Surface	(9)	6 years									
	Involved			1,580	1,110	1,200	1,580	1,110	1,210	1,580	1,110	1,200
	Noninvolved			600	420	460	600	420	460	600	420	460
	Subsurface	(10)	(g)									
	Involved				1,310	1,310	1,310	1,310	1,310	1,310	3,270	3,270
	Noninvolved			260	260	260	260	260	260	660	660	660
<i>Closure subtotal</i>				<i>3,750</i>	<i>3,100</i>	<i>3,230</i>	<i>3,750</i>	<i>3,100</i>	<i>3,230</i>	<i>6,110</i>	<i>5,460</i>	<i>5,590</i>
Totals				63,190	52,870	53,750	63,350	53,030	53,910	66,940	56,230	57,110

a. Sources: (1) TRW (1999c, Table 6-1); (2) TRW (1999b, Table 6.1.1.1-1); (3) TRW (1999c Table 6-2); (4) TRW (1999b, Table 6.1.3.1-1); (5) TRW (1999b, Table 6.1.2.1-1); (6) TRW (1999c, Table 6-5); (7) TRW (1999c, Table 6-4); (8) TRW (1999b, Table 6.1.4.1-1); (9) TRW (1999c, Table 6-6); (10) TRW (1999b, Table 6.1.6.1-1).

b. UC = uncanistered packaging scenario.

c. DISP = disposable canister packaging scenario.

d. DPC = dual-purpose canister packaging scenario.

e. High thermal load and intermediate thermal load = 21 years; low thermal load = 22 years.

f. NA = not applicable.

g. High thermal load = 6 years; intermediate thermal load = 6 years; low thermal load = 15 years.

Lost workday cases, which are described above, include cases that result in the loss of more than half a workday. These statistical categories, which have been standardized by the U.S. Department of Labor and the Bureau of Labor Statistics, must be reported annually by employers with 11 or more employees. Table F-2 summarizes the health and safety impact statistics used for this analysis.

Table F-2. Health and safety statistics for estimating industrial safety impacts common to the workplace.^a

Phase	Total recordable cases incidents per 100 FTEs ^b		Lost workday cases per 100 FTEs		Fatalities per 100,000 FTEs (involved and noninvolved) ^c	Data set for TRCs and LWCs ^d
	Involved	Noninvolved	Involved	Noninvolved		
<i>Construction</i>						
Surface	6.1	3.3	2.9	1.6	2.9	(1)
Subsurface	6.1	3.3	2.9	1.6	2.9	(1)
<i>Operation and Monitoring</i>						
<i>Operation period</i>						
Surface	3	3.3	1.2	1.6	2.9	(3)
Subsurface - emplacement	3	3.3	1.2	1.6	2.9	(3)
Subsurface - drift development	6.8	1.1	4.8	0.7	2.9	(2)
<i>Monitoring period</i>						
Surface	3	3.3	1.2	1.6	2.9	(3)
Subsurface	3	3.3	1.2	1.6	2.9	(3)
<i>Closure</i>						
Surface	6.1	3.3	2.9	1.6	2.9	(1)
Subsurface	6.1	3.3	2.9	1.6	2.9	(1)

a. See text below for source of data in Data Sets 1, 2, and 3.

b. FTEs = full-time equivalent work years.

c. See the discussion about Data Set 4 for source of fatality statistic for normal industrial activities.

d. TRCs = total recordable cases; LWCs = lost workday cases.

Table F-2 cites three sets of statistics that were used to estimate total recordable cases and lost workday cases for workers during activities at the Yucca Mountain site. In addition, there is a fourth statistic related to the occupational fatality projections for the Yucca Mountain site activities. The source of information from which the sets of impact statistics were derived is discussed below. All of the statistics are based on DOE experience for similar types of activities and were derived from the DOE CAIRS (Computerized Accident/Incident Reporting and Recordkeeping System) data base (DOE 1999, all).

Data Set 1, Construction and Construction-Like Activities

This set of statistics from the DOE CAIRS data base was applied to construction or construction-like activities. Specifically, it was used for both surface and subsurface workers during the construction phase and the closure phase (closure phase activities were deemed to be construction-like activities). The statistics were based on a 6.75-year period (1992 through the third quarter of 1998).

For involved workers the impact statistic numbers were derived from the totals for all of the DOE construction activities over the period. For noninvolved workers, the values were derived from the combined government and services contractor noninvolved groups for the same period. The noninvolved worker statistic, then, is representative of impacts for oversight personnel who would not be involved in

the actual operation of equipment or resources. The basic statistics derived from the CAIRS data base for each of the groups include:

- Involved worker total recordable cases: 764 recordable cases for approximately 12,400 full-time equivalent work years
- Involved worker lost workday cases: 367 lost workday cases for approximately 12,400 full-time equivalent work years
- Noninvolved worker total recordable cases: 1,333 recordable cases for approximately 40,600 full-time equivalent work years
- Noninvolved worker lost workday cases: 657 lost workday cases for approximately 40,600 full-time equivalent work years

Data Set 2, Excavation Activities

This set of statistics was derived from experience at the Yucca Mountain Project over a 30-month period (fourth quarter of 1994 through the first quarter of 1997). DOE selected this period because it coincided with the exploratory tunnel boring machine operations at Yucca Mountain, reflecting a high level of worker activity during ongoing excavation activities. This statistic was applied for the Yucca Mountain Project subsurface development period, which principally involves drift development activities. The Yucca Mountain Project experience from which the statistic is derived is presented in Table F-3. Stewart (1998, all) contains the Yucca Mountain statistics, which were derived from the CAIRS data base (DOE 1999, all).

Table F-3. Yucca Mountain Project worker industrial safety loss experience.^a

Factor	Value ^b	Basis
<i>TRCs^c per 100 FTEs^d</i>		
Involved worker	6.8	56 TRCs for 825 construction FTEs
Noninvolved worker	1.1	2.3 TRCs for 2,015 nonconstruction FTEs
<i>LWCs^e per 100 FTEs</i>		
Involved worker	4.8	40 LWCs for 825 construction FTEs
Noninvolved worker	0.7	14 LWCs for 2,015 nonconstruction FTEs
<i>Fatality rate occurrence per 100,000 FTEs</i>		
Involved worker	0.0	No fatalities for 825 construction FTEs
Noninvolved worker	0.0	No fatalities for 2,015 nonconstruction FTEs

- a. Fourth quarter 1994 through first quarter 1997.
- b. Source: Adapted from the CAIRS data base (DOE 1999, all) by Stewart (1998, all) for the fourth quarter of 1994 through the first quarter of 1997.
- c. TRCs = total recordable cases of injury and illness.
- d. FTEs = full-time equivalent work years.
- e. LWCs = lost workday cases.

Data Set 3, Activities Involving Work in a Radiological Environment

This set of statistics is from the DOE CAIRS data base (DOE 1999, all). In arriving at the statistics listed in Table F-2, information from the Savannah River Site, the Hanford Site, and the Idaho National Engineering and Environmental Laboratory was averaged individually for the 6.5 years from 1992 through the second quarter of 1998. The averages were then combined to produce an overall average. The reason these three sites were selected as the basis for this set of statistics is that the DOE Savannah River, Hanford, and Idaho National Engineering and Environmental Laboratory sites currently conduct most of the operations in the DOE complex involving handling, sorting, storing, and inspecting spent

nuclear fuel and high-level radioactive waste materials, as well as similar activities for low-level radioactive waste materials. The Yucca Mountain Repository phases for which this set of statistics was applied included the receipt, handling, and packaging of spent nuclear fuel and high-level radioactive waste in the surface facilities; subsurface emplacement activities; and surface and subsurface monitoring activities, including decontamination of the surface facilities. These activities involve handling, storing, and inspecting spent nuclear fuel and high-level radioactive waste, so the worker activities at the Yucca Mountain site are expected to be similar to those cited above for the other sites in the DOE complex.

The basic statistics for the involved and noninvolved workers include:

- Involved worker total recordable cases: 1,246 for about 41,600 full-time equivalent work years
- Involved worker lost workday cases: 538 for about 41,600 full-time equivalent work years
- Noninvolved worker total recordable cases: 1,333 for about 40,600 full-time equivalent work years
- Noninvolved worker lost workday cases: 657 for about 40,600 full-time equivalent work years

Data Set 4, Statistics for Worker Fatalities from Industrial Hazards

There have been no reported fatalities as a result of workplace activities for the Yucca Mountain project. Similarly, there are no fatalities listed in the Mine Safety and Health Administration data base for stone mining workers (MSHA 1999, all). Because fatalities in industrial operations sometimes occur, the more extensive overall DOE data base was used to estimate a fatality rate for the activities at the Yucca Mountain site. Statistics for the DOE facility complex for the 10 years between 1988 and 1997 were used (DOE 1999, all). These fatality statistics are for both government and contractor personnel working in the DOE complex who were involved in the operation of equipment and resources (involved workers). The activities in the DOE complex covered by this statistic were governed by safety and administrative controls (under the DOE Order System) that are similar to the safety and administrative controls that would be applied for Yucca Mountain Repository work. These fatality statistics were also applied to the noninvolved worker population because they are the most inclusive statistics in the CAIRS data base. However, the statistics probably are conservatively high for the noninvolved worker group.

F.2.2.3 Estimates of Radiological Exposures

DOE considered the following potential sources of radiation exposure for assessing radiological health impacts to workers:

- Inhalation of gaseous radon-222 and its decay products. Subsurface workers could inhale the radon-222 present in the air in the repository drifts. Workers on the surface could inhale radon-222 released to the environment in the exhaust air from the subsurface ventilation system.
- External exposure of surface workers to radioactive gaseous fission products that could be released during handling and packaging of spent nuclear fuel with failed cladding for emplacement in the repository. Such impacts would be of most concern for the uncanistered shipping cask scenario.
- Direct external exposure of workers in the repository drifts as a result of naturally occurring radionuclides in the walls of the drifts (primarily potassium-40 and radionuclides of the naturally occurring uranium and thorium decay series).
- External exposure of workers to direct radiation emanating from the waste packages containing spent nuclear fuel and high-level radioactive waste either during handling and packaging (surface facility workers) or after it is placed within the waste package (largely subsurface workers).

Section F.1.1.6 describes the approach taken to estimate exposures to workers as a result of release of gaseous radon-222 from the drift walls to the subsurface atmosphere. For radon exposures to subsurface workers, the analysis assumed a subsurface occupancy factor of 1.0 for involved workers, an occupancy factor of 0.6 for noninvolved workers for construction and drift development activities, and an occupancy factor of 0.4 for noninvolved workers for emplacement, monitoring, and closure (Rasmussen 1998a, all; Rasmussen 1999, all; Jessen 1999, all).

As discussed in Section F.1.1.6, the average concentration of radon-222 in the subsurface atmosphere varies with the ventilation rate and repository volume. Table F-4 lists the correction factors (multipliers) applied to the average value for the concentration of radon-222 measured in the Exploratory Studies Facility for the Proposed Action.

Table F-4. Correction factors and annual exposures from radon-222 and its decay products for each of the project phases or periods under the Proposed Action.^a

Project phase or period	Correction factor			Annual dose rate (millirem per year)		
	Thermal load scenario			Thermal load scenario		
	High	Intermediate	Low	High	Intermediate	Low
Construction	1.9	2.2	2.2	114	132	132
Drift development	0.6	0.6	0.6	36	36	36
Emplacement	1.1	1.5	2.9	66	90	174
Monitoring	3.2	4.1	4.4	192	246	264
Closure	3.2	4.1	4.4	192	246	264
Retrieval ^b	3.2	3.2	3.2	192	192	192

- a. Based on the measured value of 60 rem per year corrected for repository volume and ventilation rate; see Section F.1.1.6 and Appendix G (Section G.2.3.1).
 b. Multiplier for retrieval is not dependent on thermal load.

Appendix G, Section G.2.4.2 describes the approach taken to estimate source terms and associated doses to workers from the potential release of gaseous fission products from spent nuclear fuel with failed cladding.

Subsurface workers would also be exposed to background gamma radiation from naturally occurring radionuclides in the subsurface rock (largely from the uranium-238 decay series radionuclides and from potassium-40, both in the rock). DOE has based its projection of worker external gamma dose rates on the data obtained during Exploratory Studies Facility operations (Section F.1.1.6). The collective ambient radiation exposures for subsurface workers were calculated assuming occupancy factors cited in the previous paragraph for subsurface workers for emplacement and monitoring activities (Rasmussen 1998a, all; Rasmussen 1999, all; Jessen 1999, all).

Table F-5 lists dose rates in the fourth column for cases in which the annual full-time equivalent surface worker exposure values vary with the shipping package scenario. The table also lists the sources from which the data were obtained. The dose rates to subsurface workers from the radiation emitted from waste packages would vary with the thermal load, as indicated in the fourth column of Table F-5.

Table F-6 lists the annual exposures to subsurface workers from radiation emanating from the waste packages for the high, intermediate, and low thermal load scenarios, under the Proposed Action and Module 1 and 2 inventories. Section F.3 discusses Inventory Modules 1 and 2.

Table F-5. Radiological exposure data used to calculate worker radiological health impacts (page 1 of 2).

Phase and worker group	Exposure source ^a	Occupancy factor ^b	Annual dose (millirem, except where noted)	Annual full-time equivalent workers ^c			Data source ^e
				UC ^d	DISP ^e	DPC ^f	
<i>Construction</i>							
<i>Surface</i>							
Involved	Radon-222 inhalation	1.0	Small relative to subsurface worker exposures				(h)
Noninvolved	Radon-222 inhalation	1.0	Small relative to subsurface worker exposures				(h)
<i>Subsurface</i>							
Involved	Drift ambient	1.0	40				(1), (2)
	Radon-222 inhalation	1.0	Table F-4				(2), Table F-4
Noninvolved	Drift ambient	0.6	40				(1), (2)
	Radon-222 inhalation	0.6	Table F-4				(2), Table F-4
<i>Operations and monitoring</i>							
<i>Surface handling and loading operations</i>							
Involved	Receipt, handling and packaging of spent nuclear fuel and high-level radioactive waste	1.0	400 100	464 297	199 228	199 244	(3)
Noninvolved	Receipt, handling and packaging of spent nuclear fuel and high-level radioactive waste	1.0	25 0	175 341	150 386	149 390	(3)
<i>Surface monitoring</i>							
Involved only	Radon-222 inhalation	1.0	Small relative to subsurface workers				(i)
<i>Surface decontamination (postemplacement, involved only)</i>							
	External exposure	1.0	100	826	599	624	(4)
		1.0	25	528	383	399	(4)
<i>Subsurface emplacement</i>							
Involved	Waste package	Varies, see Table F-6	Varies, see Table F-6				Table F-6
	Drift ambient	1.0	40				(1), (2)
	Radon-222	1.0	Table F-4				(2), Table F-4
Noninvolved	Waste package	0.04	0.1 millirem per hour				(5)
	Drift ambient	0.4	40				(1), (2)
	Radon-222 inhalation	0.4	Table F-4				(2), Table F-4
<i>Subsurface drift development</i>							
Involved	Drift ambient	1.0	40				(1), (2)
	Radon-222 inhalation	1.0	Table F-4				(2), Table F-4
Noninvolved	Drift ambient	0.6	40				(1), (2)
	Radon-222 inhalation	0.6	Table F-4				(2), Table F-4
<i>Monitoring</i>							
<i>Subsurface</i>							
Involved	Waste package	Varies, see Table F-6	Varies, see Table F-6				Table F-6
	Drift ambient	1.0	40				(1), (2)
	Radon-222 inhalation	1.0	Table F-4				(2), Table F-4
Noninvolved	Waste package	0.04	0.1 millirem per hour				(5)
	Drift ambient	0.4	40				(1), (2), (6)
	Radon-222 inhalation	0.4	Table F-4				(2), (6), Table F-4

Table F-5. Radiological exposure data used to calculate worker radiological health impacts (page 2 of 2).

Phase and worker group	Exposure source ^a	Occupancy factor ^b	Annual dose (millirem per year except where noted)	Annual full-time equivalent workers ^c			Data source ^g
				UC ^d	DISP ^e	DPC ^f	
<i>Closure</i>							
<i>Surface</i>							
Involved		1.0	Small relative to subsurface worker exposures				(j)
Noninvolved		1.0	Small relative to subsurface worker exposures				(j)
<i>Subsurface</i>							
Involved	Waste package	Varies, see Table F-6	Varies, see Table F-6				Table F-6
	Drift ambient	1.0	40				(1), (2)
Noninvolved	Radon-222 inhalation	1.0	Table F-4				(2), Table F-4
	Waste package	0.04	0.1 millirem per hour				(5)
	Drift ambient	0.4	40				(1), (2)
	Radon-22 inhalation	0.4	Table F-4				(2), Table F-4

a. Exposure sources include radiation from spent nuclear fuel and high-level radioactive waste packages to surface and subsurface workers, the ambient exposure to subsurface workers from naturally occurring radiation in the drift walls, and internal exposures from inhalation of radon-222 and its decay products in the drift atmosphere.

b. Fraction of 8-hour workday that workers are exposed.

c. Number of annual full-time equivalent workers for surface facility activities when number of workers would vary with shipping package scenario.

d. UC = uncanistered packaging scenario.

e. DISP = disposable canister packaging scenario.

f. DPC = dual-purpose canister packaging scenario.

g. Sources:

(1) Section F.1.1.6.

(2) Rasmussen (1998a, all).

(3) TRW (1999c, Table 6-2).

(4) Total employment for decontamination activities taken from TRW (1999c, Table 6-4). In Table 6-2 of TRW (1999c), the distribution of involved workers for surface facility receipt, handling, and packaging phase between the 400 millirem per year and 100 millirem per year cases is 61 percent and 39 percent, respectively. For decontamination operations it was assumed that 69 percent of the involved worker population would receive 100 millirem per year and 39 percent of the involved worker population would receive 25 millirem per year.

(5) Rasmussen (1999, all).

(6) Jessen (1999, all).

h. Comparison of information in Chapter 4, Table 4-2 (surface workers) and Table F-9 (subsurface workers).

i. Comparison of information in Chapter 4, Table 4-5 (surface workers) and Table F-27 (subsurface workers).

j. Comparison of information in Chapter 4, Table 4-7 (surface workers) and Table F-30 (subsurface workers).

Table F-6. Annual involved subsurface worker exposure rates from waste packages^a (person-rem per year).

Project phase	Proposed Action			Inventory Modules		
	High	Intermediate	Low	High	Intermediate	Low
Emplacement	10.1	10.2	5.6	10.2	10.2	6.0
Monitoring	7.2	7.2	4.1	7.2	7.8	5.6
Closure	12.5	12.5	7.4	12.5	12.5	7.4

a. Sources: individual exposure values from TRW (1999b, Appendix G, Tables G-3, G-3a, G-4, and G-4a).

b. Calculated annual exposures, Rasmussen (1999, all).

F.2.3 COMPILATION OF DETAILED RESULTS FOR OCCUPATIONAL HEALTH AND SAFETY IMPACTS

F.2.3.1 Occupational Health and Safety Impacts During the Construction Phase

F.2.3.1.1 Industrial Hazards to Workers

Tables F-7 and F-8 list health and safety impacts from industrial hazards to surface and subsurface workers, respectively, for construction activities.

Table F-7. Industrial hazard health and safety impacts to surface facility workers during construction phase (44 months).^a

Worker group	Waste packaging scenario		
	Uncanistered	Disposable canister	Dual-purpose canister
<i>Involved</i>			
Full-time equivalent work years ^b	2,380	1,650	1,760
Total recordable cases	150	100	110
Lost workday cases	70	50	50
Fatalities	0.07	0.05	0.05
<i>Noninvolved</i>			
Full-time equivalent work years	900	630	670
Total recordable cases	30	21	22
Lost workday cases	15	10	11
Fatalities	0.03	0.02	0.02
<i>All workers (totals)^c</i>			
Full-time equivalent work years	3,280	2,280	2,420
Total recordable cases	180	120	130
Lost workday cases	85	59	63
Fatalities	0.10	0.07	0.07

a. Source: Impact rates from Table F-2.

b. Source: Table F-1.

c. Totals might differ from sums due to rounding.

Table F-8. Industrial hazard health and safety impacts to subsurface facility workers during construction phase (5 years).^a

Worker group	Thermal load scenario		
	High	Intermediate	Low
<i>Involved</i>			
Full-time equivalent work years ^b	2,300	2,460	2,460
Total recordable cases	140	150	150
Lost workday cases	68	72	72
Fatalities	0.07	0.07	0.07
<i>Noninvolved</i>			
Full-time equivalent work years	600	600	600
Total recordable cases	20	20	20
Lost workday cases	10	10	10
Fatalities	0.02	0.02	0.02
<i>All workers (totals)^c</i>			
Full-time equivalent work years	2,900	3,060	3,060
Total recordable cases	160	170	170
Lost workday cases	77	82	82
Fatalities	0.08	0.09	0.09

a. Source: Impact rates from Table F-2.

b. Source: Table F-1.

c. Totals might differ from sums due to rounding.

F.2.3.1.2 Radiological Health Impacts to Workers

Tables F-9 and F-10 list subsurface worker health impacts from inhalation of radon-222 in the subsurface atmosphere and from ambient radiation exposure from radionuclides in the rock of the drift walls, respectively. The radiological health impacts to surface workers from inhalation of radon-222 would be small in comparison to those for subsurface workers; therefore, they were not tabulated in this appendix (see Table F-5, Footnote h, for sources of exposure).

Table F-9. Radiological health impacts to subsurface facility workers from radon exposure during construction phase.^a

Worker group	Thermal load scenario		
	High	Intermediate	Low
<i>Involved</i>			
Full-time equivalent work years ^b	2,300	2,460	2,460
Maximally exposed individual (MEI) worker dose (millirem)	570	660	660
Latent cancer fatality probability for MEI	0.0002	0.0003	0.0003
Collective dose (person-rem)	260	320	320
Latent cancer fatality incidence	0.10	0.13	0.13
<i>Noninvolved</i>			
Full-time equivalent work years	600	600	600
Maximally exposed individual (MEI) worker dose (millirem)	430	500	500
Latent cancer fatality probability for MEI	0.0002	0.0002	0.0002
Collective dose (person-rem)	52	60	60
Latent cancer fatality incidence	0.02	0.02	0.02
<i>All workers (totals)^f</i>			
Full-time equivalent work years	2,900	3,060	3,060
Collective dose (person-rem)	310	380	380
Latent cancer fatality incidence	0.12	0.15	0.15

- a. Source: Exposure data from Table F-5.
 b. Source: Table F-1.
 c. Totals might differ from sums due to rounding.

Table F-10. Radiological health impacts to subsurface facility workers from ambient radiation exposure during construction phase.^a

Worker group	Thermal load scenario		
	High	Intermediate	Low
<i>Involved</i>			
Full-time equivalent work years ^b	2,300	2,460	2,460
Maximally exposed individual (MEI) worker dose (millirem)	200	200	200
Latent cancer fatality probability for MEI	0.00008	0.00008	0.00008
Collective dose (person-rem)	92	98	98
Latent cancer fatality incidence	0.04	0.04	0.04
<i>Noninvolved</i>			
Full-time equivalent work years	600	600	600
Maximally exposed individual (MEI) worker dose (millirem)	150	150	150
Latent cancer fatality probability for MEI	0.00006	0.00006	0.00006
Collective dose (person-rem)	18	18	18
Latent cancer fatality incidence	0.007	0.007	0.007
<i>All workers (totals)^c</i>			
Full-time equivalent work years	2,900	3,060	3,060
Collective dose (person-rem)	110	120	120
Latent cancer fatality incidence	0.04	0.05	0.05

- a. Source: Exposure data from Table F-5.
 b. Source: Table F-1.
 c. Totals might differ from sums due to rounding.

F.2.3.2 Occupational Health and Safety Impacts During the Operations Period

F.2.3.2.1 Industrial Safety Hazards to Workers

Tables F-11, F-12, and F-13 list estimated impacts for each worker group during waste receipt and packaging, drift development, and emplacement activities during the operations period.

Table F-11. Industrial hazard health and safety impacts to surface facility workers during waste receipt and packaging period (24 years).^a

Worker group	Waste packaging option		
	Uncanistered	Disposable canister	Dual-purpose canister
<i>Involved</i>			
Full-time equivalent work years ^b	17,500	11,470	11,810
Total recordable cases of injury and illness	520	340	350
Lost workday cases	210	140	140
Fatalities	0.51	0.33	0.34
<i>Noninvolved</i>			
Full-time equivalent work years	13,150	11,620	11,760
Total recordable cases of injury and illness	430	380	390
Lost workday cases	210	190	190
Fatalities	0.38	0.34	0.34
<i>All workers (totals)^c</i>			
Full-time equivalent work years	30,650	23,090	23,570
Total recordable cases of injury and illness	960	730	740
Lost workday cases	440	340	340
Fatalities	0.89	0.67	0.68

a. Source: Impact rates from Table F-2.

b. Source: Table F-1.

c. Totals might differ from sums due to rounding.

Table F-12. Industrial hazard health and safety impacts to subsurface facility workers during drift development period.^a

Worker group	Thermal load scenario		
	High (21 years)	Intermediate (21 years)	Low (22 years)
<i>Involved</i>			
Full-time equivalent work years ^b	6,230	6,230	6,530
Total recordable cases of injury and illness	420	420	440
Lost workday cases	300	300	310
Fatalities	0.18	0.18	0.19
<i>Noninvolved</i>			
Full-time equivalent work years	1,670	1,670	1,670
Total recordable cases of injury and illness	19	19	19
Lost workday cases	12	12	12
Fatalities	0.05	0.05	0.05
<i>All workers (totals)^c</i>			
Full-time equivalent work years	7,900	7,900	8,210
Total recordable cases of injury and illness	440	440	460
Lost workday cases	310	310	330
Fatalities	0.23	0.23	0.24

a. Source: Impact rates from Tables F-2 and F-3.

b. Source: Table F-1.

c. Totals might differ from sums due to rounding.

F.2.3.2.2 Radiological Health Impacts to Workers

Radiological health impacts to surface and subsurface facility workers for the operations period are the sum of the estimates of impacts to surface facility workers and subsurface facility workers during operation and monitoring (see Section F.2.3.3.2 for monitoring period).

- Table F-14 lists radiation dose to subsurface facility workers from radiation emanating from waste packages during emplacement operations.

Table F-13. Industrial hazard health and safety impacts to subsurface facility workers during emplacement period.^a

Worker group	For all thermal load scenarios
<i>Involved</i>	
Full-time equivalent work years ^b	1,780
Total recordable cases of injury and illness	53
Lost workday cases	21
Fatalities	0.05
<i>Noninvolved</i>	
Full-time equivalent work years	380
Total recordable cases of injury and illness	13
Lost workday cases	6
Fatalities	0.01
<i>All workers (totals)^c</i>	
Full-time equivalent work years	2,160
Total recordable cases of injury and illness	66
Lost workday cases	29
Fatalities	0.06

a. Source: Impact rates from Table F-2.

b. Source: Table F-1.

c. Totals might differ from sums due to rounding.

Table F-14. Radiological health impacts to subsurface facility workers from waste packages during emplacement period (24 years).^a

Worker group	Thermal load scenario		
	High	Intermediate	Low
<i>Involved</i>			
Full-time equivalent work years ^b	1,780	1,780	1,780
Dose to maximally exposed individual worker (millirem)	4,460	4,510	2,490
Latent cancer fatality probability for MEI ^c	0.002	0.002	0.001
Collective dose (person-rem)	240	240	140
Latent cancer fatality incidence	0.10	0.10	0.05
<i>Noninvolved</i>			
Full-time equivalent work years	380	380	380
Dose to maximally exposed individual worker (millirem)	190	190	190
Latent cancer fatality probability for MEI	0.00008	0.00008	0.00008
Collective dose (person-rem)	3	3	3
Latent cancer fatality incidence	0.001	0.001	0.001
<i>All workers (totals)^d</i>			
Full-time equivalent work years	2,160	2,160	2,160
Collective dose (person-rem)	240	250	140
Latent cancer fatality incidence	0.10	0.10	0.06

a. Source: Exposure data from Table F-5.

b. Source: Table F-1.

c. MEI = maximally exposed individual.

d. Totals might differ from sums due to rounding.

- Table F-15 lists radiation dose to subsurface workers from the ambient radiation in the drifts during emplacement operations. Table F-16 lists radiation doses to subsurface facility workers from ambient radiation during the drift development period.
- Table F-17 lists radiation dose to subsurface workers from inhalation of airborne radon-222 in the drift atmosphere during emplacement operations. Table F-18 lists radiation dose to subsurface workers from inhalation of airborne radon-222 during drift development operations.

Table F-15. Radiological health impacts to subsurface facility workers from ambient radiation during emplacement period.^a

Worker group	Values are independent of thermal load scenario
<i>Involved</i>	
Full-time equivalent work years ^b	1,780
Dose to maximally exposed individual worker (millirem)	960
Latent cancer fatality probability for MEI ^c	0.0004
Collective dose (person-rem)	71
Latent cancer fatality incidence	0.03
<i>Noninvolved</i>	
Full-time equivalent work years	380
Dose to maximally exposed individual worker (millirem)	480
Latent cancer fatality probability for MEI	0.0002
Collective dose (person-rem)	8
Latent cancer fatality incidence	0.003
<i>All workers (totals)^d</i>	
Full-time equivalent work years	2,160
Collective dose (person-rem)	79
Latent cancer fatality incidence	0.03

- a. Source: Exposure data from Table F-5.
b. Source: Table F-1.
c. MEI = maximally exposed individual.
d. Totals might differ from sums due to rounding.

Table F-16. Radiological health impacts to subsurface facility workers from ambient radiation during drift development period.^a

Worker group	Thermal load scenario		
	High (21 years)	Intermediate (21 years)	Low (22 years)
<i>Involved</i>			
Full-time equivalent work years ^b	6,230	6,230	6,530
Dose to maximally exposed individual worker (millirem)	880	880	880
Latent cancer fatality probability for MEI ^c	0.0004	0.0004	0.0004
Collective dose (person-rem)	250	250	260
Latent cancer fatality incidence	0.10	0.10	0.10
<i>Noninvolved</i>			
Full-time equivalent work years	1,670	1,670	1,670
Dose to maximally exposed individual worker (millirem)	660	660	660
Latent cancer fatality probability for MEI	0.0003	0.0003	0.0003
Collective dose (person-rem)	50	50	50
Latent cancer fatality incidence	0.02	0.02	0.02
<i>All workers (totals)^d</i>			
Full-time equivalent work years	7,900	7,900	8,210
Collective dose (person-rem)	300	300	310
Latent cancer fatality incidence	0.12	0.12	0.12

- a. Source: Exposure data from Table F-5.
b. Source: Table F-1.
c. MEI = maximally exposed individual.
d. Totals might differ from sums due to rounding.

Table F-17. Radiological health impacts to subsurface facility workers from airborne radon-222 during emplacement period.^a

Worker group	Thermal load scenario		
	High	Intermediate	Low
<i>Involved</i>			
Full-time equivalent work years ^b	1,780	1,780	1,780
Dose to maximally exposed individual worker (millirem)	1,580	2,160	4,180
Latent cancer fatality probability for MEI ^c	0.0006	0.0008	0.002
Collective dose (person-rem)	120	160	310
Latent cancer fatality incidence	0.05	0.06	0.12
<i>Noninvolved</i>			
Full-time equivalent work years	380	380	380
Dose to maximally exposed individual worker (millirem)	790	1,080	2,090
Latent cancer fatality probability for MEI	0.0003	0.0004	0.0008
Collective dose (person-rem)	13	17	33
Latent cancer fatality incidence	0.005	0.007	0.01
<i>All workers (totals)^d</i>			
Full-time equivalent work years	2,160	2,160	2,160
Collective dose (person-rem)	130	180	340
Latent cancer fatality incidence	0.05	0.07	0.14

- a. Source: Exposure data from Table F-5.
- b. Source: Table F-1.
- c. MEI = maximally exposed individual.
- d. Totals might differ from sums due to rounding.

Table F-18. Radiological health impacts to subsurface facility workers from airborne radon-222 during development period.^a

Worker group	Thermal load scenario		
	High (21 years)	Intermediate (21 years)	Low (22 years)
<i>Involved</i>			
Full-time equivalent work years ^b	6,230	6,230	6,530
Dose to maximally exposed individual worker (millirem)	790	790	790
Latent cancer fatality probability for MEI ^c	0.0003	0.0003	0.0003
Collective dose (person-rem)	220	220	240
Latent cancer fatality incidence	0.09	0.09	0.09
<i>Noninvolved</i>			
Full-time equivalent work years	1,670	1,670	1,670
Dose to maximally exposed individual worker (millirem)	590	590	590
Latent cancer fatality probability for MEI	0.0002	0.0002	0.0002
Collective dose (person-rem)	45	45	45
Latent cancer fatality incidence	0.02	0.02	0.02
<i>All workers (totals)^d</i>			
Full-time equivalent work years	7,900	7,900	8,210
Collective dose (person-rem)	270	270	280
Latent cancer fatality incidence	0.11	0.11	0.11

- a. Source: Exposure data from Table F-5.
- b. Source: Table F-1.
- c. MEI = maximally exposed individual.
- d. Totals might differ from sums due to rounding.

F.2.3.3 Occupational Health and Safety Impacts to Workers During the Monitoring Period

F.2.3.3.1 Health and Safety Impacts to Workers from Workplace Industrial Hazards

Health and safety impacts from industrial hazards common to the workplace for the monitoring period consist of the following:

- Impacts to surface facility workers for the 3-year surface facility decontamination period (Table F-19)
- Impacts to surface facility workers for monitoring support activities (Table F-20)
- Impacts to subsurface facility workers for monitoring and maintenance activities (Table F-21)

Table F-19. Industrial hazard health and safety impacts to surface facility workers during decontamination period.^a

Impact	Uncanistered	Disposable canister	Dual-purpose canister
Full-time equivalent work years ^b	4,060	2,950	3,070
Total recordable cases of injury and illness	120	88	92
Lost workday cases	49	35	37
Fatalities	0.13	0.08	0.11

a. Source: Incident rate data from Table F-2.

b. Source: Table F-1.

Table F-20. Industrial hazard health and safety impacts to surface facility workers during monitoring period.^a

Worker group	Phase	Annual
Full-time equivalent work years ^b	2,660	35
Total recordable cases of injury and illness	80	1.1
Lost workday cases	32	0.42
Fatalities	0.08	0.001

a. Source: Impacts rates from Table F-2.

b. Source: Table F-1.

Table F-21. Industrial hazard health and safety impacts for subsurface facility workers during monitoring period.^a

Worker group	Thermal load scenario		
	High	Intermediate	Low
<i>Involved</i>			
Full-time equivalent work years ^b	5,240	5,240	5,780
Total recordable cases of injury and illness	160	160	170
Lost workday cases	63	63	69
Fatalities	0.15	0.15	0.17
<i>Noninvolved</i>			
Full-time equivalent work years	990	990	990
Total recordable cases of injury and illness	32	32	32
Lost workday cases	16	16	16
Fatalities	0.03	0.03	0.03
<i>All workers (totals)^c</i>			
Full-time equivalent work years	6,230	6,230	6,760
Total recordable cases of injury and illness	190	190	210
Lost workday cases	84	84	91
Fatalities	0.18	0.18	0.20

a. Source: Impacts rates from Table F-2.

b. Source: Table F-1.

c. Totals may differ from sums due to rounding.

For surface monitoring support activities, annual impact values are listed to facilitate the extrapolation of the data for longer and shorter monitoring periods.

F.2.3.3.2 Radiological Health Impacts to Workers

F.2.3.3.2.1 Surface Facility Workers. During monitoring, surface workers would be involved in two types of activities—decontamination for 3 years after the completion of emplacement and support of subsurface monitoring for 76 years (starting at the end of emplacement). Surface workers providing support to the subsurface activities would receive very little radiological dose in comparison to their counterparts involved in subsurface monitoring activities. Therefore, radiological dose impacts were not included for this group; they are estimated in Appendix G, Section G.2. Radiological health impact estimates for the surface facilities decontamination activities are listed in Table F-22.

Table F-22. Radiological health impacts to surface facility workers during decontamination period.^a

Worker group	Uncanistered	Disposable canister	Dual-purpose canister
Full-time equivalent work years ^b	4,060	2,950	3,070
Maximally exposed individual worker (millirem) ^c	300	300	300
Latent cancer fatality probability for MEI ^d	0.0001	0.0001	0.0001
Collective dose (person-rem)	290	210	220
Latent cancer fatality incidence	0.11	0.08	0.09

a. Source: Dose rate data from Table F-5.

b. Source: Table F-1.

c. Source: Based on Table F-4, maximum dose of 100 millirem per year for 3 years.

d. MEI = maximally exposed individual.

F.2.3.3.2.2 Subsurface Facility Workers. Radiological health impacts to subsurface facility workers during monitoring are listed in Table F-23. Maximum worker dose values in the table were based on a maximum work period of 50 years on a monitoring assignment rather than a 76-year monitoring period.

Table F-23. Radiological health impacts to subsurface facility workers during a 50-year work period during a 76-year monitoring period.^a

Worker group	Thermal load scenario		
	High	Intermediate	Low
<i>Involved</i>			
Full-time equivalent work years ^b	5,240	5,240	5,780
Dose to maximally exposed individual worker (millirem)	16,240	18,940	17,610
Latent cancer fatality probability for MEI ^c	0.006	0.008	0.007
Collective dose (person-rem)	1,760	2,050	2,060
Latent cancer fatality incidence ^c	0.71	0.82	0.83
<i>Noninvolved</i>			
Full-time equivalent work years	990	990	990
Dose to maximally exposed individual worker (millirem)	6,200	7,550	8,000
Latent cancer fatality probability for MEI	0.003	0.003	0.003
Collective dose (person-rem)	120	150	160
Latent cancer fatality incidence	0.05	0.06	0.06
<i>All workers (totals)^d</i>			
Full-time equivalent work years	6,230	6,230	6,760
Collective dose (person-rem)	1,880	2,200	2,220
Latent cancer fatality incidence	0.75	0.88	0.89

a. Source: Exposure data from Table F-4.

b. Source: Table F-1.

c. MEI = maximally exposed individual.

d. Totals might differ from sums due to rounding.

In addition, DOE considered monitoring periods as short as 26 years and as long as 276 years. Radiological health impacts for both of these monitoring periods were evaluated; the radiological health impact estimates are listed in Table F-24. Doses to the maximally exposed worker were based on a 50-year employment period rather than the 276-year monitoring period.

Table F-24. Radiological health impacts to workers during a 26-year and a 276-year monitoring period, dual-purpose canister packaging scenario.^a

Group	26 years			276 years		
	High thermal load	Intermediate thermal load	Low thermal load	High thermal load	Intermediate thermal load	Low thermal load
<i>Involved</i>						
Full-time equivalent work years	1,790	1,790	1,980	19,040	19,040	20,980
Dose to maximally exposed individual worker (millirem)	8,440	9,850	9,160	16,240	18,940	17,610
Latent cancer fatality probability for MEI ^b	0.003	0.004	0.004	0.006	0.008	0.007
Collective dose (person-rem)	600	700	710	6,400	7,430	7,500
Latent cancer fatality incidence	0.24	0.28	0.28	2.6	3.0	3.0
<i>Noninvolved</i>						
Full-time equivalent work years	340	340	340	3,590	3,590	3,590
Dose to maximally exposed individual worker (millirem)	3,220	3,930	4,160	6,200	7,550	8,000
Latent cancer fatality probability for MEI	0.001	0.002	0.002	0.002	0.003	0.003
Collective dose (person-rem)	42	51	54	450	540	570
Latent cancer fatality incidence	0.02	0.02	0.02	0.18	0.22	0.23
<i>All workers (totals)</i>						
Full-time equivalent work years	2,130	2,130	2,320	22,630	22,630	24,570
Collective dose (person-rem)	640	750	760	6,850	7,970	8,073
Latent cancer fatality incidence	0.26	0.30	0.30	2.7	3.2	3.2

a. Sources: Tables F-1, F-4, and F-23.

b. MEI = maximally exposed individual.

F.2.3.4 Occupational Health and Safety Impacts During the Closure Phase

F.2.3.4.1 Health and Safety Impacts to Workers from Workplace Industrial Hazards

Health and safety impacts to workers from industrial hazards common to the workplace for closure are listed in Table F-25 for surface facility workers and Table F-26 for subsurface facility workers.

F.2.3.4.2 Radiological Health Impacts to Workers

Radiological health impact to workers from closure activities are the sum of the following components:

- Radiological health impacts to subsurface workers from radiation emanating from the waste packages during the closure phase (Table F-27)
- Radiological impacts to subsurface workers from the ambient radiation field in the drifts during the closure phase (Table F-28)
- Radiological impacts to subsurface workers from inhalation of radon-222 in the drift atmosphere during the closure phase (Table F-29)

Table F-25. Industrial hazard health and safety impacts to surface facility workers during closure phase.^a

Worker group	Waste packaging option		
	Uncanistered	Disposable canister	Dual-purpose canister
<i>Involved</i>			
Full-time equivalent work years ^b	1,580	1,110	1,200
Total recordable cases of injury and illness	97	68	73
Lost workday cases	46	33	35
Fatalities	0.04	0.03	0.03
<i>Noninvolved</i>			
Full-time equivalent work years	600	420	460
Total recordable cases of injury and illness	20	14	15
Lost workday cases	10	7	7
Fatalities	0.02	0.01	0.01
<i>All workers (totals)^c</i>			
Full-time equivalent work years	2,180	1,540	1,650
Total recordable cases of injury and illness	120	82	88
Lost workday cases	56	40	43
Fatalities	0.06	0.04	0.04

a. Source: Impact rates from Table F-2.

b. Source: Table F-1.

c. Totals might differ from sums due to rounding.

Table F-26. Industrial hazard health and safety impacts to subsurface facility workers during closure phase.^a

Worker group	Thermal load scenario		
	High (6 years)	Intermediate (6 years)	Low (15 years)
<i>Involved</i>			
Full-time equivalent work years ^b	1,310	1,310	3,270
Total recordable cases of injury and illness	80	80	200
Lost workday cases	39	39	96
Fatalities	0.04	0.04	0.09
<i>Noninvolved</i>			
Full-time equivalent work years	260	260	660
Total recordable cases of injury and illness	9	9	22
Lost workday cases	4	4	11
Fatalities	0.01	0.01	0.02
<i>All workers (totals)^c</i>			
Full-time equivalent work years	1,570	1,570	3,930
Total recordable cases of injury and illness	89	89	220
Lost workday cases	43	43	110
Fatalities	0.05	0.05	0.11

a. Source: Impact rates from Table F-2.

b. Source: Table F-1.

c. Totals might differ from sums due to rounding.

Because the surface facilities would be largely decontaminated at the beginning of the monitoring period (the exception would be a small facility retained to handle an operations emergency), radiological health impacts to surface facility workers during closure would be small in comparison to those to the subsurface facility workers and so are not included here.

Table F-27. Radiological health impacts to subsurface facility workers from waste package radiation exposures during closure phase.^a

Worker group	Thermal load scenario		
	High (5 years)	Intermediate (6 years)	Low (15 years)
<i>Involved</i>			
Full-time equivalent work years ^b	1,310	1,310	3,270
Dose to maximally exposed individual worker (millirem)	650	650	960
Latent cancer fatality probability for MEI ^c	0.0003	0.0003	0.0004
Collective dose (person-rem)	75	75	110
Latent cancer fatality incidence	0.03	0.03	0.04
<i>Noninvolved</i>			
Full-time equivalent work years	260	260	660
Dose to maximally exposed individual worker (millirem)	48	48	120
Latent cancer fatality probability for MEI	0.00002	0.00002	0.00005
Collective dose (person-rem)	2	2	5
Latent cancer fatality incidence	0.0008	0.0008	0.002
<i>All workers (totals)^d</i>			
Full-time equivalent work years	1,570	1,570	3,930
Collective dose (person-rem)	77	77	115
Latent cancer fatality incidence	0.03	0.03	0.05

- a. Source: Exposure data from Table F-5.
 b. Source: Table F-1.
 c. MEI = maximally exposed individual.
 d. Totals might differ from sums due to rounding.

Table F-28. Radiological health impacts to subsurface facility workers from ambient radiation exposures during closure phase.^a

Worker group	Thermal load scenario		
	High (5 years)	Intermediate (6 years)	Low (15 years)
<i>Involved</i>			
Full-time equivalent work years ^b	1,310	1,310	3,270
Dose to maximally exposed individual worker (millirem)	240	240	600
Latent cancer fatality probability for MEI ^c	0.0001	0.0001	0.0002
Collective dose (person-rem)	52	52	130
Latent cancer fatality incidence	0.02	0.02	0.05
<i>Noninvolved</i>			
Full-time equivalent work years	260	260	660
Dose to maximally exposed individual worker (millirem)	180	180	450
Latent cancer fatality probability for MEI	0.00006	0.00007	0.00018
Collective dose (person-rem)	8	8	20
Latent cancer fatality incidence	0.003	0.003	0.008
<i>All workers (totals)^d</i>			
Full-time equivalent work years	1,570	1,570	3,930
Collective dose (person-rem)	60	60	150
Latent cancer fatality incidence	0.02	0.02	0.06

- a. Source: Exposure data from Table F-5.
 b. Source: Table F-1.
 c. MEI = maximally exposed individual.
 d. Totals might differ from sums due to rounding.

Table F-29. Radiological health impacts to subsurface facility workers from radon-222 exposure during closure phase.^a

Worker group	Thermal load scenario		
	High (5 years)	Intermediate (6 years)	Low (15 years)
<i>Involved</i>			
Full-time equivalent work years ^b	1,310	1,310	3,270
Dose to maximally exposed individual worker (millirem)	1,150	1,480	3,960
Latent cancer fatality probability for MEI ^c	0.0005	0.0006	0.002
Collective dose (person-rem)	250	320	860
Latent cancer fatality incidence	0.10	0.13	0.35
<i>Noninvolved</i>			
Full-time equivalent work years	260	260	660
Dose to maximally exposed individual worker (millirem)	860	1,110	2,970
Latent cancer fatality probability for MEI	0.0003	0.0004	0.001
Collective dose (person-rem)	38	49	130
Latent cancer fatality incidence	0.02	0.02	0.05
<i>All workers (totals)^d</i>			
Full-time equivalent work years	1,570	1,570	3,930
Collective dose (person-rem)	290	370	990
Latent cancer fatality incidence	0.12	0.15	0.40

- a. Source: Exposure data from Table F-5.
- b. Source: Table F-1.
- c. MEI = maximally exposed individual.
- d. Totals might differ from sums due to rounding.

F.3 Human Health and Safety Impact Analysis for Inventory Modules 1 and 2

DOE performed an analysis to estimate the occupational and public health and safety impacts from the emplacement of Inventory Module 1 or 2. Module 1 would involve the emplacement of additional spent nuclear fuel and high-level radioactive waste in the repository; Inventory Module 2 would emplace commercial Greater-Than-Class-C waste and DOE Special-Performance-Assessment-Required waste, which is equivalent to commercial Greater-Than-Class-C waste, in addition to the inventory from Module 1. The volumes of Greater-Than-Class-C and Special-Performance-Assessment-Required waste would be less than that for spent nuclear fuel and high-level radioactive waste (TRW 1999c, Table 3.1). Waste packages containing these materials would be placed between the waste packages containing spent nuclear fuel and high-level radioactive waste (see Chapter 8, Section 8.1.2.1).

With regard to estimating health and safety impacts for the inventory modules, the characteristics of the spent nuclear fuel and high-level radioactive waste were taken to be the same as those for the Proposed Action, but there would be more material to emplace (see Appendix A, Section A.2). As described in Appendix A, the radiological content of the Greater-Than-Class-C waste and Special-Performance-Assessment-Required waste, which is the additional material in Module 2, is much less than that for spent nuclear fuel and high-level radioactive waste. Therefore, the emplacement of the Module 2 material would not meaningfully increase radiological impacts to workers over those estimated for the Module 1 inventory. Further, the facility design parameters, on which the impact estimates are based, are extrapolations from existing designs and have some uncertainty associated with them [see, for example, TRW (1999c), Section 6.2, first paragraph]. Therefore, separate occupational and public health and safety impact analyses were not performed for Module 2 because the impacts for Inventory Modules 1 and 2 would not differ meaningfully.

The calculation of health and safety impacts to workers assumed that the throughput rate of materials for the facility would remain the same as that assumed for the Proposed Action during repository operations (that is, the 70,000-MTHM case). In addition, for the inventory modules the period of operations would be extended to accommodate the additional materials, and the monitoring period would be reduced such that the Yucca Mountain repository operations and monitoring activities would still occur in a 100-year period. Table F-30 summarizes the expected lengths of the phases for Yucca Mountain Repository activities for the inventory modules. These periods were used in the occupational and public health and safety impact calculations.

Table F-30. Expected durations (years) of the Proposed Action and Inventory Modules 1 and 2.^a

Inventory	Construction phase	Operation and monitoring phase (2010-2110)			Total	Closure phase (starts in 2110)
	(2005-2010)	Development ^b	Emplacement	Monitoring		
Proposed Action	5	22	24	76	100 ^c	6-15 ^d
Module 1 or 2	5	36	38	62	100	13-27 ^e

- a. Sources: TRW (1999b, all); TRW (1999c, all); Jessen (1999, all).
- b. Continuing subsurface construction (development) activities are concurrent with emplacement activities.
- c. Closure is assumed to begin 100 years following initial emplacement for the Proposed Action and Module 1 or 2 for the evaluation of cumulative impacts.
- d. 6, 6, and 15 years for the high, intermediate, and low thermal load scenarios, respectively.
- e. 13, 17, and 27 years for the high, intermediate, and low thermal load scenarios, respectively.

This section discusses the methodologies and data used to estimate occupational radiological health and safety impacts resulting from construction, operation and monitoring, and closure of the Yucca Mountain Repository for Inventory Modules 1 and 2, and presents the detailed results. Section F.3.1 describes the methods DOE used to estimate impacts. Section F.3.2 contains tabulations of the detailed data used in the impact calculations and references to the data sources. Section F.3.3 contains detailed tabulations of results.

F.3.1 METHODOLOGY FOR CALCULATING HUMAN HEALTH AND SAFETY IMPACTS

DOE used the methodology described in Section F.2.1 to estimate health and safety impacts for the inventory modules. This methodology involved assembling data for the number of full-time equivalent workers for each repository phase. These numbers were used with statistics for the likelihood of an impact (industrial hazards) or the expected dose rate in the worker environment to calculate health and safety impacts. The way in which the input data was combined in the calculation of health and safety impacts is described in more detail in Section F.2.1. Some of the input data for the calculations for the inventory modules are different from those for the Proposed Action, as discussed in the next section.

F.3.2 DATA SOURCES AND TABULATIONS

F.3.2.1 Full-Time Equivalent Worker-Year Estimates for the Repository Phases for Inventory Modules 1 and 2

The full-time equivalent work-year estimates for the inventory modules are different from those for the Proposed Action. Table F-31 lists the number of full-time equivalent work years for the various repository phases for the inventory modules. Each full-time equivalent work year represents 2,000 work hours, the hours assumed to be worked in a normal work year.

This analysis divides the repository workforce into two groups—involved and noninvolved workers (see Section F.2 for definitions of involved and noninvolved workers). It did not consider workers whose place of employment would be other than at the repository site.

Table F-31. Full-time equivalent work years for various repository periods for Inventory Modules 1 and 2.

Phase	Period	Sources ^a	High thermal load			Intermediate thermal load			Low thermal load		
			UC ^b	DISP ^c	DPC ^d	UC	DISP	DPC	UC	DISP	DPC
<i>Construction</i>											
Surface	44 months	(1)									
Involved worker			2,380	1,650	1,760	2,380	1,650	1,760	2,380	1,650	1,760
Noninvolved worker			900	630	670	900	630	670	900	670	680
Subsurface	5 years	(2)									
Involved worker			2,460	2,460	2,460	2,460	2,460	2,460	2,460	2,460	2,460
Noninvolved worker			600	600	600	600	600	600	600	600	600
<i>Subtotal</i>			<i>6,340</i>	<i>5,340</i>	<i>5,480</i>	<i>6,340</i>	<i>5,340</i>	<i>5,480</i>	<i>6,340</i>	<i>5,380</i>	<i>5,480</i>
<i>Operation and monitoring</i>											
<i>Operation</i>											
Subsurface drift development	36 years	(5)									
Involved worker			9,110	9,110	9,110	9,540	9,540	9,540	10,370	10,370	10,370
Noninvolved worker			2,450	2,450	2,450	2,450	2,450	2,450	2,740	2,740	2,740
Subsurface emplacement	38 years	(4)									
Involved worker			2,810	2,810	2,810	2,810	2,810	2,810	3,000	3,000	3,000
Noninvolved worker			610	610	610	610	610	610	650	650	650
Surface handling	38 years	(3)									
Involved worker			27,700	18,160	18,700	27,700	18,160	18,700	27,700	18,160	18,700
Noninvolved worker			20,820	18,390	18,620	20,820	18,390	18,620	20,820	18,390	18,620
<i>Subtotal operation</i>			<i>63,500</i>	<i>51,530</i>	<i>52,290</i>	<i>63,930</i>	<i>51,960</i>	<i>52,720</i>	<i>65,270</i>	<i>53,310</i>	<i>54,070</i>
<i>Monitoring</i>											
Surface support	62 years	(6)									
Involved worker			2,170	2,170	2,170	2,170	2,170	2,170	2,170	2,170	2,170
Noninvolved worker			NA ^e	NA	NA	NA	NA	NA	NA	NA	NA
Surface facility decontamination	3 years	(7)									
Involved worker			4,060	2,950	3,070	4,060	2,950	3,070	4,060	2,950	3,070
Noninvolved worker			NA	NA	NA	NA	NA	NA	NA	NA	NA
Subsurface monitoring	62 years	(8)									
Involved worker			4,280	4,280	4,280	4,710	4,710	4,710	5,950	5,950	5,950
Noninvolved worker			810	810	810	810	810	810	1,610	1,610	1,610
<i>Subtotal monitoring</i>			<i>11,320</i>	<i>10,200</i>	<i>10,320</i>	<i>11,750</i>	<i>10,640</i>	<i>10,760</i>	<i>13,800</i>	<i>12,680</i>	<i>12,800</i>
<i>Subtotal operation and monitoring</i>			<i>74,820</i>	<i>61,730</i>	<i>62,610</i>	<i>75,680</i>	<i>62,600</i>	<i>63,480</i>	<i>79,070</i>	<i>65,990</i>	<i>66,870</i>
<i>Closure</i>											
Surface	6 years	(9)									
Involved worker			1,580	1,110	1,200	1,580	1,110	1,200	1,580	1,110	1,200
Noninvolved worker			600	420	460	600	420	460	600	420	460
Subsurface	(f)	(10)									
Involved worker			2,830	2,830	2,830	3,710	3,710	3,710	5,890	5,890	5,890
Noninvolved worker			570	570	570	750	750	750	1,190	1,190	1,190
<i>Subtotal closure</i>			<i>5,580</i>	<i>4,940</i>	<i>5,060</i>	<i>6,630</i>	<i>5,940</i>	<i>6,100</i>	<i>9,250</i>	<i>8,610</i>	<i>8,720</i>
Totals^g			86,740	72,020	73,150	88,660	73,930	75,070	94,670	79,980	81,080

- a. Sources: (1) TRW (1999c, Table 6-1); (2) TRW (1999b, Table 6.2.1.1-1); (3) TRW (1999c, Table 6-2); (4) TRW (1999b, Table 6.2.3.1-1); (5) TRW (1999b, Table 6.2.3.1-1); (6) TRW (1999c, Table 6-5); (7) TRW (1999c, Table 6-4); (8) TRW (1999b, Table 6.2.4.1-1); (9) TRW (1999c, Table 6-6); (10) TRW (1999b, Table 6.2.6.1-1).
- b. UC = uncanistered packaging scenario.
- c. DISP = disposable canister packaging scenario.
- d. DPC = dual-purpose canister packaging scenario.
- e. NA = not applicable, all workers assumed to be involved.
- f. High thermal load, 13 years; intermediate thermal load, 17 years; low thermal load, 27 years.
- g. Totals might differ from sums due to rounding.

F.3.2.2 Statistics on Health and Safety Impacts from Industrial Hazards in the Workplace

DOE used the same statistics for health and safety impacts from industrial hazards common to the workplace that were used for the Proposed Action (70,000 MTHM) for analyzing the inventory module impacts (see Table F-2).

F.3.2.3 Estimates of Radiological Exposure Rates and Times for Inventory Modules 1 and 2

DOE used the values in Table F-5 (Proposed Action) for exposure rates, occupancy times, and the fraction of the workforce that would be exposed to estimate radiological health impacts for the inventory module cases, except for doses from the waste packages and from radon-222 inhalation for the subsurface emplacement, monitoring, and closure phases. Annual exposures to subsurface workers for Inventory Modules 1 and 2 from radiation emanating from the waste packages are listed as part of Table F-6. Table F-32 lists annual dose rates from inhalation of radon-222 and its decay products. Section F.1.1.6 discusses the basis for the values in Table F-32.

Table F-32. Correction factors and annual exposures from radon-222 and its decay products for the project phases or periods for Inventory Modules 1 and 2.^a

Subsurface project period	Correction factor			Annual dose rate (millirem per year)		
	High	Intermediate	Low	High	Intermediate	Low
Construction	2.1	2.1	2.1	126	126	126
Drift development	0.6	0.6	0.6	36	36	36
Emplacement	2.0	1.7	3.5	120	120	210
Monitoring	4.2	2.7	4.1	252	160	246
Closure	4.2	2.7	4.1	252	160	246

a. Based on measured value of 60 millirem per year corrected for repository volume and ventilation rate; see the discussions in Section F.1.1.6 and Appendix G (Section G.2.3.1).

F.3.3 DETAILED HUMAN HEALTH AND SAFETY IMPACTS TO WORKERS – INVENTORY MODULES 1 AND 2

F.3.3.1 Construction Phase

F.3.3.1.1 Industrial Hazards to Workers

This section details health and safety impacts to workers from industrial hazards common to the workplace for the construction phase. Impact values for surface workers are the same as those presented for the Proposed Action in Table F-7. Impact values for subsurface workers are presented in Table F-33. The subsurface impacts are independent of thermal load or packaging scenarios.

F.3.3.1.2 Radiological Health Impacts to Workers

Table F-34 lists subsurface worker health impacts from inhalation of radon-222 and its decay products in the subsurface atmosphere and from exposure to natural radiation from radionuclides in the drift walls. The radiological health impacts to surface workers from inhalation of radon-222 and its decay products would be small in comparison to those for subsurface workers; therefore, they are not tabulated here (see Table F-5, Footnote h).

Table F-33. Industrial hazard health and safety impacts to subsurface facility workers during construction phase – Inventory Module 1 or 2.^a

Worker group	Impacts
<i>Involved</i>	
Full-time equivalent work years ^b	2,460
Total recordable cases of injury and illness	150
Lost workday cases	72
Fatalities	0.07
<i>Noninvolved</i>	
Full-time equivalent work years	600
Total recordable cases of injury and illness	20
Lost workday cases	10
Fatalities	0.02
<i>All workers (totals)^c</i>	
Full-time equivalent work years	3,060
Total recordable cases of injury and illness	170
Lost workday cases	82
Fatalities	0.09

a. Source: Impact rates from Table F-2.

b. Source: Table F-31.

c. Totals might differ from sums due to rounding.

Table F-34. Radiological health impacts to subsurface facility workers from radon inhalation and natural exposure for the construction phase – Inventory Modules 1 and 2.^a

Worker group	Radon inhalation exposure	Subsurface ambient exposure
<i>Involved</i>		
Full-time equivalent work years ^c	2,460	2,460
Dose to maximally exposed individual worker (millirem)	630	200
Latent cancer fatality probability for MEI ^c	0.0002	0.00008
Collective dose (person-rem)	310	98
Latent cancer fatality incidence	0.12	0.04
<i>Noninvolved</i>		
Full-time equivalent work years	600	600
Dose to maximally exposed individual worker (millirem)	470	150
Latent cancer fatality probability for MEI	0.0002	0.00006
Collective dose (person-rem)	57	18
Latent cancer fatality incidence	0.02	0.007
<i>All workers (totals)^d</i>		
Full-time equivalent work years	3,060	3,060
Collective dose (person-rem)	370	120
Latent cancer fatality incidence	0.15	0.05

a. Sources: Table F-5 (ambient exposure); Table F-32 (exposure from radon inhalation).

b. Source: Table F-31.

c. MEI = maximally exposed individual.

d. Totals might differ from sums due to rounding.

F.3.3.2 Operation and Monitoring Phase

F.3.3.2.1 Health and Safety Impacts to Workers from Industrial Hazards

This section details health and safety impacts to workers from industrial hazards common to the workplace for the operation and monitoring phase. These impacts would consist of four components:

- Health and safety impacts to surface workers for operations (Table F-35)
- Health and safety impacts to subsurface workers for emplacement and for drift development (Table F-36)
- Health and safety impacts to subsurface workers for the monitoring period (Table F-37)
- Health and safety impacts to surface workers for surface facility decontamination and monitoring support (Table F-38)

Table F-35. Industrial hazard health and safety impacts for surface facility workers during a 38-year operations period by packaging option – Inventory Module 1 or 2.^a

Worker group	Uncanistered	Disposable canister	Dual-purpose canister
<i>Involved</i>			
Full-time equivalent work years ^b	27,700	18,160	18,700
Total recordable cases of injury and illness	830	540	560
Lost workday cases	360	240	240
Fatalities	0.80	0.53	0.55
<i>Noninvolved</i>			
Full-time equivalent work years	20,820	18,390	18,620
Total recordable cases of injury and illness	680	600	610
Lost workday cases	340	300	300
Fatalities	0.60	0.53	0.54
<i>All workers (totals)^c</i>			
Full-time equivalent work years	48,530	36,560	37,320
Total recordable cases of injury and illness	1,520	1,150	1,170
Lost workday cases	700	530	540
Fatalities	1.4	1.1	1.1

a. Source: Impact rates from Table F-2.

b. Source: Table F-31.

c. Totals might differ from sums due to rounding.

F.3.3.2.2 Radiological Health Impacts to Workers

This section details radiological health impacts to workers during the operation and monitoring phase for the inventory modules. These impacts consist of four components:

- Radiological health impacts to surface workers during operations (Table F-39)
- Radiological health impacts to subsurface workers during operations (emplacement and drift development) (Table F-40)
- Radiological health impacts to workers during surface facility decontamination and monitoring support (Table F-41)
- Radiological health impacts to subsurface workers for the monitoring period (Table F-42)

Table F-36. Industrial hazard health and safety impacts for subsurface facility workers for development and emplacement period – Inventory Module 1 or 2.^a

Worker group	High thermal load	Intermediate thermal load	Low thermal load
<i>Involved</i>			
Full-time equivalent work years ^b	11,920	12,350	13,370
Total recordable cases of injury and illness	700	730	790
Lost workday cases	480	500	540
Fatalities	0.35	0.36	0.39
<i>Noninvolved</i>			
Full-time equivalent work years	3,060	3,060	3,380
Total recordable cases of injury and illness	48	48	52
Lost workday cases	27	27	29
Fatalities	0.09	0.09	0.10
<i>All workers (totals)^c</i>			
Full-time equivalent work years	14,980	15,410	16,750
Total recordable cases of injury and illness	750	780	850
Lost workday cases	500	530	570
Fatalities	0.42	0.45	0.49

- a. Source: Impact rates from Tables F-2 and F-3.
 b. Source: Table F-31.
 c. Totals might differ from sums due to rounding.

Table F-37. Industrial hazard health and safety impacts for subsurface facility workers during monitoring period – Inventory Module 1 or 2.^a

Worker group	High thermal load	Intermediate thermal load	Low thermal load
<i>Involved</i>			
Full-time equivalent work years ^b	4,280	4,710	5,950
Total recordable cases of injury and illness	130	140	180
Lost workday cases	55	61	77
Fatalities	0.12	0.14	0.17
<i>Noninvolved</i>			
Full-time equivalent work years	810	810	1610
Total recordable cases of injury and illness	26	26	53
Lost workday cases	13	13	26
Fatalities	0.02	0.02	0.05
<i>All workers (totals)^c</i>			
Full-time equivalent work years	5,080	5,520	7,560
Total recordable cases of injury and illness	160	170	230
Lost workday cases	68	74	100
Fatalities	0.15	0.16	0.22

- a. Source: Impact rates from Table F-2.
 b. Source: Table F-31.
 c. Totals might differ from sums due to rounding.

Table F-38. Industrial hazard health and safety impacts by packaging option to workers during surface facility decontamination and monitoring period – Inventory Module 1 or 2.^a

Involved workers	Uncanistered	Disposable canister	Dual-purpose canister
Full-time equivalent work years ^b	6,230	5,120	5,240
Total recordable cases of injury and illness	190	150	160
Lost workday cases	80	70	70
Fatalities	0.18	0.15	0.15

- a. Source: Impact rates from Table F-2.
 b. Source: Table F-31.

Table F-39. Radiological health impacts to surface facility workers for a 38-year operations period – Inventory Module 1 or 2.^a

Worker group	Uncanistered	Disposable canister	Dual-purpose canister
<i>Involved</i>			
Full-time equivalent work years ^b	27,700	18,160	18,700
Dose to maximally exposed individual worker (millirem)	15,200	15,200	15,200
Latent cancer fatality probability for maximally exposed individual	0.006	0.006	0.006
Collective dose (person-rem)	8,180	3,890	3,950
Latent cancer fatality incidence	3.3	1.6	1.6
<i>Noninvolved</i>			
Full-time equivalent work years	20,820	18,390	18,620
Dose to maximally exposed individual worker (millirem)	950	950	950
Latent cancer fatality probability for maximally exposed individual	0.0004	0.0004	0.0004
Collective dose (person-rem)	170	140	140
Latent cancer fatality incidence	0.07	0.06	0.06
<i>All workers (totals)^c</i>			
Full-time equivalent work years	48,530	36,560	37,320
Collective dose (person-rem)	8,350	4,030	4,090
Latent cancer fatality incidence	3.3	1.6	1.6

a. Source: Exposure data from Table F-5.

b. Source: Table F-31.

c. Totals might differ from sums due to rounding.

Table F-40. Radiological health impacts to subsurface workers for emplacement and drift development during operations period – Inventory Module 1 or 2.^a

Worker group	High thermal load	Intermediate thermal load	Low thermal load
<i>Involved</i>			
Full-time equivalent work years ^b	11,900	12,350	13,370
Dose to maximally exposed individual worker (millirem)	13,220	12,530	13,460
Latent cancer fatality probability for maximally exposed individual	0.005	0.005	0.005
Collective dose (person-rem)	1,530	1,510	1,770
Latent cancer fatality incidence	0.61	0.60	0.71
<i>Noninvolved</i>			
Full-time equivalent work years	3,060	3,060	3,380
Dose to maximally exposed individual worker (millirem)	2,280	2,240	4,290
Latent cancer fatality probability for maximally exposed individual	0.0009	0.0009	0.002
Collective dose (person-rem)	190	190	240
Latent cancer fatality incidence	0.08	0.08	0.10
<i>All workers (totals)^c</i>			
Full-time equivalent work years	14,980	15,410	16,750
Collective dose (person-rem)	1,720	1,700	2,010
Latent cancer fatality incidence	0.69	0.68	0.80

a. Source: Exposure data from Table F-4 except waste package exposures, which are from Table F-6.

b. Source: Table F-31.

c. Totals might differ from sums due to rounding.

Table F-41. Radiological health impacts to surface facility workers for decontamination and monitoring support – Inventory Module 1 or 2.^a

Involved workers	Uncanistered	Disposable canister	Dual-purpose canister
Full-time equivalent work years ^b	6,230	5,120	5,240
Dose to maximally exposed individual worker (millirem)	300	300	300
Latent cancer fatality probability for maximally exposed individual	0.0001	0.0001	0.0001
Collective dose (person-rem)	290	210	220
Latent cancer fatality incidence	0.11	0.08	0.09

a. Source: Exposure data from Table F-4.

b. Source: Table F-31.

Table F-42. Radiological health impacts to subsurface facility workers for a 62-year monitoring period – Inventory Module 1 or 2.^a

Worker group	High thermal load	Intermediate thermal load	Low thermal load
<i>Involved</i>			
Full-time equivalent work years ^b	4,280	4,710	5,950
Dose to maximally exposed individual worker (millirem)	19,240	14,740	16,710
Latent cancer fatality probability for maximally exposed individual	0.008	0.006	0.007
Collective dose (person-rem)	1,700	1,440	2,050
Latent cancer fatality incidence	0.68	0.58	0.82
<i>Noninvolved</i>			
Full-time equivalent work years	810	810	1,610
Dose to maximally exposed individual worker (millirem)	7,700	5,450	7,550
Latent cancer fatality probability for maximally exposed individual	0.003	0.002	0.003
Collective dose (person-rem)	120	88	240
Latent cancer fatality incidence	0.05	0.04	0.10
<i>All workers (totals)^c</i>			
Full-time equivalent work years	5,080	5,520	7,560
Collective dose (person-rem)	2,300	2,050	2,470
Latent cancer fatality incidence	0.92	0.82	3.0

a. Source: Exposure data from Table F-5 except for exposure from waste packages, which is from Table F-6.

b. Source: Table F-31.

c. Totals might differ from sums due to rounding.

F.3.3.3 Closure Phase

F.3.3.3.1 Health and Safety Impacts to Workers from Industrial Hazards

This section details health and safety impacts to workers from industrial hazards common to the workplace for the closure phase. The impacts would consist of two components—impacts to surface workers supporting the closure operations, and impacts to subsurface workers during the closure phase. These impacts are listed in Tables F-43 and F-44, respectively.

Table F-43. Industrial hazard health and safety impacts to surface workers during the closure phase – Inventory Module 1 or 2.^a

Worker group	Uncanistered	Disposable canister	Dual-purpose canister
<i>Involved</i>			
Full-time equivalent work years ^b	1,580	1,110	1,200
Total recordable cases of injury and illness	97	68	73
Lost workday cases	46	33	35
Fatalities	0.05	0.03	0.04
<i>Noninvolved</i>			
Full-time equivalent work years	600	420	460
Total recordable cases of injury and illness	20	14	15
Lost workday cases	10	7	7
Fatalities	0.02	0.01	0.01
<i>All workers (totals)^c</i>			
Full-time equivalent work years	2,180	1,540	1,650
Total recordable cases of injury and illness	116	82	88
Lost workday cases	56	40	43
Fatalities	0.06	0.04	0.05

- a. Source: Impact rates from Table F-2.
 b. Source: Table F-31.
 c. Totals might differ from sums due to rounding.

Table F-44. Health and safety impacts to subsurface facility workers from industrial hazards during the closure phase – Inventory Module 1 or 2.^a

Worker group	High thermal load	Intermediate thermal load	Low thermal load
<i>Involved</i>			
Full-time equivalent work years ^b	2,830	3,710	5,890
Total recordable cases of injury and illness	170	230	360
Lost workday cases	84	110	170
Fatalities	0.08	0.11	0.17
<i>Noninvolved</i>			
Full-time equivalent work years	570	750	1,190
Total recordable cases of injury and illness	19	25	39
Lost workday cases	9	12	19
Fatalities	0.02	0.02	0.03
<i>All workers (totals)^c</i>			
Full-time equivalent work years	3,410	4,450	7,070
Total recordable cases of injury and illness	193	250	400
Lost workday cases	93	120	190
Fatalities	0.10	0.13	0.21

- a. Source: Impact rates from Table F-2.
 b. Source: Table F-31.
 c. Totals might differ from sums due to rounding.

F.4 Human Health and Safety Impact Analysis for the Retrieval Contingency

Nuclear Regulatory Commission regulations state that the period for which DOE must maintain the ability to retrieve waste is at least 50 years after the start of emplacement operations [10 CFR 60.111(b)]. Although DOE does not anticipate retrieval and it is not part of the Proposed Action, the Department would maintain the ability to retrieve the waste for at least 100 years and possibly for as long as 300 years

after the start of emplacement. Factors that could lead to a decision to retrieve the waste would be (1) to protect the public health and safety or the environment or (2) to recover resources from spent nuclear fuel. This EIS evaluates retrieval as a contingency action and describes potential impacts should it occur. The analysis assumes that under this contingency DOE would retrieve all the waste associated with the Proposed Action and would place it on surface storage pads pending future decisions about its ultimate disposition.

The analysis of health and safety impacts to workers divided the retrieval period into two subperiods, as follows:

- First, a construction subperiod in which DOE would (1) build the surface facilities necessary to handle and enclose retrieved waste packages in concrete storage units in preparation for placement on concrete storage pads, and (2) construct the concrete storage pads.

No radioactive materials would be involved in the construction subperiod, so health and safety impacts would be limited to those associated with industrial hazards in the workplace. DOE expects this subperiod would last 2 to 3 years, although construction of the concrete storage pads probably would continue on an as-needed basis during most of the operations subperiod. The analysis assumed a 3-year period.

- Second, an operations subperiod during which the waste packages would be retrieved and moved to the Waste Retrieval Transfer Building. Surface facility workers would unload the waste package from the transfer vehicle and place it on a concrete base. The package and concrete base would then be enclosed in a concrete storage unit that would be placed on the concrete storage pad. The analysis assumed an 11-year period.

This section discusses the methodologies and data used to estimate human health and safety impacts resulting from the retrieval contingency. Section F.4.1 describes the methods DOE used to estimate impacts. Section F.4.2 contains tabulations of the detailed data used in the impact calculations and references to the data sources. Section F.4.3 contains detailed tabulations of the results.

F.4.1 METHODOLOGY FOR CALCULATING HUMAN HEALTH AND SAFETY IMPACTS

DOE used the methodology summarized in Section F.2.1 to estimate health and safety impacts for the retrieval contingency. This involved assembling data for the number of full-time equivalent workers for each retrieval activity. These numbers were used with statistics on the likelihood of an impact (industrial hazards), or the estimated radiological dose rate in the worker environment, to calculate health and safety impacts. The way in which the input data were combined to calculate health and safety impacts is described in more detail in Section F.2.1. Some of the input data in the retrieval impact calculations are different from those for the Proposed Action, as described in the next section.

F.4.2 DATA SOURCES AND TABULATIONS

F.4.2.1 Full-Time Equivalent Work-Year Estimates for the Retrieval Contingency

This analysis divides the repository workforce into two groups—involved and noninvolved workers (see Section F.2 for definitions of involved and noninvolved workers).

Table F-45 lists the number of workers involved in the two subperiods of the retrieval operation and the sources of the numbers. They are tabulated as full-time equivalent work years. Each full-time equivalent

Table F-45. Full-time equivalent work-year estimates for retrieval.

Subperiod and worker group	Length of subperiod (years)	Full-time equivalent work years
<i>Surface facilities, construction^a</i>	3	
Involved		1,130
Noninvolved		430
<i>Surface facilities, retrieval support^b</i>	11	
Involved		320
Noninvolved		870
<i>Subsurface facility retrieval operations^c</i>	11	
Involved		810
Noninvolved		180
Total		3,740

a. Source: TRW (1999c, Table I-2).

b. Source: TRW (1999c, Table I-3).

c. Source: TRW (1999b, Table 6.1.5.1-1).

work year represents 2,000 work hours, the hours assumed to be worked in a normal work year. The full-time equivalent work year estimates are independent of thermal load.

F.4.2.2 Statistics on Health and Safety Impacts from Industrial Hazards in the Workplace

For the retrieval contingency, DOE used the same set of statistics on health and safety impacts from industrial hazards common to the workplace that were used for the Proposed Action (70,000 MTHM) (see Table F-2). The specific statistics that were applied to the retrieval contingency subphases are listed in Table F-46.

Table F-46. Statistics for industrial hazard impacts for retrieval.

Subperiod and worker group	Total recordable incidents (rate per 100 FTEs) ^a	Lost workday cases (rate per 100 FTEs)	Fatalities (rate per 100,000 FTEs) ^b
<i>Construction, surface workers^c</i>			2.9
Involved	6.1	2.9	
Noninvolved	3.3	1.6	
<i>Retrieval, surface workers^d</i>			2.9
Involved	3.0	1.2	
Noninvolved	3.3	1.6	
<i>Retrieval, subsurface workers^d</i>			2.9
Involved	3.0	1.2	
Noninvolved	3.3	1.6	

a. FTE = full-time equivalent work years.

b. Source: Data Set 4, Section F.2.2.

c. Source: Data Set 1, Section F.2.2.

d. Source: Data Set 3, Section F.2.2.

F.4.2.3 Estimated Radiological Exposure Rates and Times for the Retrieval Contingency

DOE used the same set of worker exposure rates and exposure times as those used for evaluating radiological worker impacts for the Proposed Action. Table F-47 presents the specific application of this data to the retrieval contingency subphases. The source of the information is also referenced. The rates used in the analysis did not take into account radioactive decay for the period between emplacement and retrieval.

Table F-47. Radiological doses and exposure data used to calculate worker exposures during retrieval.^a

Subperiod and worker group	Source of exposure	Occupancy factor for exposure rate (fraction of 8-hour workday)	Annual dose (millirem, except where noted)	Full-time equivalent workers ^b	Source ^c
<i>Construction</i>					
<i>Surface</i>					
Involved	None				
Noninvolved	None				
<i>Operations</i>					
<i>Surface</i>					
Involved	Waste package Radiation	1.0	400	13	(1)
			100	16	(1)
Noninvolved		1.0	25	22	(2)
			0	57	(2)
<i>Subsurface</i>					
Involved	Waste package	1.0	Variable	--	(3)
	Radon-222	1.0	Table F-4		(5), Table F-4
	Drift ambient	1.0	40		(4), (5)
Noninvolved	Waste package	0.04 (0.4 for 10% of workers)	0.1 millirem per hour		(7)
	Radon-222	0.4	Table F-4		(6), Table F-4
	Drift ambient	0.4	40		(4), (6)

- a. External exposures include radiation from spent nuclear fuel and high-level radioactive waste packages to surface and subsurface workers, the ambient exposure to subsurface workers from naturally occurring radiation in the drift walls, and subsurface worker exposure from inhalation of radon-222.
- b. Number of full-time equivalent workers by dose category for surface facility activities.
- c. Sources:
- (1) Adapted from TRW (1999c, Table 6.2) for waste receipt, handling, and packaging operations. Values are based on dose rate distribution (fractions) from TRW (1999c, Table 6.2) for involved workers for dual-purpose canister scenario adjusted for fewer workers for retrieval. Forty-five percent of 29 involved workers would be in the 400-millirem-per-year category and 55 percent would be in the 100-millirem-per-year category.
 - (2) Adapted from TRW (1999c, Table 6.2) for waste receipt, handling, and packaging operations. Values based on dose rate distribution (fractions) from TRW (1999c, Table 6.2) for noninvolved workers for dual-purpose canister scenario adjusted for fewer workers for retrieval. Twenty-eight percent of the 79 workers would be in the 25-millirem-per-year category and 72 percent would be in the 0-rem-per-year category.
 - (3) Table F-4.
 - (4) Section F.1.1.6.
 - (5) Rasmussen (1998a, all).
 - (6) Rasmussen (1999, all).
 - (7) Rasmussen (1998b, all).

F.4.3 DETAILED RESULTS FOR THE RETRIEVAL CONTINGENCY

F.4.3.1 Construction Phase

F.4.3.1.1 Human Health and Safety Impacts to Workers from Industrial Hazards

The construction phase would entail only surface-facility activities. Table F-48 summarizes health and safety impacts to workers from industrial hazards during construction. There would be no radiological sources present during surface facility construction activities for retrieval and, hence, no radiological health and safety impacts to workers.

F.4.3.2 Operations Period

F.4.3.2.1 Health and Safety Impacts to Workers from Industrial Hazards

Chapter 4, Table 4-47, summarizes health and safety impacts to workers from industrial hazards common to the workplace for the retrieval operations period. The impacts in that table consist of two

Table F-48. Industrial hazard health and safety impacts to workers during construction.^a

Worker group	Impacts
<i>Involved</i>	
Full-time equivalent work years ^b	1,130
Total recordable cases of injury and illness	69
Lost workday cases	33
Fatalities	0.03
<i>Noninvolved</i>	
Full-time equivalent work years	430
Total recordable cases of injury and illness	14
Lost workday cases	7
Fatalities	0.01
<i>All workers (totals)^b</i>	
Full-time equivalent work years	1,560
Total recordable cases of injury and illness	83
Lost workday cases	40
Fatalities	0.05

a. Source: Impact rates from Table F-46.

b. Source: Table F-45.

components—health impacts to surface workers and health impacts to subsurface workers. Tables F-49 and F-50 list health impacts from industrial hazards during retrieval operations for surface and subsurface workers, respectively.

Table F-49. Industrial hazard health and safety impacts to surface facility workers during retrieval.^a

Worker group	Impacts
<i>Involved</i>	
Full-time equivalent work years ^b	320
Total recordable cases of injury and illness	10
Lost workday cases	4
Fatalities	0.009
<i>Noninvolved</i>	
Full-time equivalent work years	870
Total recordable cases of injury and illness	29
Lost workday cases	14
Fatalities	0.03
<i>All workers (totals)^c</i>	
Full-time equivalent work years	1,190
Total recordable cases of injury and illness	37
Lost workday cases	18
Fatalities	0.03

a. Source: Impact rates from Table F-46.

b. Source: Table F-45.

c. Totals might differ from sums due to rounding.

F.4.3.2.2 Radiological Health and Safety Impacts to Workers

Potential radiological health impacts to workers during the operations period of retrieval consist of the following components:

- Impacts to surface facility workers involved in handling the waste packages and placing them in concrete storage units

Table F-50. Industrial hazard health and safety impacts to subsurface facility workers during retrieval.^a

Worker group	Impacts
<i>Involved</i>	
Full-time equivalent work years ^b	810
Total recordable cases of injury and illness	24
Lost workday cases	11
Fatalities	0.02
<i>Noninvolved</i>	
Full-time equivalent work years	180
Total recordable cases of injury and illness	6
Lost workday cases	3
Fatalities	0.01
<i>All workers (totals)^b</i>	
Full-time equivalent work years	990
Total recordable cases of injury and illness	30
Lost workday cases	13
Fatalities	0.03

a. Source: Impact rates from Table F-46.

b. Source: Table F-45.

c. Totals might differ from sums due to rounding.

- Impacts to subsurface facilities workers from direct radiation emanating from the waste packages
- Impacts to subsurface workers from inhalation of radon-222 in the atmosphere of the drifts
- Impacts to subsurface workers from ambient radiation from naturally occurring radionuclides in the drift walls

Tables F-51 and F-52 list potential radiological health impacts for each of these component parts. The impacts to subsurface workers only vary slightly (less than 2 percent) with thermal load and are highest for the low thermal load. Thus, the values in Table F-52 for the low thermal load case, would produce the largest impacts.

Table F-51. Radiological health impacts to surface facility workers from waste handling during retrieval.^a

Worker group	Impacts
<i>Involved</i>	
Full-time equivalent work years ^b	320
Maximally exposed individual dose (millirem)	4,400
Latent cancer fatality probability for maximally exposed individual	0.002
Collective dose (person-rem)	75
Latent cancer fatality incidence for overall worker group	0.03
<i>Noninvolved</i>	
Full-time equivalent work years	870
Maximally exposed individual dose (millirem)	280
Latent cancer fatality probability for maximally exposed individual	0.0001
Collective dose (person-rem)	6
Latent cancer fatality incidence for overall worker group	0.002
<i>All workers (totals)^c</i>	
Full-time equivalent work years	1,190
Collective dose (person-rem)	81
Latent cancer fatality	0.03

a. Source: Exposure rate data from Table F-47.

b. Source: Table F-45.

c. Totals might differ from sums due to rounding.

Table F-52. Components of radiological health impacts to subsurface workers during retrieval for the low thermal load scenario.^{a,b}

Group	Source of exposure			Total ^c
	Waste packages	Ambient	Radon-222 inhalation	
<i>Involved</i>				
Full-time equivalent work years ^d	840	840	840	840
Maximally exposed individual dose (millirem)	4,400	440	2,110	6,950
Latent cancer fatality probability for maximally exposed individual	0.002	0.0002	0.0008	0.003
Collective dose (person-rem)	200	33	160	390
Latent cancer fatality incidence for overall worker group	0.08	0.01	0.06	0.16
<i>Noninvolved</i>				
Full-time equivalent work years	180	180	180	180
Maximally exposed individual dose (millirem)	88	220	1,060	1,370
Latent cancer fatality probability for maximally exposed individual	0.00004	0.00009	0.0004	0.0005
Collective dose (person-rem)	1	4	17	22
Latent cancer fatality incidence for overall worker group	0.0004	0.001	0.007	0.009
<i>All workers (totals)^c</i>				
Full-time equivalent work years	1,010	1,010	1,010	1,010
Collective dose (person-rem)	200	37	180	420
Latent cancer fatality incidence for overall worker group	0.08	0.01	0.07	0.17

a. Source: Exposure data from Table F-47.

b. The variation in values among the thermal load scenarios was small. Therefore, only the largest values (for the low thermal load) are listed.

c. Totals might differ from sums due to rounding.

d. Source: Table F-45.

REFERENCES

- DOE 1995 DOE (U.S. Department of Energy), 1995, *YMP Erionite Control Protocol*, Office of Civilian Radioactive Waste Management, Yucca Mountain Project Office, Las Vegas, Nevada. [MOL.19950925.0124]
- DOE 1998a DOE (U.S. Department of Energy), 1998a, *Implementation Guide for use with DOE Order 440.1 Occupational Exposure Assessment*, DOE G 440.1-3, Department of Energy, Office of Worker Health and Safety, Washington, D.C. [240305]
- DOE 1998b DOE (U.S. Department of Energy), 1998b, *Air Quality Control Design Analysis*, BCAD00000-01717-0200-00008, Revision 00, Office of Civilian Radioactive Waste Management, Washington, D.C. [MOL.19980729.0044]
- DOE 1999 DOE (U.S. Department of Energy), 1999, "CAIRS Database, DOE and Contractor Injury and Illness Experience by Operation Type by Year and Quarter, 1993 through 1998," <http://tis.eh.doe.gov/cairs/cairs/dataqtr/q984a.htm>, May 22, Washington, D.C. [244036]

- Eckerman and Ryman 1993 Eckerman, K. F., and J. C. Ryman, 1993, *External Exposure to Radionuclides in Air, Water, and Soil, Exposure-to-Dose Coefficients for General Application, Based on the 1987 Federal Radiation Protection Guidance: Federal Guidance Report No. 12*, EPA 402-R-93-081, Office of Radiation and Indoor Air, U.S. Environmental Protection Agency, Washington D.C. [225472]
- Eckerman, Wolbarst, and Richardson 1988 Eckerman, K. F., A. B. Wolbarst, and A. C. B. Richardson, 1988, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, Federal Guidance Report No. 11, EPA-520/1-88-020, U.S. Environmental Protection Agency, Office of Radiation Programs, Oak Ridge National Laboratory, Oak Ridge, Tennessee. [203350]
- EPA 1996 EPA (U.S. Environmental Protection Agency), 1996, *Ambient Levels and Noncancer Health Effects of Inhaled Crystalline and Amorphous Silica: Health Issue Assessment*, EPA/600/R-95/115, National Center for Environmental Assessment, Office of Research and Development, Washington, D.C. [243562]
- Gotchy 1987 Gotchy, R. L., 1987, *Potential Health and Environmental Impacts Attributable to the Nuclear and Coal Fuel Cycles*, NUREG-0332, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. [234603]
- IARC 1987 IARC (International Agency for Research on Cancer), 1987, *Silica and Some Silicates*, World Health Organization, United Nations, Lyon, France. [226502]
- IARC 1997 IARC (International Agency for Research on Cancer), 1997, *IARC Monographs on the Evaluation of Carcinogenic Risks to Humans, Silica, Some Silicates, Coal Dust and para-Aramid Fibrils, Volume 68*, IARC Working Group on the Evaluation of Carcinogenic Risks to Humans, World Health Organization, United Nations, Lyon, France. [236833]
- ICRP 1977 ICRP (International Commission on Radiological Protection), 1977, *Recommendations of the International Commission on Radiological Protection*, ICRP Publication 26, Pergamon Press, Elmsford, New York. [221568]
- ICRP 1991 ICRP (International Commission on Radiological Protection), 1991, *1990 Recommendations of the International Commission on Radiological Protection*, Publication 60, Volume 21, Numbers 1-3, Pergamon Press, Elmsford, New York. [235864]
- ICRP 1994 ICRP (International Commission on Radiological Protection), 1994, *Protection Against Radon-222 at Home and at Work*, Publication 65, Pergamon Press, Oxford, Great Britain. [236754]
- Jessen 1999 Jessen, J., 1999, "Final Closure Phase Years based on March 99 EF's," electronic communication to Ikenberry et al., Jason Technologies Corporation, Las Vegas, Nevada. [MOL.19990526.0030]

- Kamrin 1988 Kamrin, M. A., 1988, *Toxicology – A Primer on Toxicology Principles and Applications: Indoor & Outdoor Air, Drinking Water, Food, Workplace Environment*, Lewis Publishers, Inc., Chelsea, Michigan. [243888]
- Maheras and Thorne 1993 Maheras, S. J., and D. J. Thorne, 1993, *New Production Reactor Exposure Pathways at the Idaho National Engineering Laboratory*, NPR-8957, EG&G Idaho, Inc., Idaho Falls, Idaho. [243737]
- McKenzie 1998 McKenzie, D., 1998, "Erionite Encounters in Expanded Layouts," electronic mail to D. Walker (Jason Technologies Corporation), December 21, Morrison Knudsen Corporation, Las Vegas, Nevada. [MOL.19990511.0294]
- Mettler and Upton 1995 Mettler, F. A., Jr., and A. C. Upton, 1995, *Medical Effects of Ionizing Radiation, Second Edition*, W. B. Saunders Company, Philadelphia, Pennsylvania. [244122]
- MSHA 1999 MSHA (Mine Safety and Health Administration), 1999, "Table 6. – Number of Contractor Injuries, Injury-Incidence Rates, Average Numbers of Employees, and Employee Hours, by Work Location and Mineral Industry," <http://www.msha.gov.stats/wq964+06.htm>, March 11, Washington, D.C. [243568]
- NCRP 1987 NCRP (National Council on Radiation Protection and Measurements), 1987, *Ionizing Radiation Exposure of the Population of the United States: Recommendations of the National Council on Radiation Protection and Measurements*, Report No. 93, Bethesda, Maryland. [229033]
- NCRP 1993 NCRP (National Council on Radiation Protection and Measurements), 1993, *Risk Estimates for Radiation Protection*, Report No. 115, Bethesda, Maryland. [232971]
- NCRP 1996 NCRP (National Council on Radiation Protection and Measurements), 1996, *Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground, Recommendations of the National Council on Radiation Protection and Measurements*, Report No. 123, Bethesda, Maryland. [225158, Volume 1; 234986, Volume 2]
- NIOSH 1996 NIOSH (National Institute for Occupational Safety and Health), 1996, "Silica, crystalline (as respirable dust), IDLH Documentation" (downloaded from <http://www.cdc.gov/niosh/idlh/14808607.html>, April 8, 1999), Washington, D.C. [243424]
- NJDHSS 1996 NJDHSS (New Jersey Department of Health and Senior Services), 1996, "Hazardous Substance Fact Sheet – Silica, Cristobalite," Trenton, New Jersey. [243425]
- Rasmussen 1998a Rasmussen, D. G., 1998a, "Radiation exposure information," electronic communication with J. Jessen (Jason Technologies Corporation), July 22, TRW Environmental Safety Systems Inc., Las Vegas, Nevada. [MOL.19990526.0029]

- Rasmussen 1998b Rasmussen, D. G., 1998b, "Radiation exposure information," electronic communication with R. Orthen (Jason Technologies Corporation), July 29, TRW Environmental Safety Systems Inc., Las Vegas, Nevada. [MOL.19990511.0386]
- Rasmussen 1999 Rasmussen, D., 1999, "Additional matrix," electronic communication with attachment to D. Walker (Jason Technologies Corporation), April 16, TRW Environmental Safety Systems Inc., Las Vegas, Nevada. [MOL.19990602.0180]
- Stewart 1998 Stewart B., 1998, "YMP EIS Information Request – CAIRS Statistics for Construction and Non-Construction Activities During TBM Operations," electronic communication with attachment to J. Steinhoff (TRW Environmental Safety Systems Inc.), December 17, Las Vegas, Nevada. [MOL.19990511.0298]
- Technical Resources 1994 Technical Resources, Inc., 1994, *Seventh Annual Report on Carcinogens 1994*, Rockville, Maryland. [243694]
- TRW 1999a TRW (TRW Environmental Safety Systems Inc.), 1999a, *Environmental Baseline File for Human Health*, B00000000-01717-5705-00114, Revision 01, Las Vegas, Nevada. [MOL.19990608.0035]
- TRW 1999b TRW (TRW Environmental Safety Systems Inc.), 1999b, *Engineering File – Subsurface Repository*, BCA000000-01717-5705-00005, Revision 02 with DCN1, Las Vegas, Nevada. [MOL.19990622.0202, document; MOL.19990621.0157, DCN1]
- TRW 1999c TRW (TRW Environmental Safety Systems Inc.), 1999c, *Repository Surface Design Engineering Files Report*, BCB000000-01717-5705-00009, Revision 03, Las Vegas, Nevada. [MOL.19990615.0238]



Appendix G

Air Quality

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APPENDIX G. AIR QUALITY

Potential releases of nonradiological and radiological pollutants associated with 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 and additional data and intermediate results that the U.S. Department of Energy (DOE) used to estimate impacts from potential releases to air. Final results are presented in Chapter 4, Section 4.1.2, and Chapter 8, Section 8.2.2.

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 very small quantities of hazardous and toxic pollutants; therefore, these pollutants were not considered in the analysis. Concentrations of six criteria pollutants are regulated under the National Ambient Air Quality Standards (40 CFR Part 50) established by the Clean Air Act. This analysis evaluated releases and potential impacts of four of these pollutants—carbon monoxide, nitrogen dioxide, sulfur dioxide, and particulate matter with an aerodynamic diameter of 10 micrometers or less (PM_{10})—quantitatively. It addresses the other two criteria pollutants—lead and ozone—and the concentration of particulate matter with an aerodynamic diameter of 2.5 micrometers or less ($PM_{2.5}$), qualitatively. In addition, this analysis considers potential releases to air of cristobalite, a form of crystalline silica that can cause silicosis and is a potential carcinogen. These pollutants could be released during all project phases. Section G.1 describes the methods DOE used to calculate impacts from releases of criteria pollutants and cristobalite.

Radionuclides that repository-related activities could release to the atmosphere include the noble gas krypton-85 from spent nuclear fuel handling during the operation and monitoring phase, and naturally occurring radon-222 and its decay products from ventilation of the subsurface facility during all project phases. Other radionuclides would not be released or would be released in such small quantities they would result in very small impacts to air quality. Such radionuclides are not discussed further in this appendix. Section G.2 describes the methods DOE used to calculate impacts of radionuclide releases.

G.1 Nonradiological Air Quality

This section describes the methods DOE used to analyze potential impacts to air quality at the proposed Yucca Mountain Repository from releases of nonradiological air pollutants during the construction, operation and monitoring, and closure phases, and a retrieval scenario. It also describes intermediate results for various repository activities. Table G-1 lists the six criteria pollutants regulated under the National Ambient Air Quality Standards or the Nevada Administrative Code along with their regulatory limits and the periods over which pollutant concentrations are averaged. The criteria pollutants addressed quantitatively in this section are nitrogen dioxide, sulfur dioxide, particulate matter 10 micrometers or less in aerodynamic diameter (PM_{10}), and carbon monoxide. Lead was not considered further in this analysis because there would be no airborne sources at the repository. Particulate matter 2.5 micrometers or less in aerodynamic diameter ($PM_{2.5}$) and ozone are discussed below, as is cristobalite, a mineral occurring naturally in the subsurface rock at Yucca Mountain.

The U.S. Environmental Protection Agency revised the primary and secondary standards for particulate matter in 1997 (62 *FR* 38652, July 18, 1997), establishing annual and 24-hour $PM_{2.5}$ standards at 15 micrograms per cubic meter and 65 micrograms per cubic meter, respectively. Primary standards set limits to protect public health, including the health of “sensitive” populations. Secondary standards set limits to protect public welfare, including protection against decreased visibility, damage to animals, crops, vegetation, and buildings. Because the new particulate standard will regulate $PM_{2.5}$ for the first time, the agency has allowed 5 years for the creation of a national monitoring network and the analysis of collected data to help develop state implementation plans. The new $PM_{2.5}$ standards have not been implemented and the imposition of local area controls will not be required until 2005. By definition, $PM_{2.5}$ levels can be no more than, and in the real world are always substantially less than, PM_{10} levels. In

Table G-1. Criteria pollutants and regulatory limits.

Pollutant	Period	Regulatory limit ^a	
		Parts per million	Micrograms per cubic meter
Nitrogen dioxide	Annual	0.053	100
Sulfur dioxide	Annual	0.03	80
	24-hour	0.14	365
	3-hour	0.50	1,300
	8-hour	9	10,000
Carbon monoxide	1-hour	35	40,000
	Annual		50
PM ₁₀	24-hour		150
	Annual		15
PM _{2.5} ^b	24-hour		65
	8-hour	0.08	157
Ozone	1-hour	0.12 ^c	235
	Quarterly		1.5

a. Sources: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

b. Standard not yet implemented.

c. The 1-hour standard does not apply to Nevada because the State was in attainment when the 8-hour standard was adopted in July 1997.

general, PM_{2.5} levels would be approximately one-third of the PM₁₀ levels. As the analysis for PM₁₀ shows, even the maximum PM₁₀ levels that could be generated by the Proposed Action are substantially below the PM_{2.5} standards. Thus, although no detailed PM_{2.5} analysis has been conducted, the PM₁₀ analysis can be regarded as a surrogate for a PM_{2.5} analysis and illustrates that potential PM_{2.5} levels would be well below applicable regulatory standards.

The purpose of the ozone standard is to control the ambient concentration of ground-level ozone, not naturally occurring ozone in the upper atmosphere. Ozone is not emitted directly into the air; rather, it is formed when volatile organic compounds react in the presence of sunlight. Nitrogen dioxides are also important precursors to ozone. Small quantities of volatile organic compounds would be released from repository activities; the peak annual release would be about 540 kilograms (1,200 pounds) (TRW 1999a, Table 6-2, page 75). Because Yucca Mountain is in an attainment area for ozone, the analysis compared the estimated annual release to the Prevention of Significant Deterioration of Air Quality emission threshold for volatile organic compounds from stationary sources (40 CFR 52.21). The volatile organic compound emission threshold is 35,000 kilograms (77,000 pounds) per year, so the peak annual release from the repository would be well below this level. Accordingly, the analysis did not address volatile organic compounds and ozone further, although this does not preclude future, more detailed analyses if estimates of volatile organic compound emissions change.

Cristobalite, one of several naturally occurring crystalline forms of silica (silicon dioxide), is a major mineral constituent of Yucca Mountain tuffs (TRW 1999b, page 4-81). Prolonged high exposure to crystalline silica can cause silicosis, a disease characterized by scarring of lung tissue. An increased cancer risk to humans who already have developed adverse noncancer effects from silicosis has been shown, but the cancer risk to otherwise healthy individuals is not clear (EPA 1996, page 1-5). Cristobalite is principally a concern for involved workers because it could be inhaled during subsurface excavation operations. Appendix F, Section F.1, contains additional information on crystalline silica.

While there are no limits for exposure of the general public to cristobalite, there are limits to workers for exposure (29 CFR 1000.1910). Therefore, this analysis used a comparative benchmark of 10 micrograms per cubic meter, based on a cumulative lifetime exposure of 1,000 micrograms per cubic meter multiplied by years (that is, the average annual exposure concentration times the number of years exposed). At this level, an Environmental Protection Agency health assessment (EPA 1996, pages 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 was rounded down to 10 micrograms per cubic meter to establish the benchmark.

Cristobalite would be emitted from the subsurface in exhaust ventilation air during excavation operations and would be released as fugitive dust from the excavated rock pile, so members of the public and noninvolved workers could be exposed. Fugitive dust from the excavated rock pile would be the largest potential source of cristobalite exposure to the public. The analysis assumed that 28 percent of the fugitive dust released from this rock pile and from subsurface excavation would be cristobalite, reflecting the cristobalite content of the parent rock, which ranges from 18 to 28 percent (TRW 1999b, page 4-81). Using the parent rock percentage probably overestimates the airborne cristobalite concentration, because 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 will deposit rapidly on the surface (EPA 1996, page 3-26).

G.1.1 COMPUTER MODELING AND ANALYSIS

DOE used the Industrial Source Complex computer program to estimate the annual and short-term (24-hour or less) air quality impacts at the proposed Yucca Mountain Repository. The Department has used this program in recent EISs (DOE 1995, all; 1997a,b, all) to estimate nonradiological air quality impacts. The program contains both a short-term model (which uses hourly meteorological data) and a long-term model (which uses joint frequency meteorological data). The program uses steady-state Gaussian plume models to estimate pollutant concentrations from a variety of sources associated with industrial complexes (EPA 1995a, all). This modeling approach assumes that (1) the time-averaged pollutant concentration profiles at any distance downwind of the release point may be represented by a Gaussian (normal) distribution in both the horizontal and vertical directions; and (2) the meteorological conditions are constant (persistent) over the time of transport from source to receptor. The Industrial Source Complex program is appropriate for either flat or rolling terrain, and for either urban or rural environments. The Environmental Protection Agency has approved this program for specific regulatory applications. Input requirements for the program include source configuration and pollutant emission parameters. The short-term model was used in this analysis to estimate all nonradiological air quality impacts and uses hourly meteorological data that include wind speed, wind direction, and stability class to compute pollutant transport and dispersion.

Because the short-term pollutant concentrations were based on annual usage or release parameters, conversion of annual parameter values to short-term values depended on the duration of the activity. Many of the repository activities were assumed to 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 estimating pollutant emissions at the Yucca Mountain site. In these cases, generic information was used and conservative assumptions were made that tended to overestimate actual air concentrations.

As noted in Section G.1, the total nonradiological air quality impacts are described in Chapter 4, Section 4.1.2, for the Proposed Action and in Chapter 8, Section 8.2.2, for the inventory modules. These impacts are the sum of air quality impacts from individual sources and activities that take place during each of the project phases and that are discussed later in this section (for example, dust emissions from the concrete batch facility during the construction phase). The maximum air quality impact (that is, air concentration) resulting from individual sources or activities could occur at different land withdrawal area boundary locations depending on the release period and the regulatory averaging time (see Section G.1.3). These maximums generally occur in a westerly or southerly direction. The total nonradiological air quality impacts presented in Sections 4.1.2 and 8.2.2 are the sum of the calculated maximum concentrations regardless of direction. Therefore, the values presented would be larger than the actual sum of the

concentrations for a particular distance and direction. This approach was selected to simplify the presentation of air quality results.

G.1.2 LOCATIONS OF HYPOTHETICALLY EXPOSED INDIVIDUALS

The location of the public maximally exposed individual was determined by calculating the maximum ground-level pollutant concentrations. Because unrestricted public access would be limited to the site boundary, the analysis 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 G-1).

Table G-2 lists the distances from the North and South Portals to the land withdrawal area boundary where the analysis assumed members of the public would be present. 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 Nellis Air Force Range). The distance to the nearest unrestricted public access in these directions would be so large that there would be no air quality impacts. 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 traveling in these directions tend to enter Fortymile Wash and turn south. The analysis used the distance to the south land withdrawal area boundary for those sectors.

Table G-2. Distance to the nearest point of unrestricted public access (kilometers).^{a,b,c}

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 ^d	21	18
Southeast ^d	21	18

a. Source: DOE (1997c, all).

b. Numbers are rounded to two significant figures.

c. To convert kilometers to miles, multiply by 0.6217.

d. Distances assumed to be the same as those to the south.

G.1.3 METEOROLOGICAL DATA AND REFERENCE CONCENTRATIONS

DOE estimated the concentrations of criteria pollutants in the region of the repository by using the Industrial Source Complex program and site-specific meteorological data for 1993 to 1997 from air quality and meteorology monitoring Site 1 (TRW 1999c, electronic addendum). Site 1 is less than 1 kilometer (0.6 mile) south of the proposed North Portal surface facility location. Similar topographic exposure leads to similar prevailing northerly and southerly winds at both locations. DOE used Site 1 data because an analysis of the data collected at all the sites showed that site to be most representative of the surface facilities (TRW 1999c, page 7). Wind speed data are from the 10-meter (33-foot) level, as are atmospheric stability data, using the night-adjusted sigma-theta method (EPA 1987, pages 6-20 to 6-32). Mixing height measurements were not available for Yucca Mountain so the analysis assumed a mixing height of approximately 140 meters (470 feet), which is one-tenth of the 1,420 meters (4,700 feet) mixing-layer depth for Desert Rock, Nevada. Desert Rock is the nearest upper air meteorological station, about 44 kilometers (27 miles) east-southeast near Mercury, Nevada. The average mixing height at Desert Rock was divided by 10 to simulate the mixing height during very stable conditions, which is when the highest concentrations from a ground-level source would normally occur. All nonradiological

pollutant releases were assumed to come from ground-level point sources. Both of these conservative assumptions, made because of a lack of site-specific information, tend to overestimate actual air concentrations. Fugitive dust emissions could be modeled as an area source, but the distance from the source to the exposure location would be large [more than 10 kilometers (6 miles)] so a point source provides a good approximation. Some sources would have plume rise, such as boiler emissions, but this was not considered because there is inadequate information to characterize the rise.

The analysis estimated unit release concentrations at the land withdrawal area boundary points of maximum exposure for ground-level point-source releases. 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, or 1-hour). Various activities at the Yucca Mountain site could result in pollutants being released over four different periods in a 24-hour day [continuously, 8-hour, 12-hour (two 6-hour periods), or 3-hour]. 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 to be zero for all other hours of the day. Similarly, it assumed that the 3-hour releases would occur from 9 a.m. to 12 p.m. and to 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 G-3 lists the maximum unit release concentrations for the 11 combinations of the Yucca Mountain site-specific release periods and regulatory limit averaging times. The analysis estimated the unit

Table G-3. Unit release concentrations (micrograms per cubic meter based on a release of 1 gram per second) and direction to maximally exposed individual location for 11 combinations of 4 release periods and 5 regulatory limit averaging times.^a

	Direction from South Portal Operations area	Unit release concentration	Direction from North Portal Operations Area	Unit release concentration
<i>Continuous release – annual average concentration (1995)^b</i>				
	South-southeast	0.12	South-southeast	0.099
<i>Continuous release – 24-hour average concentration (1993)</i>				
	Southeast	1.0	West	0.95
<i>Continuous release – 8-hour average concentration (1995)</i>				
	Southeast	3.0	Southeast	2.5
<i>Continuous release – 3-hour average concentration (1995)</i>				
	West	6.1	West	6.1
<i>Continuous release – 1-hour average concentration (1995)</i>				
	West	18	West	18
<i>8-hour release (8 a.m. to 4 p.m.) – 24-hour average concentration (1997)</i>				
	West-southwest	0.19	West-northwest	0.18
<i>8-hour release (8 a.m. to 4 p.m.) – 8-hour average concentration (1997)</i>				
	West-southwest	0.57	West-northwest	0.52
<i>8-hour release (8 a.m. to 4 p.m.) – 3-hour average concentration (1997)</i>				
	West-southwest	1.5	West-northwest	1.4
<i>8-hour release (8 a.m. to 4 p.m.) – 1-hour average concentration (1997)</i>				
	West-northwest	3.3	West-northwest	3.3
<i>12-hour release (9 a.m. to 3 p.m. and 5 p.m. to 11 p.m.) – 24-hour average concentration (1997)</i>				
	West	0.95	West	0.95
<i>3-hour release (9 a.m. to 12 p.m.) – 24-hour average concentration (1997)</i>				
	West-northwest	0.17	West-northwest	0.17

a. Numbers are rounded to two significant figures.

b. Number in parentheses is the year from 1993 through 1997 for which meteorological data would result in the highest unit concentration.

concentrations and directions using the meteorological data during a single year from 1993 through 1997 (TRW 1999c, electronic addendum) that would result in the highest unit concentration. For all years, the unit release concentrations for a particular averaging time are within a factor of 2 of each other. Table G-3 lists the 24-hour averaged concentration for the 3- and 12-hour release scenarios because the activities associated with these scenarios would only release PM₁₀, which has annual and 24-hour regulatory limits. The estimated concentration at the point of exposure was calculated by multiplying the estimated source release rate (presented for each source in the following sections) by the maximum unit release concentration for that averaging period.

G.1.4 CONSTRUCTION PHASE

This section describes the method used to estimate air quality impacts during the 5-year construction phase. DOE would complete the surface facilities during the construction phase, as well as sufficient excavation of the subsurface to support initial emplacement activities.

This analysis used calculations of the pollutant concentrations from various construction activities to determine air quality impacts. To calculate these impacts, estimated pollutant emission rates discussed in this section were multiplied by the unit release concentration (see Section G.1.3). This produced the pollutant concentration for comparison to regulatory limits. Short-term pollutant emission rates and concentrations were estimated using the method described in Section G.1.1.

The principal emission sources of particulates would be fugitive dust from construction activities on the surface, excavation of rock from the repository, storage of material on the excavated rock pile, and dust emissions from the concrete batch facility. The principal sources of nitrogen dioxide, sulfur dioxide, and carbon monoxide would be fuel combustion in trucks, cranes, and graders and emissions from a boiler in the South Portal Operations Area. Nitrogen dioxide, sulfur dioxide, and carbon monoxide would also be emitted during maintenance of the excavated rock pile. The following sections describe these sources in more detail.

G.1.4.1 Fugitive Dust Emissions from Surface Construction

Fugitive dust would be generated during such construction activities as earth moving and truck traffic. All surface construction activities and associated fugitive dust releases were assumed to occur during 250 working days per year with one 8-hour shift per day. The preferred method suggested by the Environmental Protection Agency would be to break the construction activities into component activities (for example, earth moving, truck traffic) and calculate the emissions for each component. However, detailed information was not available for the construction phase, so a generic, conservative approach was taken. The release rate of total suspended particulates (particulates with aerodynamic diameters of 30 micrometers or less) was estimated as 0.27 kilogram per square meter (1.2 tons per acre) per month (EPA 1995b, pages 13.2.3-1 to 13.2.3-7). This estimated emission rate for total suspended particulates was based on measurements made during the construction of apartments and shopping centers.

The amount of PM₁₀ (the pollutant of interest) emitted from the construction of the Yucca Mountain Repository probably would be less than 0.27 kilogram per square meter (1.2 tons per acre) per month because many of the particulates suspended during construction would be at the larger end of the 30-micrometer range and would tend to settle rapidly (Seinfeld 1986, pages 26 to 31). Experiments on dust suspension 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 PM₁₀ range (EPA 1988, pages 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 PM₁₀ from construction activities. Because the default emission rate was based on continuous emissions over 30 days, the daily PM₁₀ emission rate would be 0.0027 kilogram per square meter (0.012 ton per acre) per day, or 0.00011 kilogram per square meter (0.00050 ton per acre)

per hour. Dust suppression activities would reduce PM₁₀ emissions; however, the analysis took no credit for normal dust suppression activities.

The estimation of the annual and 24-hour average PM₁₀ emission rates required an estimate of the size of the area to be disturbed 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 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 construction activities would last. Table G-4 lists the total areas of disturbance at various repository operation areas. The analysis assumed that the entire land area required for excavated rock storage (for both the construction and operation phases) would be disturbed by excavated rock storage preparation activities, although only a portion of it would be used during the construction phase. The much larger volume of rock that DOE would remove during excavation for the low thermal load scenario would require that the excavated rock pile not be in the South Portal Operations Area. Rather, it would be about 5 kilometers (3 miles) east of the South Portal (TRW 1999b, pages 6-41 and 6-43). The excavated rock could be piled higher in this location [to about 15 meters (50 feet)] than in the South Portal Operations Area [where the piles could be no more than about 6 meters (20 feet) high], requiring less land area under this option and making the area required for all three thermal load scenarios about the same. Table G-5 lists fugitive dust emissions from surface construction; Table G-6 lists estimated air quality impacts from fugitive dust as the pollutant concentration in air and as the percent of the applicable regulatory limit.

Table G-4. Land area (square kilometers)^a disturbed during the construction phase for each thermal load scenario.^{b,c}

Operations area	High	Intermediate	Low
North Portal and roads	0.62	0.62	0.62
South Portal	0.15	0.15	0.15
Ventilation shafts	0.02	0.02	0.06
Total excavated rock storage	1.0	1.2	1.1
Rail construction on site ^d	0.6	0.6	0.6
Totals^b	2.4	2.6	2.6
Area disturbed per year	0.48	0.52	0.50

- a. To convert square kilometers to acres, multiply by 247.1.
- b. Numbers are rounded to two significant figures; therefore, totals might differ from sums of values.
- c. Source: Jessen (1998, all).
- d. Onsite rail line assumed to be 10 kilometers (6 miles) long and 0.06 kilometer (0.04 mile) wide.

Table G-5. Fugitive dust releases from surface construction (PM₁₀).^a

Thermal load scenario	Period	Pollutant emission (kilograms) ^b	Emission rate (grams per second ^c)
High	Annual	110,000 per year	3.4
	24-hour	430 per day	15 ^d
Intermediate	Annual	120,000 per year	3.6
	24-hour	460 per day	16 ^d
Low	Annual	120,000 per year	3.7
	24-hour	460 per day	16 ^d

- a. Numbers are rounded to two significant figures.
- b. To convert kilograms to pounds, multiply by 2.2046.
- c. To convert grams per second to pounds per hour, multiply by 7.9366.
- d. Based on an 8-hour release period.

Fugitive dust from construction would produce small offsite PM₁₀ concentrations. The annual and 24-hour average concentrations of PM₁₀ would be about 1 percent and about 2 percent, respectively, of the regulatory limit for all three thermal load scenarios. The differences between the thermal load

Table G-6. Estimated fugitive dust air quality impacts (micrograms per cubic meter) from surface construction (PM₁₀).

Thermal load scenario	Period	Maximum concentration ^a	Regulatory limit	Percent of limit ^a
High	Annual	0.41	50	0.83
	24-hour	2.9	150	1.9
Intermediate	Annual	0.44	50	0.88
	24-hour	3.0	150	2.0
Low	Annual	0.44	50	0.88
	24-hour	3.1	150	2.0

a. Numbers are rounded to two significant figures.

scenarios would be very small; the high thermal load would have the smallest impacts due mainly to the smaller area required for excavated rock storage.

For Modules 1 and 2, the same technique was used as for the Proposed Action, but the amount of land disturbed would be about 1.1, 1.1, and 1.3 times larger than for the Proposed Action for the high, intermediate, and low thermal load scenarios, respectively (Jessen 1998, all). The increase in disturbed land area would lead to estimated air quality impacts about 1.1, 1.1, and 1.3 times larger than the Proposed Action for the high, intermediate, and low thermal load scenarios, respectively.

G.1.4.2 Fugitive Dust from Subsurface Excavation

Fugitive dust would be released during the excavation of rock from the repository. Subsurface excavation activities would take place 250 days per year in three 8-hour shifts per day. Excavation would generate dust in the tunnels, and some of the dust would be emitted to the surface atmosphere through the ventilation system. DOE estimated the amount of dust that would be emitted by the ventilation system by using engineering judgment and best available information (DOE 1998, page 37). Table G-7 lists the release rates of PM₁₀ for excavation activities. Table G-8 lists estimated air quality impacts from fugitive dust as pollutant concentration in air and percentage of regulatory limit.

Table G-7. Fugitive dust releases from excavation activities (PM₁₀).^a

Period	Emission (kilograms) ^b	Emission rate (grams per second) ^c
Annual	920 per year	0.029
24-hour	3.7 per day	0.043 ^d

a. Numbers are rounded to two significant figures.

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

c. To convert grams per second to pounds per hour, multiply by 7.9366.

d. Based on a 24-hour release period.

Table G-8. Fugitive dust (PM₁₀) and cristobalite air quality impacts (micrograms per cubic meter) from excavation activities.

Period	Maximum concentration ^a	Regulatory limit	Percent of regulatory limit ^a
<i>PM₁₀</i>			
Annual	0.0035	50	0.0070
24-hour	0.044	150	0.029
<i>Cristobalite</i>			
Annual	0.0010	10 ^b	0.010

a. Numbers are rounded to two significant figures.

b. This value is a benchmark; there is no regulatory limit for cristobalite. See Section G.1.

Fugitive dust emissions from excavation operations would produce small offsite PM₁₀ concentrations. Both annual and 24-hour average concentrations of PM₁₀ would be much less than 1 percent of the regulatory standards. The highest estimated annual and 24-hour excavation rates, and hence the highest estimated fugitive dust concentrations, would be the same for all three thermal load scenarios.

Dust generated during excavation would contain cristobalite, a naturally occurring form of crystalline silica discussed in Section G.1. The analysis estimated the amount of cristobalite released by multiplying the amount of dust released annually (shown in Table G-7) by the percentage of cristobalite in the parent rock (28 percent). Table G-8 also lists the potential air quality impacts for releases of cristobalite from excavation of the repository. Because there are no public exposure limits for cristobalite, the annual average concentration was compared to a derived benchmark level for the prevention of silicosis, as discussed in Section G.1. The offsite cristobalite concentration would be about 0.01 percent of this benchmark.

The air quality impacts from fugitive dust emissions from excavation operations under the construction phase would be the same for Modules 1 and 2 as for the Proposed Action.

G.1.4.3 Fugitive Dust from Excavated Rock Pile

The disposal and storage of excavated rock on the surface excavated rock pile would generate fugitive dust. Dust would be released during the unloading of the excavated rock and subsequent smoothing of the excavated rock pile, as well as by wind erosion of the material. DOE used the total suspended particulate emission for active storage piles from a report by Cowherd, Muleski, and Kinsey (1988, pages 4-17 to 4-37) to estimate fugitive dust emission. The equation is:

$$E = 1.9 \times (s \div 1.5) \times [(365 - p) \div 235] \times (f \div 15)$$

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 or more of precipitation
 f = percentage of time wind speed exceeds 5.4 meters per second (12 miles per hour) at pile height

For this analysis, s is equal to 4 percent [no value was available for this variable, so the average silt content of limestone quarrying material (EPA 1995b, page 13.2.4-2) was used], p is 37.75 (Fransioli 1999, all) and f is 16.5 (calculated from meteorological data used in the Industrial Source Complex model). Thus, E is equal to 7.8 kilograms of total particulates per day per hectare (6.9 pounds per day per acre). Only about 50 percent of the total particulates would be PM₁₀ (Cowherd, Muleski, and Kinsey 1988, pages 4-17 to 4-37); therefore, the emission rate for PM₁₀ would be 3.9 kilograms per day per hectare (3.5 pounds per day per acre).

The analysis estimated fugitive dust from disposal and storage using the size of the area actively involved in storage and maintenance. Only a portion of the excavated rock pile would be actively disturbed by the unloading of excavated rock and the subsequent contouring of the pile, and only that portion would be an active source of fugitive dust. The analysis assumed that the rest of the excavated rock pile would be stabilized by either natural processes or DOE stabilization measures and would release small amounts of dust.

DOE based its estimate of the size of the active portion of the excavated rock pile on the amount of material it would store there each year. The volume of rock placed on the excavated rock pile from excavation activities during the construction phase (TRW 1999b, page 6-7) was divided by the height of the storage pile. The average height of the excavated rock pile would be about 6 meters (20 feet) for the

high and intermediate thermal load scenarios (TRW 1999b, page 6-42) and 15 meters (50 feet) for the low thermal load scenario (TRW 1999b, page 6-43). Table G-9 lists the areas of the excavated rock pile and the active portion for each thermal load scenario. The active area of the excavated rock pile was estimated using the total area of the rock pile at the end of the construction phase divided by the number of years of construction multiplied by 2 (Smith 1999, all). As noted in Section G.1.4.1, under the low thermal load scenario the excavated rock pile would be several kilometers east of the South Portal Operations Area. Under this option the pile could be higher in this location, allowing for a smaller area of disturbance than for the excavated rock piles of the high and intermediate thermal load scenarios in the South Portal Operations Area.

Table G-9. Active area (square kilometers)^a of excavated rock pile during the construction phase.^{b,c}

Thermal load	Area	Number of years	Average annual active area
High	0.34	5	0.14
Intermediate	0.41	5	0.17
Low	0.17	5	0.066

- To convert square kilometers to square miles, multiply by 0.3861.
- Numbers are rounded to two significant figures.
- The construction phase would last 5 years. Subsurface excavation and rock pile activities would continue during the operation and monitoring phase (see Section G.1.5).

Table G-10 lists the fugitive dust release rate from disposal and storage of the excavated rock pile by thermal load scenario. Table G-11 lists the air quality impacts from fugitive dust as pollutant concentration and percent of regulatory limit.

Table G-10. Fugitive dust released from the excavated rock pile during the construction phase (PM₁₀).^a

Thermal load	Period	Emission (kilograms) ^b	Emission rate (grams per second) ^c
High	Annual	19,000 per year	0.61
	24-hour	53 per day	0.61 ^d
Intermediate	Annual	23,000 per year	0.74
	24-hour	64 per day	0.74 ^d
Low	Annual	9,400 per year	0.30
	24-hour	26 per day	0.30 ^d

- Numbers are rounded to two significant figures.
- To convert kilograms to pounds, multiply by 2.2046.
- To convert grams per second to pounds per hour, multiply by 7.9366.
- Based on a continuous release.

Fugitive dust emissions from the excavated rock pile during the construction phase would produce small offsite PM₁₀ concentrations. Both the annual and 24-hour average concentrations of PM₁₀ would be less than 1 percent of the regulatory standards. The low thermal load scenario would have the smallest concentrations due to the smaller area of active disturbance, which is directly related to the taller pile with a resultant smaller surface-area-to-volume ratio.

Table G-11 also lists potential air quality impacts for releases of cristobalite. The methods used were the same as those described in Section G.1.4.2 for the construction phase, where cristobalite was assumed to be 28 percent of the fugitive dust released, based on its percentage in parent rock. The land withdrawal area boundary cristobalite concentration would be small, about 0.25 percent or less of the benchmark level discussed in Section G.1.

Table G-11. Fugitive dust (PM₁₀) and cristobalite air quality impacts (micrograms per cubic meter) from the excavated rock pile during the construction phase.

Thermal load	Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
<i>PM₁₀</i>				
High	Annual	0.074	50	0.15
	24-hour	0.62	150	0.41
Intermediate	Annual	0.090	50	0.18
	24-hour	0.76	150	0.51
Low	Annual	0.036	50	0.071
	24-hour	0.30	150	0.19
<i>Cristobalite</i>				
High	Annual	0.021	10 ^c	0.21
Intermediate	Annual	0.025	10 ^c	0.25
Low	Annual	0.010	10 ^c	0.010

a. Numbers are rounded to two significant figures.

b. Sources: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

c. This value is a benchmark; there are no regulatory limits for cristobalite other than worker exposure limits. See Section G.1.

For Modules 1 and 2, the volume of rock excavated during the construction phase would be nearly 1.8 million cubic meters (2.3 million cubic yards) for all three thermal load scenarios (TRW 1999b, pages 6-7 and 6-53). This represents an increase of about 16 percent over the Proposed Action for the high thermal load scenario, and a slight decrease of about 5 percent for the intermediate and low thermal load scenarios. The estimated air quality impacts would change proportionately from Proposed Action impacts, increasing 16 percent for the high thermal load scenario and decreasing by 5 percent for the intermediate and low thermal load scenarios.

G.1.4.4 Fugitive Dust from Concrete Batch Facility

The concrete batch facility for the fabrication and curing of tunnel inverters and tunnel liners would emit dust. This facility would run 3 hours a day and would produce 115 cubic meters (150 cubic yards) of concrete per hour of operation (TRW 1999b, pages 4-4 and 4-5). It would operate 250 days per year. Table G-12 lists emission factor estimates for the concrete batch facility (EPA 1995b, pages 11.12-1 to 11.12-5). About 0.76 cubic meter (1 cubic yard) of typical concrete weighs 1,800 kilograms (4,000 pounds) (EPA 1995b, page 11.12-3). The size of the aggregate storage pile for the concrete batch facility would be 800 square meters (0.2 acre) (TRW 1999b, pages 4-4 and 4-5).

Table G-12. Dust release rates for the concrete batch facility (kilograms per 1,000 kilograms of concrete).^{a,b}

Source/activity	Emission rate
Sand and aggregate transfer to elevated bin	0.014
Cement unloading to elevated storage silo	0.13
Weight hopper loading	0.01
Mixer loading	0.02
Wind erosion from aggregate storage	3.9 kilograms per hectare ^c per day

a. Source: EPA (1995b, page 11.12-3).

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

c. 3.9 kilograms per hectare = about 21 pounds per acre.

Table G-13 lists the dust release rates of the concrete batch facility. The releases would be the same for all thermal load scenarios. Table G-14 lists estimated potential air quality impacts as the estimated pollutant concentration and percent of regulatory limit.

Table G-13. Dust release rates for the concrete batch facility during the operation and monitoring phase (PM₁₀).^a

Period	Emission (kilograms) ^b	Emission rate (grams per second) ^c
Annual	36,000 per year	1.1
24-hour	140 per day	13 ^d

- a. Numbers are rounded to two significant figures.
- b. To convert kilograms to pounds, multiply by 2.2046.
- c. To convert grams per second to pounds per hour, multiply by 7.9366.
- d. Based on a 3-hour release.

Table G-14. Particulate matter (PM₁₀) air quality impacts (micrograms per cubic meter) from the concrete batch facility during the construction phase.

Period	Maximum concentration ^a	Regulatory limit	Percent of regulatory limit ^a
Annual	0.14	50	0.27
24-hour	2.2	150	1.1

- a. Numbers are rounded to two significant figures.

Dust emissions from the concrete batch facility during the operation and monitoring phase would produce small offsite PM₁₀ concentrations. The annual and 24-hour averaged concentrations of PM₁₀ would be less than 1 percent and about 1.5 percent of the regulatory standards, respectively.

For Modules 1 and 2, the air quality impacts from the concrete batch facility during the construction phase would be the same as for the Proposed Action.

G.1.4.5 Exhaust Emissions from Construction Equipment

Diesel- and gasoline-powered equipment would emit all four criteria pollutants during the construction phase. EPA (1991, pages II-7-1 to II-7-7) provided pollutant emission rate estimates for heavy-duty equipment. This analysis assumed construction equipment would emit the average of the EPA reference emission rates. Table G-15 lists the emission rates for this equipment.

Table G-15. Pollutant emission rates (kilograms^a per 1,000 liters^b of fuel) for construction equipment.^c

Pollutant	Estimated emission	
	Diesel	Gasoline
Carbon monoxide	15	450
Nitrogen dioxide	39	13
PM ₁₀	3.5	0.86
Sulfur dioxide	3.7	0.63

- a. To convert kilograms to pounds, multiply by 2.2046.
- b. To convert liters to gallons, multiply by 0.26418.
- c. Source: Average of rates from EPA (1991, pages II-7-1 to II-7-7).

Table G-16 lists the estimated average amount of fuel per year for the construction of the North and South Portal Operations Areas. The fuel for the South Portal Operations Area would include fuel consumed during maintenance of the excavated rock pile.

Table G-16. Amount of fuel consumed per year during the construction phase (liters).^{a,b}

Thermal load	South Portal Operations Area ^c		North Portal Operations Area ^d
	Diesel	Gasoline	Diesel
High	360,000	20,000	640,000
Intermediate	360,000	20,000	640,000
Low	560,000	20,000	640,000

- a. To convert liters to gallons, multiply by 0.26418.
b. Numbers are rounded to two significant figures.
c. Source: Based on total fuel use from TRW (1999b, page 6-3).
d. Source: Based on total fuel use from TRW (1999a, Table 6.1, page 71).

Table G-17 lists pollutant releases from construction equipment for each thermal load scenario. The emission rate for the annual concentration was calculated from the total fuel consumed, assuming the same amount of fuel would be consumed each year.

Table G-17. Pollutant release rates from surface equipment during the construction phase.^a

Pollutant	Period	Mass of pollutant per averaging period (kilograms) ^b		Emission rate ^c (grams per second) ^d	
		South	North	South	North
<i>High and intermediate thermal load</i>					
Nitrogen dioxide	Annual	14,000	25,000	0.46	0.80
Sulfur dioxide	Annual	1,400	2,400	0.043	0.076
	24-hour	5.4	9.6	0.019	0.33
	3-hour	2.0	3.6	0.019	0.33
Carbon monoxide	8-hour	57	39	2.0	1.3
	1-hour	7.2	4.8	2.0	1.3
PM ₁₀	Annual	1,300	2,200	0.040	0.071
	24-hour	5.1	8.9	0.18	0.31
<i>Low thermal load</i>					
Nitrogen dioxide	Annual	22,000	25,000	0.71	0.80
Sulfur dioxide	Annual	2,100	2,400	0.067	0.076
	24-hour	8.4	9.6	0.29	0.33
	3-hour	3.2	3.6	0.29	0.33
Carbon monoxide	8-hour	69	39	2.4	1.3
	1-hour	8.7	4.8	2.4	1.3
PM ₁₀	Annual	2,000	2,200	0.062	0.071
	24-hour	7.9	8.9	0.27	0.31

- a. Numbers are rounded to two significant figures.
b. To convert kilograms to pounds, multiply by 2.2046.
c. Based on an 8-hour release for averaging periods 24 hours or less.
d. To convert grams per second to pounds per hour, multiply by 7.9366.

Table G-18 lists the impacts on air quality from construction equipment emission by thermal load scenario as the pollutant concentration in air and the percent of the regulatory limit. Emissions from surface equipment during the construction phase would produce small offsite (outside the land withdrawal area) criteria pollutant concentrations. All concentrations would be less than 1 percent of the regulatory standards.

For Modules 1 and 2, the same analysis method was used as that for the Proposed Action, but the amount of fuel used in the South Portal Operations Area would vary from the Proposed Action. Diesel fuel use would be about 7.4 times larger for the high and intermediate thermal load scenarios and about 4.8 times larger for the low thermal load scenario. Gasoline use would be two times larger for all thermal load scenarios (TRW 1999b, page 6-45). There would be no change in the amount of fuel used during the

Table G-18. Air quality impacts from construction equipment during the construction phase (micrograms per cubic meter).^a

Pollutant	Period	Maximum concentration	Regulatory limit ^b	Percent of regulatory limit
<i>High and intermediate thermal load</i>				
Nitrogen dioxide	Annual	0.13	100	0.13
Sulfur dioxide	Annual	0.013	80	0.016
	24-hour	0.096	365	0.026
	3-hour	0.77	1,300	0.059
Carbon monoxide	8-hour	1.8	10,000	0.018
	1-hour	11	40,000	0.028
PM ₁₀	Annual	0.012	50	0.024
	24-hour	0.090	150	0.060
<i>Low thermal load</i>				
Nitrogen dioxide	Annual	0.16	100	0.16
Sulfur dioxide	Annual	0.016	80	0.020
	24-hour	0.12	365	0.032
	3-hour	0.93	1,300	0.071
Carbon monoxide	8-hour	2.1	10,000	0.020
	1-hour	12	40,000	0.031
PM ₁₀	Annual	0.014	50	0.029
	24-hour	0.11	150	0.072

a. Numbers are rounded to two significant figures.

b. Source: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

construction of the North Portal. These increases in fuel use would lead to estimated air quality impacts that would be about 3.5 times larger for the high and intermediate thermal load scenarios and about 2.5 times larger for the low thermal load scenario except for carbon monoxide. Carbon monoxide air quality impacts, which are more heavily weighted towards gasoline, would be about 2.5, 2.5 and 2.0 times larger than the Proposed Action for the high, intermediate, and low thermal load scenarios, respectively.

G.1.4.6 Exhaust from Boiler

A proposed boiler in the South Portal Operations Area would emit the four criteria pollutants. The boiler would use diesel fuel and provide steam and hot water for the heating, ventilation, and air conditioning system. The analysis assumed that this boiler would be the same size as the boiler that would operate in the North Portal Operations Area during the operation and monitoring phase (TRW 1999a, Table 6-2, page 75) but not during construction. Table G-19 lists the annual emission rates of the boiler in the South Portal Operations Area. To estimate the short-term (24 hours or less) emission rate, the analysis assumed the boiler would run 250 days (6,000 hours) per year. Given the annual boiler emissions, this was a conservative assumption because continuous operation 365 days (8,760 hours) per year would result in lower daily emissions. This assumption considered periods when the boiler would not be operating. The actual period of boiler operation is not known. In addition, specific information on the boiler stack height and exhaust air temperature (which would affect plume rise) has not been developed. The analysis assumed that releases would be from ground level, which overestimates actual concentrations. Table G-20 lists releases of criteria pollutants by the boiler. Table G-21 lists estimated potential air quality impacts as pollutant concentrations in air and percent of regulatory limit.

Table G-19. Annual pollutant release rates (kilograms per year)^a for the South Portal Operations Area boiler.^{b,c}

Pollutant	Annual emission rate
Nitrogen dioxide	58,000
Sulfur dioxide	20,000
Carbon monoxide	15,000
PM ₁₀	5,600

a. To convert kilograms to tons, multiply by 0.0011023.

b. Source: TRW (1999a, Table 6-2, page 75).

c. Numbers are rounded to two significant figures.

Table G-20. Pollutant release rates from the boiler during the construction phase.^a

Pollutant	Period	Mass of pollutant (kilograms) ^b per averaging time	Emission rate ^c (grams per second) ^d
Nitrogen dioxide	Annual	58,000	1.83
Sulfur dioxide	Annual	20,000	0.63
	24-hour	80	0.92
	3-hour	10	0.92
Carbon monoxide	8-hour	20	0.67
	1-hour	2.5	0.67
PM ₁₀	Annual	5,600	0.18
	24-hour	22	0.25

- a. Numbers are rounded to two significant figures.
b. To convert kilograms to pounds, multiply by 2.2046.
c. Based on an 8-hour release for averaging periods of 24 hours or less.
d. To convert grams per second to pounds per hour, multiply by 7.9366.

Table G-21. Air quality impacts from boiler pollutant releases from the South Portal Operations Area during the construction phase (micrograms per cubic meter of pollutant).

Pollutant	Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
Nitrogen dioxide	Annual	0.22	100	0.22
Sulfur dioxide	Annual	0.076	80	0.095
	24-hour	0.94	365	0.26
	3-hour	5.5	1,300	0.43
Carbon monoxide	8-hour	2.0	10,000	0.020
	1-hour	12	40,000	0.031
PM ₁₀	Annual	0.022	50	0.044
	24-hour	0.27	150	0.18

- a. Numbers are rounded to two significant figures.
b. Sources: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

Emissions from the boiler during the construction phase would produce small offsite (outside the land withdrawal area) criteria pollutant concentrations. All concentrations would be less than 1 percent of the regulatory standards.

For Modules 1 and 2, the air quality impacts from the boiler during the construction phase would be the same as those for the Proposed Action.

G.1.5 OPERATION AND MONITORING PHASE

This section describes the method DOE used to estimate air quality impacts during the operation and monitoring phase (2010 to 2110). Activities during this phase would include the continued development of the subsurface facilities, which would last 22 years for all thermal load scenarios. Emplacement activities in the surface and subsurface facilities would continue concurrently with development operations for 24 years; 76 years of monitoring and maintenance would begin after the end of emplacement operations. The duration of the monitoring and maintenance period has not been finalized, but could be as long as 276 years for a 300-year operation and monitoring phase. For purposes of analysis, workers would use the following schedule for activities during the operation and monitoring phase: three 8-hour shifts a day, 5 days a week, 50 weeks a year; the maintenance of the excavated rock pile would occur in one 8-hour shift a day, 5 days a week, 50 weeks a year.

For Modules 1 and 2, the continued development of the subsurface facilities would last 36 years for all thermal load scenarios. Emplacement activities in the surface and subsurface facilities would continue concurrently with development operations for 38 years. The duration of the monitoring and maintenance period has not been finalized, but could be as long as 262 years for a 300-year operation and monitoring phase.

The analysis estimated air quality impacts by calculating pollutant concentrations from various operation and monitoring activities. Emission rates were developed for each activity that would result in pollutant releases. The emission rates were multiplied by the unit release concentrations (see Section G.1.3) to calculate the pollutant concentration for comparison to the various regulatory limits.

The principal emission sources of particulates would be dust emissions from concrete batch facility operations and fugitive dust emissions from excavation and storage on the excavated rock pile. Fuel combustion from maintenance of the excavated rock pile and emissions from the North Portal and South Portal boilers would be principal sources of nitrogen dioxide, sulfur dioxide, and carbon monoxide. The following sections describe these sources in more detail.

G.1.5.1 Fugitive Dust from Concrete Batch Facility

The concrete batch facility for the fabrication and curing of tunnel inverters and liners would emit dust. The analysis assumed that the dust emissions from the concrete batch facility would be the same as those during the construction phase. Thus, the dust release rate and potential air quality impacts would be the same as those listed in Tables G-13 and G-14.

G.1.5.2 Fugitive Dust from Subsurface Excavation

The excavation of rock from the repository would generate fugitive dust in the drifts. Some of the dust would reach the external atmosphere through the repository ventilation system. Fugitive dust emission rates from excavation during operations would be the same as those during the construction phase. Thus, the fugitive dust release rate and potential air quality impacts for excavation of rock would be the same as those listed in Tables G-7 and G-8. Air quality impacts from cristobalite released during excavation of the repository would be the same as those listed in Table G-8.

G.1.5.3 Fugitive Dust from Excavated Rock Pile

The disposal and storage of excavated rock on the excavated rock pile would release fugitive dust. The analysis used the same method to estimate fugitive dust releases from the excavated rock pile during operations that it used for the construction phase (See Section G.1.4.3). Table G-22 lists the areas of the active portion of the excavated rock pile by thermal load scenario. The total land area used for storage and the active portion of the excavated rock pile was based on the amount of rock that would be stored during operations (TRW 1999b, page 6-17). Sections G.1.4.1 and G.1.4.3 compare the excavated rock pile areas for the three thermal load scenarios.

Table G-22. Estimated active excavated rock pile area (square kilometers)^a during subsurface excavation activities during the operation and monitoring phase.^b

Thermal load	Storage area	Years of repository development	Annual average active area
High	0.63	22	0.058
Intermediate	0.76	22	0.069
Low	1.0	22	0.095

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

b. Numbers are rounded to two significant figures.

While the land area used for storage of excavated rock during the operation and monitoring phase would be nearly twice as large as that used during the construction phase for the high and intermediate thermal load scenarios, the active area per year would be about half of that for construction due to the larger number of years over which storage would occur (22 years compared to 5 years). The land area used during the operation and monitoring phase for the low thermal load scenario would be nearly 10 times that used during the construction phase. The annual active area would be larger during the operation and monitoring phase than during the construction phase, but only about twice as large because of the longer period over which storage would take place (22 years compared to 5 years). Table G-23 lists fugitive dust releases from the excavated rock pile; Table G-24 lists potential air quality impacts as the pollutant concentration and percent of the regulatory limit.

Table G-23. Fugitive dust release rate from the excavated rock pile during the operation and monitoring phase (PM₁₀).^a

Thermal load	Period	Emissions (kilograms) ^b	Emission rate ^c (grams per second) ^d
High	Annual	8,200 per year	0.26
	24-hour	22 per day	0.26
Intermediate	Annual	9,800 per year	0.31
	24-hour	27 per day	0.31
Low	Annual	13,000 per year	0.42
	24-hour	37 per day	0.42

- Numbers are rounded to two significant figures.
- To convert kilograms to pounds, multiply by 2.2046.
- Based on a continuous release.
- To convert grams per second to pounds per hour, multiply by 7.9366.

Table G-24. Fugitive dust (PM₁₀) and cristobalite air quality impacts from the excavated rock pile during the operation and monitoring phase (micrograms per cubic meter).

Thermal load	Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
<i>PM₁₀</i>				
High	Annual	0.031	50	0.062
	24-hour	0.27	150	0.18
Intermediate	Annual	0.038	50	0.075
	24-hour	0.32	150	0.21
Low	Annual	0.051	50	0.10
	24-hour	0.43	150	0.29
<i>Cristobalite</i>				
High	Annual	0.0087	10 ^c	0.087
Intermediate	Annual	0.011	10 ^c	0.11
Low	Annual	0.014	10 ^c	0.14

- Numbers are rounded to two significant figures.
- Source: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.
- This value is a benchmark; there is no regulatory limit for cristobalite. See Section G.1.

Fugitive dust emissions from the excavated rock pile during the operation and monitoring phase would produce very small offsite (outside the land withdrawal area) PM₁₀ concentrations. Both annual and 24-hour average concentrations of PM₁₀ would be less than 1 percent of the regulatory standards for all three thermal load scenarios.

Table G-24 also lists potential air quality impacts for releases of cristobalite. The methods used were the same as those described in Section G.1.4.2 for the construction phase, where cristobalite was assumed to be 28 percent of the fugitive dust released, based on its percentage in parent rock. The site boundary

cristobalite concentration would be small, about 0.1 percent of the benchmark level discussed in Section G.1.

The Module 1 and 2 analysis used the same technique as for the Proposed Action, but the estimated active excavated rock pile area would be about 1.4, 1.2, and 1.1 times larger than the Proposed Action for the high, intermediate, and low thermal load scenarios, respectively, based on the volumes of rock added annually to the pile (TRW 1999b, page 6-56). The estimated air quality impacts from the excavated rock pile would also be 1.4, 1.2, and 1.1 times larger than the Proposed Action for the high, intermediate, and low thermal load scenarios, respectively.

G.1.5.4 Exhaust from Excavated Rock Pile Maintenance Equipment

Surface equipment would emit the four criteria pollutants during excavated rock pile maintenance. The analysis used the same method to determine air quality impacts for surface equipment during operations that it used for the construction phase (see Section G.1.4.5). Table G-15 lists the pollutant release rates of the equipment. Table G-25 lists the average amount of fuel consumed each year during the operation and monitoring phase at the South Portal Operations Area.

Table G-25. Annual amount of fuel (liters)^a consumed during the operation and monitoring phase.^{b,c}

Thermal load	Diesel	Gasoline
High	350,000	4,500
Intermediate	350,000	4,500
Low	2,800,000	9,000

- a. To convert liters to gallons, multiply by 0.26418.
- b. Source: Based on total fuel use from TRW (1999b, pages 6-14 and 6-21).
- c. Numbers are rounded to two significant figures.

Table G-26 lists pollutant release rates for surface equipment during operations activities of the operation and monitoring phase. Monitoring activity emissions would be much smaller. Table G-27 lists potential air quality impacts.

Table G-26. Pollutant release rates from surface equipment during the operation and monitoring phase.^a

Pollutant	Period	Mass of pollutant per averaging time (kilograms) ^b	Emission rate ^c (grams per second) ^d
<i>High and intermediate thermal load</i>			
Nitrogen dioxide	Annual	14,000	0.44
Sulfur dioxide	Annual	1,300	0.041
	24-hour	5.2	0.18
	3-hour	4.9	0.18
Carbon monoxide	8-hour	29	1.0
	1-hour	3.6	1.0
	Annual	1,200	0.039
PM ₁₀	24-hour	4.9	0.17
<i>Low thermal load</i>			
Nitrogen dioxide	Annual	110,000	3.5
Sulfur dioxide	Annual	10,000	0.33
	24-hour	42	1.4
	3-hour	16	1.4
Carbon monoxide	8-hour	180	6.4
	1-hour	23	6.4
	Annual	9,700	0.31
PM ₁₀	24-hour	39	1.4

- a. Numbers are rounded to two significant figures.
- b. To convert kilograms to pounds, multiply by 2.2046.
- c. Based on an 8-hour release for averaging periods of 24 hours or less.
- d. To convert grams per second to pounds per hour, multiply by 7.9366.

Table G-27. Air quality impacts from surface equipment during the operation and monitoring phase (micrograms per cubic meter of pollutant).

Pollutant	Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
<i>High and intermediate thermal load</i>				
Nitrogen dioxide	Annual	0.052	100	0.052
Sulfur dioxide	Annual	0.0049	80	0.0063
	24-hour	0.034	365	0.0094
	3-hour	0.27	1,300	0.021
Carbon monoxide	8-hour	0.58	10,000	0.0056
	1-hour	3.3	40,000	0.0084
PM ₁₀	Annual	0.0046	50	0.0092
	24-hour	0.032	150	0.021
<i>Low thermal load</i>				
Nitrogen dioxide	Annual	0.42	100	0.42
Sulfur dioxide	Annual	0.040	80	0.051
	24-hour	0.28	365	0.076
	3-hour	2.2	1,300	0.17
Carbon monoxide	8-hour	3.7	10,000	0.036
	1-hour	21	40,000	0.053
PM ₁₀	Annual	0.037	50	0.074
	24-hour	0.26	150	0.17

a. Numbers are rounded to two significant figures.

b. Sources: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

Emissions from surface equipment during operation and monitoring would produce very small concentrations of offsite (outside the land withdrawal area) criteria pollutants. All estimated concentrations would be less than 1 percent of the regulatory standards.

The Module 1 and 2 analysis used the same technique as for the Proposed Action, but the amount of fuel used during the operation and monitoring phase would increase. Annual diesel fuel use during development would increase by 1.6, 3.0, and 2.0 times the Proposed Action; annual gasoline use would increase by 1.2, 1.8, and 1.5 times the Proposed Action for the high, intermediate, and low thermal load scenarios, respectively, based on total fuel use (TRW 1999b, page 6-53). Annual diesel fuel use during emplacement would increase only by about 1 percent over the Proposed Action for all thermal load scenarios (TRW 1999b, page 6-61). Estimated air quality impacts for surface equipment during the operation and monitoring phase under Module 1 and 2 would increase by about 1.6, 3.0, and 2.0 times the Proposed Action for the high, intermediate, and low thermal load scenarios.

G.1.5.5 Exhaust from Boiler

Boilers in the North and South Portal Operations Areas would emit the four criteria pollutants. The annual emission rates of the boiler in the North Portal Operations Area would be the same as those listed in Table G-19 (the boilers were assumed to be the same size). There would be small variations in the North Portal boiler emissions for the transportation and waste packaging options because of different operational requirements. The emissions listed in Table G-19 are for the combination of legal-weight truck transport and uncanistered waste scenario, which would require the largest boiler because a larger Waste Handling Building would be required (TRW 1999a, pages 66 to 75). Other options would require a slightly smaller boiler (TRW 1999a, Table 6-2, page 75) and the release rate of pollutants would be about 15 percent smaller. The size of the boiler would not depend on the thermal load scenario. The analysis assumed the boiler would run 250 days (6,000 hours) per year. Given an annual emission rate, this was a conservative assumption because continuous operation 365 days (8,760 hours) per year would result in lower daily emissions. This assumption considered periods when the boiler would not be operating. The actual period of boiler operation is not known. Rates from the North Portal boiler for

evaluating pollutant releases during the operation and monitoring phase would be the same as those listed in Table G-20 for the South Portal boiler.

Table G-28 lists estimated potential air quality impacts as pollutant concentrations in air and percent of regulatory limit. These impacts would be due to emissions from the boilers in the North and South Portal Operations Areas. Although total emissions during the operation and monitoring phase would be double those during the construction phase (when only the South Portal boiler would operate), air quality impacts would not double because of different atmospheric dispersion factors from the two operations areas to the location of the hypothetically maximally exposed individual. Emissions from the two boilers during the operation and monitoring phase would produce small offsite criteria pollutant concentrations. All concentrations would be less than 1 percent of the regulatory standards.

Table G-28. Air quality impacts from boiler pollutant releases from both North and South Portal Operations Areas (micrograms per cubic meter of pollutant).

Pollutant	Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
Nitrogen dioxide	Annual	0.40	100	0.40
Sulfur dioxide	Annual	0.14	80	0.18
	24-hour	1.8	365	0.49
	3-hour	11	1,300	0.85
Carbon monoxide	8-hour	3.7	10,000	0.037
	1-hour	24	40,000	0.061
PM ₁₀	Annual	0.039	50	0.078
	24-hour	0.51	150	0.34

a. Numbers are rounded to two significant figures.

b. Sources: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

For Module 1 or 2, the estimated air quality impacts from boilers during the operation and monitoring phase would be the same as those for the Proposed Action.

G.1.6 CLOSURE PHASE

This section describes the method used to estimate air quality impacts during the closure phase at the proposed repository. The closure phase would last 6, 6, or 15 years for the high, intermediate, or low thermal load scenario, respectively. For Modules 1 and 2, the closure phase would last 13, 17, and 27 years for the high, intermediate, and low thermal load scenarios, respectively. The work schedule would be one 8-hour shift per day, 5 days a week, 50 weeks a year.

The analysis estimated air quality impacts by calculating pollutant concentrations from various closure activities. Emission rates were developed for each activity that would result in releases of pollutants. These pollutant emission rates were then multiplied by the unit release concentration (see Section G.1.3) to calculate the pollutant concentration for comparison to the various regulatory limits.

The sources of particulates would be emissions from the backfill plant and the concrete batch facility and fugitive dust from closure activities on the surface and 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.

G.1.6.1 Dust from Backfill Plant

The Closure Backfill Preparation Plant would process (separate, crush, screen, and wash) rock from the excavated rock pile for use as backfill for the underground access openings (TRW 1999b, pages 4-77 and 4-78). The facility would have the capacity to handle 91 metric tons (100 tons) an hour (TRW 1999b,

pages 4-77 and 4-78). For purposes of analysis, the backfill plant would run 6 hours a shift, 2 shifts a day, 5 days a week, 50 weeks a year.

The plant was assumed to have emissions similar to a crushed-stone processing plant. Table G-29 lists the emission rates for various activities associated with a crushed stone processing plant (EPA 1995b, pages 11.19.2-1 to 11.19.2-8). Table G-30 lists estimated pollutant release rates for the backfill plant. Table G-31 lists potential air quality impacts as pollutant concentrations in air and percent of regulatory limit.

Table G-29. Emission rates from a crushed stone processing plant.^{a,b}

Source/activity	Emission rate (kilogram ^c per 1,000 kilograms of material processed)
Dump to conveyor or truck	0.00005
Screening	0.0076
Crusher	0.0012
Fine screening	0.036

a. Source: EPA (1995b, pages 11.19.2-1 to 11.19.2-8).

b. Numbers are rounded to two significant figures.

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

Table G-30. Dust release rates from the backfill plant (PM₁₀).^a

Period	Emission (kilograms) ^b	Emission rate (grams per second) ^c
Annual	12,000 per year	0.39
24-hour	49 per day	1.1 ^d

a. Numbers are rounded to two significant figures.

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

c. To convert grams per second to pounds per hour, multiply by 7.9366.

d. Based on a 12-hour release period.

Table G-31. Particulate matter (PM₁₀) air quality impacts from backfill plant (micrograms per cubic meter).

Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
Annual	0.047	50	0.093
24-hour	1.1	150	0.71

a. Numbers are rounded to two significant figures.

b. Sources: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

Dust emissions from the backfill plant would produce small PM₁₀ concentrations. Both annual and 24-hour average concentrations of PM₁₀ would be less than 1 percent of the regulatory standards for all thermal load scenarios.

For Modules 1 and 2, the estimated air quality impacts for the backfill plant would be the same as those for the Proposed Action.

G.1.6.2 Fugitive Dust from Concrete Batch Facility

A concrete batch facility for the fabrication of seals would be similar to the facility that would operate during the construction and operation and monitoring phases (see Sections G.1.4.4 and G.1.5.1). The only difference would be that it would run only ten 3-hour shifts a year per concrete seal (TRW 1999b, page 4-78). The analysis assumed that two seals per year would be produced. Table G-12 lists activities associated with the concrete batch facility and their emissions. Table G-32 lists emissions from the concrete batch facility during closure. Table G-33 lists potential air quality impacts as pollutant concentration in air and percent of regulatory limit.

Table G-32. Dust release rates from the concrete batch facility during the closure phase (PM₁₀).^a

Period	Mass of pollutant (kilograms) ^b	Emission rate (grams per second) ^c
Annual	2,800 per year	0.090
24-hour	140 per day	13 ^d

- Numbers are rounded to two significant figures.
- To convert kilograms to pounds, multiply by 2.2046.
- To convert grams per second to pounds per hour, multiply by 7.9366.
- Based on a 3-hour release period.

Table G-33. Particulate matter (PM₁₀) air quality impacts from the concrete batch facility during the closure phase (micrograms per cubic meter).

Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
Annual	0.011	50	0.022
24-hour	2.2	150	1.5

- Numbers are rounded to two significant figures.
- Source: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

Dust emissions from the concrete batch facility during closure would produce small offsite (outside the land withdrawal area) PM₁₀ concentrations. The annual and 24-hour average concentrations of PM₁₀ would be less than 1 percent and around 1.5 percent, respectively, of the regulatory standards.

For Modules 1 and 2, the estimated air quality impacts from the concrete batch facility during the closure phase would be the same as those for the Proposed Action.

G.1.6.3 Fugitive Dust from Closure Activities

Closure activities such as smoothing and reshaping the excavated rock pile and demolishing buildings would produce the same fugitive dust releases as construction activities because they would disturb nearly the same amount of land. Thus, the pollutant release and air quality impacts from fugitive dust emissions from surface closure activities would be the same as those listed in Tables G-5 and G-6, respectively.

G.1.6.4 Fugitive Dust from Excavated Rock Pile

During backfill operations, fugitive dust would occur from the removal of excavated rock from the storage pile. The analysis used the same method to estimate fugitive dust emission from the excavated rock pile during the closure phase that it used for the construction phase (Section G.1.4.3). Table G-34 lists the total area of the excavated rock pile disturbed and the active portion, based on the amount of material to be removed from the pile (TRW 1999b, page 6-39). The analysis assumed that the rock used

Table G-34. Active excavated rock pile area (square kilometers)^a during the closure phase.^b

Thermal load	Total area disturbed for backfill operation	Number of years of closure	Active area (per year)
High	0.21	6	0.069
Intermediate	0.27	6	0.091
Low	0.26	15	0.035

- To convert square kilometers to acres, multiply by 247.1.
- Numbers are rounded to two significant figures.

in backfill would be from a limited area of the excavated rock pile, rather than from all over the pile. Table G-35 lists fugitive dust releases from the excavated rock pile. Table G-36 lists potential air quality impacts from the pile as pollutant air concentration and percent of regulatory limit.

Table G-35. Fugitive dust release rates from the excavated rock pile during the closure phase (PM₁₀).^a

Thermal load	Period	Emission (kilograms) ^b	Emission rate ^c (grams per second) ^d
High	Annual	9,800 per year	0.31
	24-hour	27 per day	0.31
Intermediate	Annual	13,000 per year	0.41
	24-hour	35 per day	0.41
Low	Annual	5,000 per year	0.16
	24-hour	14 per day	0.16

a. Numbers are rounded to two significant figures.

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

c. Based on a continuous release.

d. To convert grams per second to pounds per hour, multiply by 7.9366.

Table G-36. Fugitive dust (PM₁₀) and cristobalite air quality impacts from the excavated rock pile during the closure phase (micrograms per cubic meter).

Thermal load	Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
<i>PM₁₀</i>				
High	Annual	0.037	50	0.074
	24-hour	0.32	150	0.21
Intermediate	Annual	0.049	50	0.098
	24-hour	0.42	150	0.28
Low	Annual	0.019	50	0.038
	24-hour	0.16	150	0.11
<i>Cristobalite</i>				
High	Annual	0.010	10 ^c	0.10
Intermediate	Annual	0.014	10 ^c	0.14
Low	Annual	0.0053	10 ^c	0.053

a. Numbers are rounded to two significant figures.

b. Source: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

c. This value is a benchmark; there is no regulatory limit for cristobalite. See Section G.1.

Fugitive dust emissions from the excavated rock pile during closure would produce small offsite PM₁₀ concentrations. Both the annual and 24-hour average concentrations of PM₁₀ would be less than 1 percent of the regulatory standards for all three thermal load scenarios.

Table G-36 also lists potential air quality impacts for releases of cristobalite. The methods used were the same as those described in Section G.1.4.2 for the construction phase, where cristobalite was assumed to be 28 percent of the fugitive dust released, based on its percentage in parent rock. The land withdrawal area boundary cristobalite concentration would be small, about 0.1 percent of the benchmark level discussed in Section G.1.

For Modules 1 and 2, the same technique was used, but the estimated active excavated rock pile area would be about 20 percent larger, 4 percent smaller, and 6 percent larger than the Proposed Action for the high, intermediate, and low thermal load scenarios, respectively, based on the volume of rock added to the pile (TRW 1999b, page 6-79). The estimated air quality impacts from the excavated rock pile would also be about 20 percent larger, 4 percent smaller, and 6 percent larger than the Proposed Action for the high, intermediate, and low thermal load scenarios, respectively.

G.1.6.5 Exhaust Emissions from Surface Equipment

The consumption of diesel fuel and gasoline by surface equipment would emit the four criteria pollutants during closure. The analysis used the same method to determine pollutant release rates during closure that it used for the construction phase (see Section G.1.4.5). Table G-15 lists the estimated pollutant release rates of the equipment that would consume the fuel. Table G-37 lists by thermal load scenario the average amount of fuel consumed per year. The length of the closure phase would be 6, 6, or 15 years for the high, intermediate, or low thermal load scenario, respectively. Closure of the North Portal Operations Area would last 6 years (TRW 1999a, page 79).

Table G-37. Annual amount of fuel consumed (liters)^a during the closure phase.^b

Thermal load	South Portal diesel ^c	North Portal diesel ^d
High	250,000	340,000
Intermediate	620,000	340,000
Low	510,000	340,000

a. To convert liters to gallons, multiply by 0.26418.

b. Numbers are rounded to two significant figures.

c. Source: Based on total fuel consumed from TRW (1999b, page 6-37).

d. Source: Based on total fuel consumed from TRW (1998, page 87).

Table G-38 lists pollutant releases from surface diesel consumption. Table G-39 lists potential air quality impacts as pollutant concentration in air and percent of regulatory limit. Concentrations would be less than 1 percent of the regulatory limit for all thermal load scenarios.

Table G-38. Pollutant release rates from surface equipment during the closure phase.^a

Pollutant	Period	Mass of pollutant per averaging period (kilograms) ^b		Emission rate ^c (grams per second) ^d	
		South	North	South	North
<i>High thermal load</i>					
Nitrogen dioxide	Annual ^d	9,800	13,000	0.31	0.42
Sulfur dioxide	Annual	930	1,300	0.030	0.040
	24-hour ^e	3.7	5.1	0.13	0.18
	3-hour ^f	1.4	1.9	0.13	0.18
Carbon monoxide	8-hour ^g	15	21	0.52	0.71
	1-hour ^h	1.9	2.6	0.52	0.71
PM ₁₀	Annual	870	1,200	0.028	0.038
	24-hour	3.5	4.7	0.12	0.16
<i>Intermediate thermal load</i>					
Nitrogen dioxide	Annual	24,000	13,000	0.77	0.42
Sulfur dioxide	Annual	2,300	1,300	0.073	0.040
	24-hour	9.2	5.1	0.32	0.18
	3-hour	3.5	1.9	0.32	0.18
Carbon monoxide	8-hour	37	21	1.3	0.71
	1-hour	4.7	2.6	1.3	0.71
PM ₁₀	Annual	2,100	1,200	0.068	0.038
	24-hour	8.6	4.7	0.30	0.16
<i>Low thermal load</i>					
Nitrogen dioxide	Annual	20,000	13,000	0.63	0.42
Sulfur dioxide	Annual	1,900	1,300	0.060	0.040
	24-hour	7.6	5.1	0.26	0.18
	3-hour	2.8	1.9	0.26	0.18
Carbon monoxide	8-hour	31	21	1.1	0.71
	1-hour	3.8	2.6	1.1	0.71
PM ₁₀	Annual	1,800	1,200	0.056	0.038
	24-hour	7.1	4.7	0.24	0.16

a. Numbers are rounded to two significant figures.

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

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

d. To convert grams per second to pounds per hour, multiply by 7.9366.

Table G-39. Air quality impacts (micrograms per cubic meter) from surface construction equipment during the closure phase.

Pollutant	Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
<i>High thermal load</i>				
Nitrogen dioxide	Annual	0.080	100	0.080
Sulfur dioxide	Annual	0.0076	80	0.0095
	24-hour	0.057	365	0.016
	3-hour	0.45	1,300	0.035
Carbon monoxide	8-hour	0.67	10,000	0.0065
	1-hour	4.1	40,000	0.010
PM ₁₀	Annual	0.0071	50	0.014
	24-hour	0.053	150	0.035
<i>Intermediate thermal load</i>				
Nitrogen dioxide	Annual	0.13	100	0.13
Sulfur dioxide	Annual	0.013	80	0.016
	24-hour	0.093	365	0.025
	3-hour	0.74	1,300	0.057
Carbon monoxide	8-hour	1.1	10,000	0.011
	1-hour	6.6	40,000	0.017
PM ₁₀	Annual	0.012	50	0.024
	24-hour	0.087	150	0.058
<i>Low thermal load</i>				
Nitrogen dioxide	Annual	0.12	100	0.12
Sulfur dioxide	Annual	0.011	80	0.015
	24-hour	0.082	365	0.022
	3-hour	0.66	1,300	0.050
Carbon monoxide	8-hour	0.98	10,000	0.0095
	1-hour	5.9	40,000	0.015
PM ₁₀	Annual	0.010	50	0.020
	24-hour	0.076	150	0.051

a. Numbers are rounded to two significant figures.

b. Sources: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

For Modules 1 and 2, the same technique was used, but the amount of fuel used during the closure phase would increase. The annual diesel fuel use during closure would be 1.9, 0.81, and 1.2 times that of the Proposed Action for the high, intermediate, and low thermal load scenarios, respectively, based on total fuel use (TRW 1999b, page 6-77). The annual diesel fuel use for closure of the North Portal facility would be the same as that for the Proposed Action for all thermal load scenarios. Estimated air quality impacts for surface equipment during the operation and monitoring phase under Modules 1 and 2 would increase by about 1.4, 0.87, and 1.1 times the Proposed Action for the high, intermediate, and low thermal load scenarios, respectively.

G.1.7 RETRIEVAL SCENARIO

This section describes the method used to estimate air quality impacts during possible retrieval at the proposed repository. The retrieval contingency includes the construction of a retrieval storage facility and storage pad, and retrieval of the waste. Retrieval would last 11 years (TRW 1999b, page 6-32), while construction of the retrieval storage facility and storage pads would last 10 years (TRW 1999a, page I-20). DOE would construct the storage facility before beginning retrieval activities. Storage pads would be constructed in modules concurrently with retrieval activities. The analysis considered concurrent air quality impacts of retrieval and construction. The retrieval scenario work schedule would be one 8-hour shift a day, 5 days a week, 50 weeks a year.

The analysis estimated air quality impacts by calculating pollutant concentrations from various activities associated with retrieval. Emission rates were developed for each activity that would result in releases of pollutants. These rates were multiplied by the unit release concentration (see Section G.1.3) to calculate pollutant concentrations for comparison to the various regulatory limits.

The principal sources of particulates would be fugitive dust emissions from construction activities associated with the waste retrieval facility. The principal source of nitrogen dioxide, sulfur dioxide, and carbon monoxide would be fuel combustion during the construction of the waste retrieval facility and during retrieval of the waste. The following sections describe these sources in more detail.

G.1.7.1 Fugitive Dust from Construction of Retrieval Storage Facility

Construction activities such as earth moving and truck traffic would produce fugitive dust during the construction of the retrieval storage facility and storage pad. The analysis used the same method to estimate fugitive dust releases during retrieval as that for construction (see Section G.1.4.1). The amount of land disturbed to build the retrieval storage facility and storage pad would be 1 square kilometer (250 acres) (TRW 1999a, Table I-2, page I-22). In addition, a 1.8-kilometer (1.1-mile) rail line (TRW 1999a, page I-16) would also be constructed. Assuming the rail line is 0.06 kilometer (0.04 mile) wide, the rail line would require an additional 0.11 square kilometer (27 acres) of land to be disturbed.

Table G-40 lists fugitive dust release rates from construction of the retrieval facility and storage pad. Table G-41 lists air quality impacts as pollutant concentration in air and percent of regulatory limit. Fugitive dust emissions from construction of the retrieval facility and storage pad would produce small offsite (outside the land withdrawal area) PM₁₀ concentrations. Annual and 24-hour average concentrations of PM₁₀ would be less than 1 percent for facility construction and about 2 percent for storage pad construction of the regulatory standards for all three thermal load scenarios.

Table G-40. Fugitive dust release rates from surface construction of retrieval storage facility and storage pad (PM₁₀).^a

Period	Pollutant emission (kilograms) ^b	Emission rate (grams per second) ^c
Annual	25,000 per year	0.80
24-hour	100 per day	3.5 ^d

a. Numbers are rounded to two significant figures.

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

c. To convert grams per second to pounds per hour, multiply by 7.9366.

d. Based on an 8-hour release period.

Table G-41. Fugitive dust (PM₁₀) air quality impacts from surface construction of the retrieval storage facility and storage pad (micrograms per cubic meter).

Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
Annual	0.096	50	0.19
24-hour	0.67	150	0.44

a. Numbers are rounded to two significant figures.

b. Sources: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

G.1.7.2 Exhaust from Construction Equipment

Surface equipment would emit the four criteria pollutants during retrieval and during the construction of the retrieval storage facility and storage pad. The analysis used the same method to estimate pollutant release rates from fuel consumed by construction equipment during retrieval that was used for the construction phase (see Section G.1.4.5). During retrieval, fuel would be consumed at the South Portal

Operations Area; during the construction of the retrieval facility and storage pad, fuel would be consumed at the North Portal Operations Area. Table G-15 lists the pollutant release rates of the equipment that would consume the diesel fuel. The maximum amount of fuel used annually would be about 1.46 million liters (390,000 gallons) for surface construction (TRW 1999a, Table I-2, page I-22), about 1.7 million liters (460,000 gallons) for surface retrieval operations (TRW 1999a, Table I-3, page I-24), and about 27,000 liters (7,200 gallons) for subsurface retrieval operations (TRW 1999b, page 6-33). Total maximum annual usage would be about 1.9 million liters (500,000 gallons).

Table G-42 lists pollutant release rates for surface equipment during retrieval. Table G-43 lists the potential air quality impacts. Emissions from surface equipment during retrieval would produce small offsite criteria pollutant concentrations. All concentrations would be less than 1 percent of the regulatory standards.

Table G-42. Pollutant release rates from surface equipment during the retrieval scenario.^a

Pollutant	Period	Mass of pollutant per averaging time (kilograms) ^b	Emission rate ^c (grams per second) ^d
Nitrogen dioxide	Annual	75,000	2.4
Sulfur dioxide	Annual	7,100	0.22
	24-hour	28	0.98
	3-hour	11	0.98
Carbon monoxide	8-hour	110	4.0
	1-hour	14	4.0
PM ₁₀	Annual	6,600	0.21
	24-hour	26	0.92

a. Numbers are rounded to two significant figures.

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

c. Based on an 8-hour release period for averaging periods of 24 hour or less.

d. To convert grams per second to pounds per hour, multiply by 7.9366.

Table G-43. Air quality impacts from surface equipment during the retrieval scenario (micrograms per cubic meter of pollutant).

Pollutant	Period	Maximum concentration ^a	Regulatory limit ^b	Percent of regulatory limit ^a
Nitrogen dioxide	Annual	0.23	100	0.24
Sulfur dioxide	Annual	0.022	80	0.028
	24-hour	0.18	365	0.049
	3-hour	1.4	1,300	0.11
Carbon monoxide	8-hour	2.1	10,000	0.020
	1-hour	13	40,000	0.033
PM ₁₀	Annual	0.021	50	0.042
	24-hour	0.17	150	0.11

a. Numbers are rounded to two significant figures.

b. Sources: 40 CFR 50.4 through 50.11 and Nevada Administrative Code 445B.391.

G.2 Radiological Air Quality

This section describes the methods DOE used to analyze potential radiological impacts to air quality at the proposed Yucca Mountain Repository during the construction, operation and monitoring, and closure phases, and a possible retrieval scenario. The results are presented in Chapter 4, Section 4.1.2. It discusses the radioactive noble gas krypton-85, which would be released from surface facilities during the handling of spent nuclear fuel, and naturally occurring radon-222 and its radioactive decay products, which would be released from the rock to the subsurface facility and then to the ventilation air. The excavated rock pile would not be a notable additional source of radon-222, because the rock would not have enhanced concentrations of uranium or radium (the sources of radon-222) in comparison to surface

rock. Somewhat higher concentrations of radon-222 could be present at the rock pile itself but, in general, concentrations of radon-222 released from the excavated rock pile would not differ greatly from naturally occurring surface concentrations of radon.

G.2.1 LOCATIONS OF HYPOTHETICALLY EXPOSED INDIVIDUALS AND POPULATIONS

Members of the public and noninvolved workers could be exposed to atmospheric releases of radionuclides from repository activities. Doses to the maximally exposed individual and population within 80 kilometers (50 miles) were evaluated for the public. The dose to the maximally exposed noninvolved worker and the noninvolved worker populations at the repository and at the Nevada Test Site were also evaluated.

Public

The location of the maximally exposed individual member of the public would be about 20 kilometers (12 miles) south of the repository at the land withdrawal area boundary. This was determined to be the location of unrestricted public access that would have the highest annual average concentration of airborne radionuclides (see Section G.2.2). The locations calculated for nonradiological air quality impacts (Section G.1.2) would be somewhat different because the analysis estimated exposure to nonradiological pollutants for acute (short-term) exposures (1 to 24 hours) and for annual (continuous) exposures.

Table G-44 lists the estimated population of about 28,000 within 80 kilometers (50 miles) of the repository. This is the predicted population for 2000, based on projected changes in the region, including the towns of Beatty, Pahrump, Indian Springs, and the surrounding rural areas. The population in the vicinity of Pahrump was included in Table G-44 and evaluated for air quality impacts, even though the

Table G-44. Projected year 2000 population distribution within 80 kilometers (50 miles) of repository site.^{a,b,c}

Direction	Distance (kilometers)										Totals
	8	16	24	32	40	48	56	64	72	80	
S	0	0	16	238	430	123	0	10	0	0	817
SSW	0	0	0	315	38	0	0	7	0	0	360
SW	0	0	0	0	0	0	868	0	0	0	868
WSW	0	0	0	0	0	0	0	0	87	0	87
W	0	0	0	638	17	0	0	0	0	0	655
WNW	0	0	0	936	0	0	0	0	0	20	956
NW	0	0	0	28	2	0	0	0	33	0	63
NNW	0	0	0	0	0	0	0	0	0	0	0
N	0	0	0	0	0	0	0	0	0	0	0
NNE	0	0	0	0	0	0	0	0	0	0	0
NE	0	0	0	0	0	0	0	0	0	0	0
ENE	0	0	0	0	0	0	0	0	0	0	0
E	0	0	0	0	0	0	0	0	0	0	0
ESE	0	0	0	0	0	0	0	0	1,055	0	1,055
SE	0	0	0	0	3	0	13	0	0	206	222
SSE	0	0	0	0	23	172	6	17	6,117	16,399 ^d	22,734
Grand Total											27,817

- a. Source: 2000 population projected based on population data in TRW (1998, page 3-7).
- b. To convert kilometers to miles, multiply by 0.62137.
- c. There is a 4-kilometer (about 2.5-mile)-radius area around the North Portal, from which the analysis determined the 80-kilometer (50-mile) area.
- d. Includes the Pahrump vicinity population, which extends beyond the 80-kilometer region.

population extends beyond the 80-kilometer region. The analysis calculated both annual population dose and cumulative dose for the project phases over more than 100 years of construction, operation and monitoring, and closure.

Noninvolved (Surface) 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), or about 23 percent of the total number of hours in a year (8,760). All surface workers, regardless of work responsibility, were considered to be noninvolved workers for evaluation of exposure to radon-222 and radon decay products released from the subsurface facilities. For releases of noble gases (principally krypton-85) from spent fuel handling activities, potentially exposed noninvolved workers would be all surface workers except those in the Waste Handling and Waste Treatment Buildings. The noble gases would be released from the stack of the Waste Handling Building and workers in these facilities would not be exposed.

The maximally exposed noninvolved worker location would be in the South Portal Operations Area, where air from repository development activities would be exhausted. The analysis assumed that this worker would be in the office building about 100 meters (330 feet) northeast of the South Portal. This worker would be exposed to the annual average concentration of radon during the construction phase as radon concentrations increased with the increasing level of subsurface development. However, during operational activities, the radon level would remain approximately constant at the baseline concentration because the development area of the repository, ventilated and exhausted through the South Portal, would remain relatively constant. There would be no South Portal ventilation during monitoring activities and the closure phase, but the maximally exposed noninvolved worker would still be in the South Portal Operations Area.

The population and distribution of repository workers required to staff the North Portal Operations Area surface facilities would depend on the commercial spent nuclear fuel packaging scenario. As shown in Table G-45, the uncanistered packaging scenario would have the highest labor requirements for all project

Table G-45. Noninvolved (surface) worker population distribution for Yucca Mountain activities.^a

Worker location	Packaging scenario		
	Uncanistered	Disposable canister	Dual-purpose canister
<i>Construction</i>			
North Portal	656	457	485
South Portal	70	70	70
<i>Operation and monitoring</i>			
Emplacement and development			
North Portal	781 ^b	630 ^b	636 ^b
South Portal	1,277	962	982
Monitoring and maintenance			
North Portal – decommissioning	70	70	70
North Portal – monitoring and maintenance	1,354	982	1,023
South Portal	35	35	35
	6	6	6
<i>Closure</i>			
North Portal	363	256	275
South Portal	6	6	6
<i>Retrieval</i>			
North Portal – construction	780	780	780
North Portal – operations	108	108	108
South Portal	70	70	70

a. Sources: North Portal: TRW (1999a, pages 74, 75, and 79 to 81); South Portal: TRW (1999b, page 4-85).

b. Total workers exposed to krypton-85 releases from surface facilities. Does not include Waste Handling Building or Waste Treatment Building workers; does include 70 workers at the South Portal.

phases and activities in comparison to the disposable canister and dual-purpose canister scenarios. The number of North Portal workers would not vary for different thermal load scenarios. The estimated population of workers in the South Portal Operations Area was based on the number of full-time equivalents. This includes many workers who would be on the surface for only a portion of a day, as they prepared for underground work in the surface operations area. The number of South Portal workers was also assumed to remain constant for all thermal load scenarios.

Also evaluated as a potentially exposed noninvolved worker population were DOE workers at the Nevada Test Site. The analysis used a Nevada Test Site worker population of 6,576 workers (DOE 1996, Volume I, Appendix A, page A-69). For purposes of analysis, all these workers were assumed to be about 50 kilometers (30 miles) east-southeast of the repository at Mercury, Nevada.

G.2.2 METEOROLOGICAL DATA AND ATMOSPHERIC DISPERSION FACTORS

The basis for the atmospheric dispersion factors used in the dose calculations was a joint frequency distribution file for 1993 to 1997. These data were based on site-specific meteorological measurements made at air quality and meteorology monitoring Site 1, combined for 1993 to 1997 (TRW 1999c, page 11). Site 1 is about 1 kilometer (0.6 mile) south of the proposed North Portal surface facility location. Similar topographic exposure would lead to similar prevailing northerly and southerly winds at both locations. DOE used these data because an analysis of the data collected at all the sites showed Site 1 to be most representative of the surface facilities (TRW 1999c, page 7). The joint frequency data are somewhat different from the more detailed meteorological data used for the nonradiological air quality analysis. The dose calculations required only annual average data because they compare doses to annual limits, whereas criteria pollutant limits have 1-, 3-, 8-, or 24-hour averaging periods and the calculation of short-term criteria pollutant concentrations required hourly meteorological data. The nonradiological analysis also calculated concentrations only at the land withdrawal area boundary, not at onsite locations where workers would be.

Depending on the project phase and level of activity, subsurface ventilation air could be exhausted from any or all of three locations: the South Portal, emplacement (exhaust) shaft 1 or emplacement (exhaust) shaft 2. Both of these exhaust shafts would be on the ridge above the repository. Table G-46 lists the distribution of exhaust ventilation air among the three subsurface release points for project phases and activities. These distributions were used to calculate annual average atmospheric dispersion factors for radon releases from the subsurface.

The GENII software system (Napier et al. 1997, all) was used to calculate annual average atmospheric dispersion factors for radon released from the subsurface exhaust points and for noble gases released from the Waste Handling Building stack. The releases from the South Portal would be at ground level, while releases from the two emplacement shafts (ES-1 and ES-2) on the ridge above the repository were modeled as 60-meter (200-foot) releases. Noble gas releases from the Waste Handling Building would be from a 60-meter (200-foot) stack, also modeled as an elevated release. The population distribution data in Tables G-44 and G-45 were used to calculate population-weighted dispersion factors for public and noninvolved worker populations, which were then used to calculate collective doses. Table G-47 lists the individual and population-weighted atmospheric dispersion factors for the radon and krypton-85 release points at the site. These values do not incorporate the release distribution data in Table G-46. The radon dispersion factors would vary slightly among some combinations of project phase and thermal load scenarios because of the slight differences in release point contributions noted in Table G-46. Krypton-85 dispersion factors would not be affected.

Table G-46. Distribution (percent) of repository subsurface exhaust ventilation air.^a

Project phase and activity	Thermal load scenario	South Portal	Emplacement (exhaust) shaft 1	Emplacement (exhaust) shaft 2
Proposed Action				
<i>Construction</i>	All	100		
<i>Operation and monitoring</i>				
Development and emplacement	High	47	53	
	Intermediate	47	53	
	Low	55	42	3
Monitoring and maintenance	All		100	
<i>Closure</i>	Same exhaust distribution as monitoring and maintenance			
<i>Retrieval scenario</i>	Same exhaust distribution as monitoring and maintenance			
Inventory Modules 1 and 2				
<i>Construction</i>	All	100		
<i>Operation and monitoring</i>				
Development and emplacement	High	46	54	
	Intermediate	39	61	
	Low	42	40	18
Monitoring and maintenance	High		100	
	Intermediate		100	
	Low		50	50
<i>Closure</i>	Same exhaust distribution as monitoring and maintenance			

a. Source: Rasmussen (1998, all); TRW (1999b, pages 4-33 to 4-48).

G.2.3 RADIOLOGICAL SOURCE TERMS

There would be two distinctly different types and sources of radionuclides released to the air from activities at the repository. Naturally occurring radon-222 and its radioactive decay products would be released from the subsurface facility during all phases as the repository ventilation system removed airborne particulates from development operations and exhausted air heated by the emplaced materials. Radioactive noble gases would be released from commercial spent nuclear fuel during handling and transfer operations in the surface facilities during the operation and monitoring phase. Section G.2.3.1 discusses the releases of radon-222 and radon decay products. Section G.2.3.2 discusses the releases of radioactive noble gases from commercial spent nuclear fuel.

G.2.3.1 Release of Radon-222 and Radon Decay Products from the Subsurface Facility

In the subsurface facility the noble gas radon-222 would diffuse continually from the rock into the air of the repository drifts. Radioactive decay of the radon in the air of the drift would produce radon decay products, which would begin to come into equilibrium (having the same activity) with the radon-222 because their radioactive half-lives are much shorter than the 3.8-day half-life of radon-222. Key radionuclide members of the radon-222 decay chain are polonium-218 (sometimes known as radium A) and polonium-214 (radium C'), with half-lives of 3.05 minutes and 164 microseconds, respectively. Exhaust ventilation would carry the radon-222 and the radon decay products from the repository.

The estimates of radon-222 and radon decay product releases were based on concentration observations made in the Exploratory Studies Facility subsurface areas during site characterization. Because the repository would encompass the subsurface areas of the Exploratory Studies Facility, the analysis assumed that these observations would be a reasonable baseline. Concentrations at the 7,350-meter (4.6-mile) measuring station in the South Ramp ranged from 0.65 to 163 picocuries per liter with the ventilation system operating (TRW 1999c, electronic file attachment 7350EBF.XLS). The measured 50th-percentile concentration was 24 picocuries per liter, with 5th- and 95th-percentile concentrations of 1.7 and 124 picocuries per liter, respectively. Because the distribution of these concentration data was

Table G-47. Atmospheric dispersion factors for potentially exposed individuals and populations from releases at the repository site.^a

Release location ^b	Release type ^c	Receptor type	Receptor location	Dispersion factor ^d
<i>Radon releases^e</i>				
Public				
South Portal	G	individual	20 km ^f south	2.2×10^{-8}
South Portal	G	population	80 km radius	1.2×10^{-4}
Emplacement shafts 1, 2 ^g	E	individual	20 km south	6.0×10^{-9}
Emplacement shafts 1, 2 ^g	E	population	80 km radius	3.0×10^{-5}
Noninvolved workers				
South Portal	G	individual	100 meters ^h northeast	6.2×10^{-5}
South Portal	G	population	South Portal Operations Area	3.2×10^{-3}
South Portal	G	individual	North Portal 2.8 km north-northeast ⁱ	1.9×10^{-7}
South Portal	G	individual	Nevada Test Site, 50 km east-southeast ⁱ	6.9×10^{-10}
Emplacement shaft 1	E	individual	North Portal 4.2 km southeast	9.0×10^{-9}
Emplacement shaft 1	E	individual	South Portal 6.3 km south-southeast	2.0×10^{-8}
Emplacement shaft 2	E	individual	North Portal, 4.5 km east-southeast	4.9×10^{-9}
Emplacement shaft 2	E	individual	South Portal, 5.3 km southeast	6.7×10^{-9}
Emplacement shafts 1, 2 ^g	E	individual	Nevada Test Site, 50 km east-southeast	2.7×10^{-10}
<i>Krypton-85 releases</i>				
Public				
Waste Handling Bldg. stack	E	individual	20 km south	6.0×10^{-9}
Waste Handling Bldg. stack	E	population	80 km radius	3.0×10^{-5}
Noninvolved workers				
Waste Handling Bldg. stack	E	individual	North Portal, 0.4 km north-northwest	1.5×10^{-6}
Waste Handling Bldg. stack	E	individual	South Portal, 2.8 km south-southwest	5.4×10^{-8}
Waste Handling Bldg. stack	E	population	Uncanistered packaging scenario	2.4×10^{-4}
Waste Handling Bldg. stack	E	population	Disposable canister packaging scenario	1.9×10^{-4}
Waste Handling Bldg. stack	E	population	Dual-purpose canister packaging scenario	1.9×10^{-4}
Waste Handling Bldg. stack	E	individual	Nevada Test Site, 50 km east-southeast ⁱ	2.7×10^{-10}

a. Numbers are rounded to two significant figures.

b. Source: Radon releases: TRW (1999b, pages 4-33 to 4-48); krypton-85 releases: TRW (1999a, page 41).

c. G = ground level; E = elevated.

d. Dispersion factor units are seconds per cubic meter for individuals, and person-seconds per cubic meter for populations.

e. Radon includes radon-222 and its radioactive decay products.

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

g. Difference in dispersion between the two emplacement shafts is small for this application.

h. To convert meters to feet, multiply by 3.2808.

i. The population dose was calculated at this point by multiplying the individual dispersion factor times population size.

highly skewed, the analysis assumed that the 50th-percentile value was most representative of the entire concentration range.

Exhaust ventilation flowrates in the South Ramp when the radon concentration measurements were made measured from about 100 to 125 cubic meters per second (214,000 to 265,000 cubic feet per minute) (TRW 1999c, electronic file attachment DECRPT.XLS). A value of 110 cubic meters per second (230,000 cubic feet per minute) was used as a representative South Ramp flowrate. This information, combined with an Exploratory Studies Facility excavated volume of 360,000 cubic meters (470,000 cubic yards) (TRW 1999b, page 4-27), yielded a calculated repository air exchange rate of about 1 per 3,300 seconds (about one exchange per hour) and a baseline for radon-222 releases. The exchange rate is the excavated volume (in cubic meters) divided by the ventilation flowrate (in cubic meters per second). The analysis assumed these conditions would be representative for the Exploratory Studies Facility through the beginning of the construction phase. The estimated release of radon-222 and radon decay products for this configuration would be about 80 curies per year.

Table G-48 lists the key input parameters, namely the beginning and ending excavated repository volumes, repository average ventilation rates, and repository average air exchange rates, for each of the phases and thermal load scenarios of the Proposed Action. The analysis assumed that increases in excavated repository volume and ventilation flowrate would occur linearly. In addition, Table G-48 lists the estimated releases of radon-222 and radon decay products annually and by phase.

Table G-48. Estimated radon-222 releases for repository activities for the Proposed Action inventory.^a

Period and thermal load	Repository volume (millions of cubic meters) ^{b,c}		Average ventilation rate (cubic meters per second)	Average air exchange rate	Annual average radon ^d release (curies)	Total radon ^d release (curies)
	Beginning	Ending				
<i>Construction (5 years)</i>						
High	0.36	1.9	205	6,200	300	1,500
Intermediate	0.36	2.2	205	7,200	340	1,700
Low	0.36	2.2	205	7,200	340	1,700
<i>Operations (24 years)</i>						
High	1.9	4.7	570	6,700	880	21,000
Intermediate	2.2	5.7	570	7,900	1,000	25,000
Low	2.2	14	680	13,000	1,900	46,000
<i>Monitoring (76 years)</i>						
High	4.7	4.7	190	24,000	1,100	83,000
Intermediate	5.7	5.7	190	29,000	1,300	99,000
Low	14	14	490	28,000	3,200	240,000
<i>Total Operation and Monitoring Phase (100 years)</i>						
High					1,000	100,000
Intermediate					1,200	120,000
Low					2,900	290,000
<i>Closure phase (6, 6, and 15 years)</i>						
High	4.7	4.7	190	24,000	1,100	6,600
Intermediate	5.7	5.7	190	29,000	1,300	7,900
Low	14	14	490	28,000	3,200	48,000
<i>Total, all phases (111, 111, 120 years)</i>						
High						110,000
Intermediate						130,000
Low						340,000
<i>Retrieval scenario (14 years)</i>						
High	4.7	4.7	190	24,000	1,100	14,000

a. Numbers are rounded to two significant figures; totals might not equal sums of values due to rounding.

b. Source: TRW (1999b, pages 4-27, 6-6, and 6-16).

c. To convert cubic meters to cubic yards, multiply by 1.3079.

d. Includes radon-222 and radon decay products.

Construction Phase

During the 5 years of construction, 1.5 million cubic meters (1.96 million cubic yards) of rock would be removed for the high thermal load scenario and 1.9 million cubic meters (2.4 million cubic yards) for the intermediate and low thermal load scenarios (TRW 1999b, page 6-6). During the same period, the ventilation flow would increase from 110 cubic meters per second (230,000 cubic feet per minute) to 270 cubic meters per second (570,000 cubic feet per minute) (TRW 1999b, pages 4-33 to 4-38). Releases of radon-222 would be low but would vary within 15 percent among all three thermal load scenarios, because they would have the same ventilation flow rates but different repository volumes.

Operation and Monitoring Phase

Operation Activities. Development activities would last 22 years during operation and monitoring. During this period about 2.9 million, 3.4 million, and 11.8 million cubic meters (3.8 million, 4.5 million, and 15.4 million cubic yards) of rock would be removed for the high, intermediate, and thermal load

scenarios, respectively (TRW 1999b, page 6-16). The repository excavation would be complete during the last two years of the operation activity period, as emplacement activities continued. The flowrate for the repository during emplacement and development activities of the high and intermediate thermal load scenarios would be the maximum development side flowrate [270 cubic meters per second (570,000 cubic feet per minute)], and the maximum emplacement side flowrate [300 cubic meters per second (640,000 cubic feet per minute)] (TRW 1999b, pages 4-33 to 4-38). The flowrate during the low thermal load scenario would vary from 570 to 740 cubic meters per second (1.2 million to 1.6 million cubic feet per minute), depending on the stage of emplacement activities.

The estimation of radon releases for the high and intermediate thermal load scenarios was based on development and emplacement activities taking place only in the upper (primary) block. However, for the low thermal load scenario development and emplacement would be incremental, beginning in the upper block, moving on to the lower block, and finally to the Area 5 block (TRW 1999b, page 3-3). When emplacement in a block was complete, that block would enter an interim period of monitoring and maintenance as activities continued in the other blocks. The analysis assumed that the upper block would be in this interim status for 10 years and the lower block for 5 years.

The high and intermediate thermal load scenarios would have the lowest radon releases because they would use only the upper (primary) block. The low thermal load scenario would have a higher radon release because of the greater repository volume, which would require three blocks, and the added contribution from exhaust ventilation during the interim monitoring and maintenance of the upper and lower blocks.

Monitoring Activities. No excavation would take place during monitoring, and the exhaust flowrate would remain constant. The much greater repository volume for the low thermal load scenario, which would require larger exhaust flowrates, would result in larger releases of radon-222 and radon decay products to the atmosphere through the exhaust ventilation.

Monitoring and maintenance activities would last from 26 to 276 years. Total releases of radon over 26 years would be approximately 29,000, 34,000, and 84,000 curies for the high, intermediate, and low thermal load scenarios, respectively. Total releases of radon over 276 years would be approximately 300,000, 360,000, and 890,000 curies for the high, intermediate, and low thermal load scenarios, respectively. The estimated annual radon release and concentration would be the same as those listed for monitoring in Table G-48.

For 100 years of operation and monitoring, the low thermal load scenario would involve approximately 2.5 times more radon release than the high or intermediate thermal load scenario. About 70 to 75 percent of the radon would be released during the monitoring and maintenance period for all three thermal load scenarios, not including the interim monitoring and maintenance for the low thermal load scenario.

Closure Phase

Annual releases of radon-222 and radon decay products during the closure phase would be the same as for the monitoring period. Differences in the lengths of the closure phases for the three thermal load scenarios would lead to differences in the total amount of radon released. Differences among the thermal load scenarios would be for the same reasons as for the monitoring period, namely the larger repository volume and exhaust ventilation flowrate of the low thermal load scenario.

Retrieval

Only the high thermal load scenario was evaluated for a postulated retrieval scenario. Annual releases of radon-222 and radon decay products would be the same as for the monitoring activities and closure phases. Releases were estimated for 13 years, including 2 years of retrieval-related construction activities plus 11 years of retrieval operations.

Inventory Modules 1 and 2

Releases of radon-222 and radon decay products for Inventory Modules 1 and 2 were estimated using the same methods as for the Proposed Action. The major differences would be the larger repository volumes and higher ventilation flowrates, which would result in larger releases of radon. In addition, 38 years would be required to complete operations (which includes 36 years of development), 62 years would be required for monitoring, and the closure phase would be longer. Table G-49 lists the estimates of radon release and key parameter values. Releases of radon would be higher for the inventory modules than for the Proposed Action in all cases.

Table G-49. Estimated radon-222 releases for repository activities for Inventory Modules 1 or 2.^a

Thermal load	Repository volume (millions of cubic meters) ^{b,c}		Average ventilation rate (cubic meters per second)	Average air exchange rate(s)	Annual average radon release (curies)	Total radon release (curies)
	Beginning	Ending				
<i>Construction (5 years)</i>						
High	0.36	2.1	205	6,900	330	1,600
Intermediate	0.36	2.1	205	6,900	330	1,600
Low	0.36	2.1	205	6,900	330	1,600
<i>Operations (38 years)</i>						
High	2.1	8.7	590	9,500	1,300	49,000
Intermediate	2.1	9.0	690	8,200	1,300	51,000
Low	2.1	24	800	16,000	3,100	120,000
<i>Monitoring (62 years)</i>						
High	8.7	8.7	300	29,000	2,000	125,000
Intermediate	9.0	9.0	490	18,000	2,100	130,000
Low	24	24	890	27,000	5,500	340,000
<i>Total operation and monitoring phase (100 years)</i>						
High					1,700	170,000
Intermediate					1,800	180,000
Low					4,600	460,000
<i>Closure (13, 17, and 27 years)</i>						
High	8.7	8.7	300	29,000	2,000	26,000
Intermediate	9.0	9.0	490	18,000	2,100	35,000
Low	24	24	890	27,000	5,500	150,000
<i>Totals (118, 122, and 132 years)</i>						
High						200,000
Intermediate						220,000
Low						610,000

a. Numbers are rounded to two significant figures; totals might not equal sums of values due to rounding.

b. Source: TRW (1999b, pages 4-27, 6-47, and 6-55).

c. To convert cubic meters to cubic yards, multiply by 1.3079.

G.2.3.2 Release of Radioactive Noble Gases from the Surface Facility

The unloading and handling of commercial spent nuclear fuel would produce the only routine emissions of manmade radioactive materials from repository facilities. No releases would occur as a result of emplacement activities. Shipping casks containing uncanistered spent nuclear fuel in dual-purpose canisters would be opened in the transfer pool of the Waste Handling Building at the North Portal Operations Area. Shipping casks containing spent nuclear fuel in disposable canisters would be opened in a dry transfer cell. During spent fuel handling and transfer, radionuclides could be released from a small percentage of fuel elements with pinhole leaks in the fuel cladding; only noble gases would escape the pool and enter the ventilation system of the Waste Handling Building (TRW 1999a, page 17). The largest release of radionuclides from surface facilities would be krypton-85, with about 2,600 curies released annually from the uncanistered and dual-purpose canister packaging options. Krypton-85 would also be the major dose contributor from the airborne pathway. Releases of other noble gas radionuclides would

be very small, with estimated annual releases of about 0.0000010 curie of krypton-81, 0.000033 curie of radon-219, 0.014 curie of radon-220, 0.0000046 curie of radon-222, and small quantities of xenon-127 (TRW 1999a, page 75). The same annual releases would occur for both the Proposed Action and for the inventory modules. Table G-50 lists estimated annual average releases of krypton-85 from fuel handling by packaging option. All spent nuclear fuel and DOE high-level radioactive waste in disposable canisters would be transferred from shipping casks to disposal containers inside shielded rooms (hot cells) in the Waste Handling Building. Because all DOE material would be in disposable canisters under all packaging scenarios, no radionuclide releases from these materials would occur.

Table G-50. Krypton-85 releases (curies) from surface facility handling activities for commercial spent nuclear fuel during the operation and monitoring phase. ^a

Packaging option	Annual release ^b	Proposed Action (24 years)	Inventory Module 1 or 2 (38 years)
Uncanistered	2,600	61,000	97,000
Disposable canister	90	2,200	3,500
Dual-purpose canister	2,600	62,000	98,000

a. Numbers are rounded to two significant figures.

b. Source: TRW (1999a, page 75).

Releases from the surface facility would be the same for the three thermal load scenarios. These releases were based on the following assumptions for commercial spent nuclear fuel (TRW 1999a, pages 18 and 19):

- Pressurized-water reactor burnup of about 40 gigawatt-days per metric ton of uranium with 3.6-percent enrichment and an average of 26 years decay
- Boiling-water reactor burnup of 32 gigawatt-days per metric ton of uranium with 3.0-percent enrichment and an average of 27 years decay
- A failure rate of 0.25 percent for fuel assemblies in the canisters, allowing gaseous radionuclides (isotopes of krypton, radon, and xenon) to escape
- Radionuclides other than noble gases (such as cobalt-60, cesium-137, and strontium-90) would not escape the transfer pool if released from fuel assemblies

G.2.4 DOSE CALCULATION METHODOLOGY

The previous three sections provided information on the location and distribution of potentially affected individuals and populations (Section G.2.1), atmospheric dispersion (Section G.2.2), and the type and quantity of radionuclides released to air (Section G.2.3) in the Yucca Mountain region. The analysis used these three types of information to estimate the radionuclide concentration in air (in picocuries of radionuclide per liter of air) at a specific location or for an area where there would be a potentially exposed population. The estimation of the radiation dose to exposed individuals or populations from concentrations of radionuclides in air used this information and published or derived dose factors. This section describes the concentration-to-dose conversion factors that the analysis used to estimate radiation dose to members of the public and noninvolved workers from releases of radionuclides at the repository.

G.2.4.1 Dose to the Public

The analysis estimated doses to members of the public using screening dose factors from the National Council on Radiation Protection and Measurements (NCRP 1996, Volume I, pages 113 and 125). The analysis considered all exposure pathways, including inhalation, ingestion, and direct external radiation from radionuclides in the air and on the ground. For noble gases such as krypton-85, only direct external

exposure from the radionuclides in the air would be a contributing pathway. For radon-222, the short-lived decay products would account for essentially all of the dose. The screening dose factors indicate that direct external radiation from radionuclides deposited on the ground would account for about 40 percent of the dose; ingestion of these decay products in foodstuffs and inadvertently consumed soil would account for about 60 percent, based on the published screening dose factors. Inhalation and external irradiation from radionuclides in the air would be minor exposure pathways. The analysis calculated the estimated dose from a specific radionuclide by multiplying the radionuclide-specific dose factor by the estimated air concentration at the exposure location. The results are reported in Chapter 4, Section 4.1.2. Table G-51 lists the screening dose factors for krypton-85 and radon-222 for members of the public. Results are presented in Chapter 4, Section 4.1.2.

Table G-51. Factors for estimating dose to the public and noninvolved workers per concentration of radionuclide in air (millirem per picocurie per liter per hour) for krypton-85 and radon-222.^{a,b}

Radionuclide	Public ^c	Noninvolved worker
Krypton-85	0.0000013	0.0000013
Radon-222	0.25 ^d	0.029 ^e

- Numbers are rounded to two significant figures.
- Dose factors for radon-222 include dose contribution from decay products.
- Source: NCRP (1996, page 61); assumed an exposure time of 8,000 hours per year.
- Includes all exposure pathways.
- Source: ICRP (1994, pages 5 and 24); 100 percent equilibrium between radon and decay products; inhalation pathway only.

G.2.4.2 Dose to Noninvolved Workers

The analysis used a National Council on Radiation Protection and Measurements screening dose factor to calculate doses to noninvolved workers from krypton-85 because the exposure pathway is simple (air submersion only) and is the same as for members of the public. Table G-51 also lists this factor. However, the analysis did not use a National Council on Radiation Protection and Measurements screening dose factor to estimate the dose to noninvolved workers from radon-222 and its decay products. The parameters and exposure scenarios used to derive the National Council on Radiation Protection and Measurements screening dose factors for radon-222 and its decay products would not be appropriate for the potential exposure scenario for noninvolved workers at the Yucca Mountain site. Dose to noninvolved workers on the surface would be due mainly to inhalation of the radon decay products, and not from the other exposure pathways noted above for the public. Therefore, the analysis developed a Yucca Mountain repository-specific exposure scenario using site-specific parameters where appropriate. The dose conversion factor is from Publication 65 of the International Commission on Radiological Protection (ICRP 1994, page 24). This dose factor, which is 0.5 rem per working level month for inhalation of radon decay products by workers, corresponds to 0.029 millirem per picocurie per liter per hour, with radon decay products in 100 percent equilibrium (equilibrium factor of 1.0) with the radon-222 parent (ICRP 1994, page 5).

In estimating dose from radon and radon decay products released from the subsurface facility, the analysis assumed the maximally exposed noninvolved worker would be in an office about 100 meters (330 feet) northeast of the South Portal. For the construction phase and development activities, the noninvolved worker exposure analysis used the distribution of radon concentration measurements made at the 7,350-meter (4.6-mile) station in the South Ramp of the Exploratory Studies Facility. These were the best available data for estimating releases of radon from the facility (TRW 1999c, page 12). There would be no releases from the South Portal during the other project phases. Measured concentrations ranged from 0.65 to 163 picocuries per liter, with a median value of 24 picocuries per liter, as noted in Section G.2.3.1. In addition, the analysis considered the distribution of the measured values of the equilibrium fraction

between radon-222 and the decay products. This value ranged from 0.0022 to 0.44, with a median of 0.14 (TRW 1999c, electronic file attachment RNFBF.XLS). The annual average atmospheric dispersion factor from the South Portal to the office building would be approximately 6.2×10^{-5} seconds per cubic meter for both the construction phase and development activities (Table G-47), although differences in exhaust flowrate (205 and 269 cubic meters per second, respectively, would result in minor differences in dispersion. The analysis assumed the maximally noninvolved worker would be exposed from 1,600 to 2,000 hours per year.

The estimated median dose to a maximally exposed noninvolved worker during the construction phase would be approximately 5 (4.7 to 5.4) millirem per year. The dose from the Proposed Action intermediate and low thermal load scenarios would be somewhat higher than that from the high thermal load scenario because of the larger average repository volume for these two scenarios during the construction phase (Table G-48). The estimated 5th-percentile dose would be about 0.2 millirem per year for both cases and the 95th-percentile dose would be 42 and 48 millirem per year, respectively. The dose during development activities would be the same for all three thermal load scenarios, with a median dose of about 3.4 millirem per year. The estimated 5th-percentile dose would be about 0.2 millirem per year and the 95th-percentile dose about 29 millirem per year. These estimates were made using a Monte Carlo uncertainty analysis. There would be a small contribution from external radiation, but the analysis did not consider it because it would be indistinguishable from normal external background radiation. The estimated dose from Module 1 or 2 would be about the same as those for the intermediate and low thermal load scenarios.

During the construction phase the maximally exposed noninvolved worker would receive a somewhat larger potential dose because of a larger average repository volume, which would be exhausted through the South Portal, and additional radon release. During operations the ventilation systems for the subsurface development and emplacement areas would be separate. The analysis assumed that the volume during Exploratory Studies Facility operations would represent the volume of the development side exhausted through the South Portal. This volume is somewhat smaller than the estimated average construction phase repository volume.

REFERENCES

- Cowherd, Muleski, and Kinsey 1988 Cowherd, C., G. E. Muleski, and J. S. Kinsey, 1988, *Control of Open Fugitive Dust Sources, Final Report*, pp. 4.1 to 5.41, EPA-450/3-88-008, Midwest Research Institute, Kansas City, Missouri. [243438]
- DOE 1995 DOE (U.S. Department of Energy), 1995, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs: Final Environmental Impact Statement*, DOE/EIS-0203-F, Office of Environmental Management, Idaho Operations Office, Idaho Falls, Idaho. [102617]
- DOE 1996 DOE (U.S. Department of Energy), 1996, *Final Environmental Impact Statement for the Nevada Test Site and Off-Site Locations in the State of Nevada*, DOE/EIS-0243-F, Nevada Operations Office, Las Vegas, Nevada. [239895]
- DOE 1997a DOE (U.S. Department of Energy), 1997a, *Waste Isolation Pilot Plant Disposal Phase Final Supplemental Environmental Impact Statement*, DOE/EIS-0026-S-2, Carlsbad Area Office, Carlsbad, New Mexico. [238195]

- DOE 1997b DOE (U.S. Department of Energy), 1997b, *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste*, DOE/EIS-0200-F, Office of Environmental Management, Washington, D.C. [232988]
- DOE 1997c DOE (U.S. Department of Energy), 1997c, *Yucca Mountain Site Characterization Project – Map for Contaminated Areas*, map, YMP-97-022.0, Office of Civilian Radioactive Waste Management, Yucca Mountain Project Office, Las Vegas, Nevada. [MOL.19990511.0292]
- DOE 1998 DOE (U.S. Department of Energy), 1998, *Air Quality Control Design Analysis*, BCAD00000-01717-0200-00008, Revision 00, Office of Civilian Radioactive Waste Management, Washington, D.C. [MOL.19980729.0044]
- EPA 1987 EPA (U.S. Environmental Protection Agency), 1987, *On-Site Meteorological Program Guidance for Regulatory Modeling Applications*, Wordperfect® reissue of the June 1987 EPA document, EPA-450/4-87-013, Office of Air Quality Planning and Standards, Office of Air and Radiation, Research Triangle Park, North Carolina. [210292]
- EPA 1988 EPA (U.S. Environmental Protection Agency), 1988, *Gap Filling PM₁₀ Emission Factors for Selected Open Area Dust Sources*, EPA-450/4-88-003, Midwest Research Institute, Kansas City, Missouri. [243553]
- EPA 1991 EPA (U.S. Environmental Protection Agency), 1991, *Compilation of Air Pollutant Emission Factors, Volume II: Mobile Sources*, AP-42, Supplement A, Washington, D.C. [243439]
- EPA 1995a EPA (U.S. Environmental Protection Agency), 1995a, *User's Guide for Industrial Source Complex (ISC3) Dispersion Models*, EPA-454/B-95-003a, Emissions, Monitoring, and Analysis Division, Office of Air Quality Planning and Standards, Research Triangle Park, North Carolina. [243563]
- EPA 1995b EPA (U.S. Environmental Protection Agency), 1995b, *Compilation of Air Pollutant Emission Factors, Fifth Edition, AP-42, Volume I: Stationary Point and Area Sources*, Research Triangle Park, North Carolina. [226367]
- EPA 1996 EPA (U.S. Environmental Protection Agency), 1996, *Ambient Levels and Noncancer Health Effects of Inhaled Crystalline and Amorphous Silica: Health Issue Assessment*, EPA/600/R-95/115, National Center for Environmental Assessment, Office of Research and Development, Washington, D.C. [243562]
- Fransioli 1999 Fransioli, P., 1999, "Telephone Log for Number of Days with Precipitation Greater Than 0.1 Inches," internal personal communication with C. Fosmire, February 4, TRW Environmental Safety Systems Inc., Las Vegas, Nevada. [MOL.19990511.0282]
- ICRP 1994 ICRP (International Commission on Radiological Protection), 1994, *Protection Against Radon-222 at Home and at Work*, Publication 65, Pergamon Press, Oxford, Great Britain. [236754]

- Jessen 1998
Jessen, J., 1998, "Additional Land Disturbance at Yucca Mountain from Repository Construction (Base Case and Extended Inventory)," internal memorandum, July 23, Jason Technologies Corporation, Las Vegas, Nevada. [MOL.19990602.0181]
- Napier et al. 1997
Napier, B. A., D. L. Streng, R. A. Peloquin, J. V. Ramsdell, and P. D. Rittmann, 1997, *RSICC Computer Code Collection, GENII 1.485, Environmental Radiation Dosimetry Software System*, CCC-601, PNL-6584, Radiation Safety Information Computational Center, Oak Ridge National Laboratory, Hanford, Washington. [206898]
- NCRP 1996
NCRP (National Council on Radiation Protection and Measurements), 1996, *Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground, Recommendations of the National Council on Radiation Protection and Measurements*, Report No. 123, Bethesda, Maryland. [225158, Volume 1; 234986, Volume 2]
- Rasmussen 1998
Rasmussen, D. G., 1998, "More Questions on Repository Ventilation," electronic communication to T. Ikenberry (Dade Moeller & Associates), September 23, TRW Environmental Safety Systems Inc., Las Vegas, Nevada. [MOL.19990511.0300]
- Seinfeld 1986
Seinfeld, J. H., 1986, *Atmospheric Chemistry and Physics of Air Pollution*, pp. 26-31, John Wiley and Sons, Inc., New York, New York. [243754]
- Smith 1999
Smith, A., 1999, "Telephone Log for Disturbed Area of Muck Pile in a Given Year," personal communication with C. Fosmire (PNNL), February 4, Argonne National Laboratory, Argonne, Illinois. [MOL.19990511.0283]
- TRW 1998
TRW (TRW Environmental Safety Systems Inc.), 1998, *Yucca Mountain Site Characterization Project: Summary of Socioeconomic Data Analyses Conducted in Support of the Radiological Monitoring Program, April 1997 to April 1998*, Las Vegas, Nevada. [MOL.19980803.0064]
- TRW 1999a
TRW (TRW Environmental Safety Systems Inc.), 1999a, *Repository Surface Design Engineering Files Report*, BCB000000-01717-5705-00009, Revision 03, Las Vegas, Nevada. [MOL.19990615.0238]
- TRW 1999b
TRW (TRW Environmental Safety Systems Inc.), 1999b, *Engineering File - Subsurface Repository*, BCA000000-01717-5705-00005, Revision 02 with DCN1, Las Vegas, Nevada. [MOL.19990622.0202, document; MOL.19990621.0157, DCN1]
- TRW 1999c
TRW (TRW Environmental Safety Systems Inc.), 1999c, *Environmental Baseline File for Meteorology and Air Quality*, B000000000-01717-5705-00126, Revision 00, Las Vegas, Nevada. [MOL.19990302.0186]



Appendix H

Potential Repository Accident
Scenarios: Analytical Methods
and Results

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APPENDIX H. POTENTIAL REPOSITORY ACCIDENT SCENARIOS: ANALYTICAL METHODS AND RESULTS

This appendix describes the methods and detailed results of the analysis the U.S. Department of Energy (DOE) performed for the Yucca Mountain Repository Environmental Impact Statement (EIS) to assess radiological impacts from potential accident scenarios at the proposed repository. The methods apply to repository accidents that could occur during preclosure only, including operation and monitoring, retrieval, and closure. In addition, this appendix describes the details of calculations for specific accidents that the analysis determined to be credible. Appendix J describes the analytical methods and results for accidents that could occur at the 72 commercial and 5 DOE sites and during transportation to the proposed repository.

The accident scenarios in this analysis, and the estimated impacts, are based on current information from the repository design (TRW 1999a, all). The results are based on assumptions and analyses that were selected to ensure that the impacts from accident scenarios are not likely to be underestimated. DOE has not developed the final design and operational details for the repository, and these details could result in lower impacts. The Department is currently engaged in preliminary efforts to identify accidents and evaluate their impacts as required to support the License Application for the repository that it will send to the Nuclear Regulatory Commission, and to show that the repository would comply with appropriate limits on radiation exposure to workers and the public from accidental releases of radionuclides. The final design could include additional systems and operational requirements to reduce the probability of accidents and to mitigate the release of radionuclides to ensure compliance with these safety requirements. The results from the accident analysis to meet licensing requirements would be more specific and comprehensive than those discussed in this appendix and would reflect final repository design and operational details.

H.1 General Methodology

Because of the large amount of radioactive material to be handled at the proposed repository (see Appendix A), the focus of the analysis was on accident scenarios that could cause the release of radioactive material to the environment. The methodology employed to estimate the impact of accidents involving radioactive material included (1) evaluation of previous accident analyses performed for the repository, (2) identification of bounding accidents (reasonably foreseeable accidents with the maximum consequences) from the previous analyses, (3) identification of other credible accidents the previous analyses did not evaluate, (4) analyses of the selected accidents to determine the amount of radioactive material an accident could release to the environment, and (5) estimation of the consequences of the release of radioactive material in terms of health effects to workers and the public.

The analysis approach involved identifying bounding accidents (that is, accidents with maximum consequences) for each operational phase of the proposed repository. The analysis evaluated the impacts for these accidents, assuming the accident occurred without regard to the estimated probability. Thus, the analysis provides the impacts that could occur for the worst credible accidents. The results do not represent risk estimates because the impacts do not include a consideration of accident probability, which in most cases is very low. The risk from all repository accidents would be likely to be far less than the low risk, which DOE estimated by assuming that all of the bounding (maximum consequence) accidents would occur.

Accident frequency estimates were derived to establish the credibility of accident sequences and were not used to establish risk. Estimates of accident frequency are very uncertain due to the preliminary nature of the currently available repository design information and would be more fully evaluated in the safety

analysis required to support a License Application for the repository. Based on the available design information, the accident analysis approach was used to ensure that impacts from accidents are not likely to be underestimated (whether they are low-probability with high-consequence accidents or high-probability with low-consequence accidents).

For accidents not involving radioactive materials, the analysis determined that application of accident statistics from other DOE operations provided a reasonable estimate of nonradiological accident impacts (see Section H.2.2).

H.2 Potential Repository Accident Scenarios

The proposed Yucca Mountain Repository has been the subject of intense evaluations for a number of years. Some of these evaluations included in-depth considerations of preclosure accidents that could occur during repository operations. The EIS used these previous evaluations, to the extent they are applicable and valid, to aid in the identification of initiating events, develop sequences, and estimate consequences. The EIS groups accidents as radiological accidents (Section H.2.1) that involve the unplanned release of radioactive material, and nonradiological accidents that involve toxic and hazardous materials (Section H.2.2).

H.2.1 RADIOLOGICAL ACCIDENT SCENARIOS

Previous analyses that considered impacts of radiological accidents during preclosure included evaluations by Sandia National Laboratories and others (Jackson et al. 1984, all; SNL 1987, all; Ma et al. 1992, all; BMI 1984, all), and include more recent evaluations (DOE 1996a,b, all; DOE 1997a,b all; Kappes 1998, all; TRW 1997a, all). These evaluations were reviewed to assist in this assessment of radiological impacts from accidents during repository operations. In addition, EISs that included accident evaluations involving spent nuclear fuel and high-level radioactive waste were reviewed and used as applicable (USN 1996, all; DOE 1995, all).

Radiological accidents involve an initiating event that can lead to a release of radioactive material to the environment. The analysis considered accidents separately for two types of initiating events: (1) internal initiating events that would originate in the repository and involve equipment failures or human errors, or a combination of 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. The analysis examined a spectrum of accidents, from high-probability/low-consequence accidents to low-probability/higher-consequence accidents.

H.2.1.1 Internal Events – Waste Handling Building

The most recent and comprehensive repository accident scenario analysis for internal events in the Waste Handling Building is presented in Kappes (1998, all). This analysis considered the other important applicable accidents that previous analyses identified. It performed an in-depth evaluation of all operations planned for the repository and identified bounding accidents (those with the highest estimated risk) for each operation. More than 150 accidents were selected for analysis in eight operational categories. The accidents were identified based on multiple sources, including the *Preliminary MGDS Hazards Analysis* (DOE 1996b, all), current facility design drawings, and discussions with design personnel. These 150 accidents were reduced to 16 bounding accidents by retaining accidents that would produce the highest doses for groups of similar events (Kappes 1998, page 35). DOE used event trees and fault tree evaluation to estimate frequencies for the accidents. A review of these evaluations indicated that they were valid for use in the EIS with a few exceptions (noted below).

The evaluation used to identify internal accidents did not evaluate criticality events quantitatively (Kappes 1998, page 34). Continuing evaluations are under way to assess the probability and consequences of a criticality event. The risk from criticality events, however, would be unlikely to exceed the risk from the bounding events considered below. This preliminary conclusion is based on several factors:

- The probability of a criticality event would be very low. This is based on the Nuclear Regulatory Commission design requirement (10 CFR Part 60) that specifies that two independent low-probability events must occur for criticality to be possible and that this requirement will be part of the licensing basis for the repository. On the basis of this requirement, the event is unlikely to be credible (Jackson et al. 1984, page 18). Further, a criticality event would require the assembly of fuel with sufficient fissionable material to sustain a criticality. Since the commercial spent-nuclear fuel to be handled at the repository is spent (that is, it has been used to produce power), the remaining fissionable material is limited. For the pressurized-water reactor fuel, the amount of fuel that contains sufficient fissionable material to achieve criticality is only a small percent spent nuclear fuel (DOE 1998a, page C-46). This material would have to be assembled in sufficient quantity to achieve criticality, and the moderator (water) would somehow have to be added to the assembled material. A quantitative estimate of criticality frequency is planned as part of the license application (Kappes 1998, page 34).
- The criticality event that could occur despite the preventive measures described above would be unlikely to compromise the confinement function of the ventilation and filtration system of the Waste Handling Building. These features would inhibit the release of particulate radionuclides. By contrast, the seismic event scenario (discussed in Section H.2.1.3) assumes failure of these mitigating features.
- Criticality could occur only if the material was moderated with water and had sufficient fissionable material in a configuration that could allow criticality. The water surrounding the material would act to inhibit the release of particulate material (DOE 1994, Volume 1, Appendix D, page F-85) and, thus, would limit the source term.
- During the monitoring and closure phase of operations, water needs to enter a waste package that contains fuel with sufficient fissionable material to go critical. Water would have to flood a drift and leak into a defective waste package to cause a criticality. Such an event is considered not credible due to the lack of sufficient water sources, detection and remediation of water in-leakage, and high-quality leak proof waste packages.

Considering these factors, the criticality event does not appear to be a large potential contributor to risk.

RISK

Risk is defined as the possibility of suffering harm. It considers both the frequency (or probability) and consequences of an accident. In the scientific community, risk is usually defined and computed as the product of the frequency of an accident and the consequences that result. This is the definition of risk used in this analysis.

Rather than develop a single, overall expression of the risks associated with proposed actions, DOE usually finds it more informative in its EIS accident scenario analyses to consider a spectrum of accidents from low-probability, relatively high-consequence accidents to high-probability, low-consequence accidents. Nevertheless, risk is a valuable concept to apply in evaluating the spectrum of accident scenarios to ensure that accidents that are expected to dominate risk have been adequately considered.

Table H-1 lists the bounding accident scenarios identified in Kappes (1998, page 40). For each accident scenario, the table lists (1) the location of the accident, (2) the material at risk, or the amount of radioactive material involved in the accident, and (3) if the analysis assumed that filtration (high-efficiency particulate air filters) would be available to mitigate radioactive material releases. Filtration would be provided in most areas of the Waste Handling Building (TRW 1999b, page 41) and in the subsurface emplacement facilities (TRW 1999a, page 4-61). The Frequency column in Table H-1 lists the estimated annual frequency of the event (Kappes 1998, all). The last column indicates if the EIS analysis retained, eliminated, or adjusted details of the accident scenario.

Table H-1. Bounding internal accident scenarios for the Waste Handling Building and emplacement operations.

Location ^a	Number	Accident ^b	Material at risk ^c	Filters	Frequency	Disposition
A	1	6.9-meter drop of shipping cask	61 BWR assemblies	No	4.5×10^{-4}	Retained
A	2	6.9-meter drop of shipping cask	61 BWR assemblies	Yes	-- ^d	Eliminated
A	3	7.1-meter drop of shipping cask	26 PWR assemblies	No	6.1×10^{-4}	Retained
A	4	7.1-meter drop of shipping cask	26 PWR assemblies	Yes	--	Eliminated
A	5	4.1-meter drop of shipping cask	61 BWR assemblies	No	1.4×10^{-3}	Retained
A	6	4.1-meter drop of shipping cask	61 BWR assemblies	Yes	--	Eliminated
A	7	4.1-meter drop of shipping cask	26 PWR assemblies	No	1.9×10^{-3}	Retained
B	8	8.6-meter drop of canister	DOE high-level waste	Yes	4.2×10^{-5}	Eliminated ^e
B	9	6.3-meter drop of multiccanister overpack	N-Reactor fuel	Yes	4.5×10^{-4}	Retained
B	10	6.3-meter drop of multiccanister overpack	N-Reactor fuel	No	2.2×10^{-7}	Added ^f
C	11	5-meter drop of transfer basket	8 PWR assemblies	Yes	1.1×10^{-2}	Retained
C	12	5-meter drop of transfer basket	8 PWR assemblies	No	2.8×10^{-7}	Added ^f
C	13	7.6-meter drop of transfer basket	16 BWR assemblies	Yes	7.4×10^{-3}	Retained
C	14	7.6-meter drop of transfer basket	16 BWR assemblies	No	1.9×10^{-7}	Added ^f
D	15	6-meter vertical drop of disposal container	21 PWR assemblies	Yes	1.8×10^{-3}	Retained
D	16	6-meter vertical drop of disposal container	21 PWR assemblies	No	8.6×10^{-7}	Added ^g
D	17	2.5-meter horizontal drop of disposal container	21 PWR assemblies	Yes	3.2×10^{-4}	Eliminated ^g
E	18	Rockfall on waste package	44 BWR assemblies	No	4.2×10^{-8}	Eliminated ^h
E	19	Transporter runaway and derailment	21 PWR assemblies	Yes	1.2×10^{-7}	Retained ⁱ

- a. Location designators: A = Cask/Carrier Transport and Handling Area, B = Canister Transfer System, C = Assembly Transfer System, D = Disposal Container Handling System, E = Waste Emplacement and Subsurface Facility.
- b. To convert meters to feet, multiply by 3.2808.
- c. BWR = boiling-water reactor; PWR = pressurized-water reactor.
- d. Eliminated from evaluation because current design does not include a filter system for this area (Kappes 1998, page 40).
- e. Eliminated on the basis that it would not be a risk contributor because the N-Reactor multiccanister overpack drop (accident scenario B10) has an estimated frequency more than 10 times higher, and the N-Reactor fuel has a higher radionuclide inventory (Appendix A).
- f. These accident scenarios, involving loss of filtration, were added because they would exceed the level of credibility recommended by DOE (frequency greater than 1×10^{-7} per year) (DOE 1993, page 28). The corresponding U.S. Nuclear Regulatory Commission limit (used in Kappes 1998, page 4) is 1×10^{-6} per year. The Commission considers accidents with frequencies less than 1×10^{-6} per year to be beyond design basis events.
- g. Eliminated because it would not contribute to risk in comparison to accident scenario 15 at location D,, a higher drop event that would produce larger consequences with a higher frequency.
- h. Eliminated on the basis of low frequency, below the credible level of 1×10^{-7} .
- i. Frequency adjusted to account for the filtration system in the current design.

The following paragraphs contain details of the postulated accident scenarios in each location.

H.2.1.1.1 Cask/Carrier Transport and Handling Area

These accidents (Table H-1, location A, accidents 1 through 7) would involve mishaps that could occur during the process of handling the transportation casks at the repository. The transportation casks would be designed to withstand impacts from collisions and drops, and this capability is augmented by impact limiters, which would be required during transportation. After cask arrival at the repository, the limiters would be removed to facilitate handling of the casks. The casks would then become more vulnerable to damage from physical impact. The analysis assumed that damage to the casks would occur if they were dropped from heights greater than the design basis of 2 meters (6.6 feet) (Kappes 1998, page 13) without the impact limiters. The various heights of the drops in the "Accident" column in Table H-1 correspond to the maximum height to which the casks could be lifted during the various operations the analysis assumed crane failure would occur. The material-at-risk column lists the contents of the casks when the accident occurred. The largest casks are designed to hold either 61 boiling-water reactor or 26 pressurized-water reactor fuel assemblies.

Accident scenarios from Kappes (1998) that assume a filtration system is available (accidents A2, A4, and A6) were eliminated from consideration in the EIS because the current design concept of the Cask/Carrier Transport and Handling Area does not include such a filtration system; they were considered in Kappes (1998, page 40) for information only.

H.2.1.1.2 Canister Transfer System

The Canister Transfer System would handle canisters that arrived at the repository and were suitable for direct transfer to the disposal container. The bounding accident scenarios for these operations would be canister drops of DOE high-level radioactive waste and N-Reactor fuel (accidents 8 and 9 at location B in Table H-1). The analysis eliminated the DOE high-level radioactive waste canister drop because it would not be a risk contributor in comparison to the N-Reactor fuel drop. The N-Reactor multiccanister overpack drop would have a frequency more than 10 times greater than that for the high-level radioactive waste canister drop, and the N-Reactor radionuclide inventory would be greater (see Appendix A). The EIS analysis added an additional accident scenario, which would be a drop of the N-Reactor fuel canister with loss of the filtration system. The analysis estimated the filtration system failure probabilities by using the fault tree analysis technique, and the results differ somewhat from the failures identified in Section H.2.1.1.3 due to design variations dependant on location in the surface facilities of the repository. DOE computed this accident scenario probability by combining the accident drop probability of 0.00045 with the filter system failure of 4.8×10^{-4} from Kappes (1998, page 4) for an accident sequence frequency of 2.2×10^{-7} per year. [Kappes (1998, page 4) did not consider accident sequences with frequencies less than 1×10^{-6} .] This sequence frequency is based on failure of the heating, ventilating, and air conditioning system such that it would not provide filtration for 24 hours following the accident, consistent with Kappes (1998, page VIII-1).

H.2.1.1.3 Assembly Transfer System

The Assembly Transfer System would handle bare, intact commercial spent nuclear fuel assemblies from pressurized- and boiling-water reactors. The assemblies would be unloaded from the transportation cask in the cask unloading pool. Next, they would be moved to the assembly staging pool where they would be placed in baskets that contained either four pressurized-water reactor assemblies or eight boiling-water assemblies. The baskets would be moved from the pool and transferred to the assembly drying station from which they would be loaded, after drying, in the disposal containers. The bounding accident scenarios found during a review of this operation (Kappes 1998, page 40) were drops of a suspended basket loaded with fuel assemblies on another loaded basket in the drying vessel (accident scenarios 11 and 13 at location C from Table H-1). DOE added two accident scenarios to the EIS analysis that

included failure of the high-efficiency particulate air filtration system (accident scenarios 12 and 14 at location C from Table H-1). DOE computed the frequency of these accidents by combining the accident drop frequency with the filter failure probability of 0.000025, which corresponds to the failure probability of the heating, ventilation, and air conditioning system in the assembly transfer area (Kappes 1998, page 11). Thus, the frequency of a drop accident and subsequent failure of the heating, ventilation, and air conditioning system during the 24 hours (the period assumed that the filtration system would need to operate to remove the particulate material effectively) would be:

- For boiling-water reactor assembly drop: $0.011 \times 0.000025 = 0.00000028$
- For pressurized-water reactor assembly drop: $0.0074 \times 0.000025 = 0.00000019$

H.2.1.1.4 Disposal Container Handling System

The Disposal Container Handling System would prepare empty disposal containers for the loading of nuclear materials, transfer disposal containers to and from the assembly and canister transfer systems, weld the inner and outer lids of the disposal containers, and load disposal containers on the waste emplacement transporter. After the disposal container had been loaded and sealed, it would become a waste package. Disposal containers would be lifted and moved several times during the process of preparing them for loading on the waste emplacement transporter. DOE examined the details of these operations and identified numerous accident scenarios that could occur (Kappes 1998, Attachment V). The bounding accident scenarios from this examination would be the disposal container drop accident scenarios listed as accident scenarios 15 and 17 at Location D in Table H-1. However, the analysis eliminated accident scenario 17 because it would be a minor contributor to risk in comparison to accident scenario 15. Accident scenario 15, which would have a higher probability (by about a factor of 6), would produce a higher radionuclide release due to the increased drop height (by a factor of more than 2). Thus, the overall risk contribution from accident scenario 17 would be less than 10 percent of the risk from accident scenario 15. For the EIS, DOE added another accident scenario (16) to account for the possibility of loss of filtration. The analysis assumed that the heating, ventilation, and air conditioning filtration system would fail with a probability of 0.00048 (Kappes 1998, page 4).

H.2.1.1.5 Waste Emplacement and Subsurface Facility Systems

The waste emplacement system would transport the loaded and sealed waste package from the Waste Handling Building to the subsurface emplacement area. This system would operate on the surface between the North Portal and the Waste Handling Building, and in the underground ramps, main drifts (tunnels), and emplacement drifts. It would use a reusable railcar for waste package transportation. The railcar would be moved into the waste emplacement area by an electric locomotive, and the waste package would be placed in the emplacement drift. The bounding accident scenarios identified (Kappes 1998, page 40) for this operation would be accident scenarios 18 and 19 at location E, as listed in Table H-1. However, DOE eliminated accident scenario 18 (rockfall on waste package) because the estimated frequency of a radioactive release from such an event is not credible (estimated frequency of 4.2×10^{-8} per year) (Kappes 1998, page VI-5).

An accident scenario involving a failure of the ventilation system in conjunction with a transporter runaway and collision (accident scenario F19 from Table H-1) would not be credible, so the sequence was not analyzed. The original transporter runaway and derailment accident scenario assumed the involvement of 44 boiling-water reactor assemblies (Kappes 1998, page 40). The EIS analysis assumed the involvement of 21 pressurized-water reactor assemblies because (1) they would represent a slightly higher impact potential due to the greater radionuclide inventory than that in the smaller 44 boiling-water reactor assemblies and would, therefore, bound the equivalent accident involving such assemblies, and

(2) an accident scenario involving pressurized-water reactor fuel would be more likely because DOE expects to emplace about twice as much of this type of fuel in the proposed repository (Appendix A).

Section H.2.1.4 describes the source terms (amount and type of radionuclide release) for these accident scenarios, and Section H.2.1.5 assesses the estimated consequences from the accident scenarios.

H.2.1.2 Internal Events – Waste Treatment Building

An additional source of radionuclides could be involved in accidents in the Waste Treatment Building. This building, which would be connected to the northeast end of the Waste Handling Building, would house the Site-Generated Radiological Waste Handling System (TRW 1999b, page 37). This system would collect site-generated low-level radioactive solid and liquid wastes and prepare them for disposal. The radioactivity of the waste streams would be low enough that no special features would be required to meet Nuclear Regulatory Commission radiation safety requirements (shielding and criticality) (TRW 1999b, page 38).

The liquid waste stream to the Waste Treatment Building would consist of aqueous solutions that could contain radionuclides resulting from decontamination and washdown activities in the Waste Handling Building. The liquid waste would be evaporated, mixed with cement (grouted), and placed in 0.21-cubic-meter (55-gallon) drums for shipment off the site (TRW 1999b, page 53). The evaporation process would reduce the volume of the liquid waste stream by 90 percent (DOE 1997c, Summary).

The solid waste would consist of noncompactible and compactible materials and spent ion-exchange resins. These materials ultimately would be encapsulated in concrete in 0.21-cubic meter (55-gallon) drums after appropriate processing (TRW 1999b, page 55).

Water in the Assembly Staging Pools of the Waste Handling Building would pass through ion exchange columns to remove radionuclides and other contaminants. These columns would accumulate radionuclides on the resin in the columns. When the resin is spent (unable to remove radionuclides effectively from the water), the water flow would be diverted to another set of columns, and the spent resin would be removed and dewatered for disposal as low-level waste or low-level mixed waste. These columns could have external radiation dose rates associated with them because of the activation and fission product radionuclides accumulated on the resins. They would be handled remotely or semiremotely. During the removal of the resin and preparation for offsite shipment in the Waste Treatment Building, an accident scenario involving a resin spill could occur. However, because the radionuclides would have been chemically bound to the resin in the column, an airborne radionuclide release would be unlikely. Containment and filter systems in the Waste Treatment Building would prevent exposure to the public or noninvolved workers. Some slight exposure of involved workers could occur during the event or during recovery operations afterward. DOE made no further analysis of this event.

Because there is no detailed design of the Waste Treatment Building at present and operational details are not yet available, DOE used the recent Waste Management Programmatic EIS (DOE 1997c, all) and supporting documentation (Mueller et al. 1996, all) to aid in identifying potential accident scenarios and evaluating radionuclide source terms. For radiological impacts, the analysis focused on accident scenarios with the potential for airborne releases to the atmosphere. The liquid stream can be eliminated because it has a very low potential for airborne release; the radionuclides would be dissolved and energy sources would not be available to disperse large amounts of the liquid into droplets small enough to remain airborne. Many low-level waste treatment operations, including evaporation, solidifying (grouting), packaging, and compaction can be excluded because they would lack sufficient mechanistic stresses and energies to create large airborne releases, and nuclear criticalities would not be credible for

low-level waste (Mueller et al. 1996, page 13). Drum-handling accidents are expected to dominate the risk of exposure to workers (Mueller et al. 1996, page 93).

The estimated frequency of an accident involving drum failure is about 0.0001 failure per drum operation (Mueller et al. 1996, page 39). The total number of drums containing grouted aqueous waste would be 2,280 per year (DOE 1997c, page 30). The analysis assumed that each drum would be handled twice, once from the Waste Treatment Building to the loading area, and once to load the drum for offsite transportation. Therefore, the frequency of a drum failure involving grouted aqueous waste would be:

$$\begin{aligned} \text{Frequency} &= 2,280 \text{ aqueous (grouted) low-level waste drums per year} \\ &\quad \times 2 \text{ handling operations per drum} \\ &\quad \times 0.0001 \text{ failure per handling operation} \\ &= 0.46 \text{ aqueous (grouted) low-level waste drum failures per year.} \end{aligned}$$

The number of solid-waste grouted drums produced would be 2,930 per year (DOE 1997c, page 35). Assuming two handling operations and the same failure rate yields a frequency of drum failure of:

$$\begin{aligned} \text{Frequency} &= 2,930 \text{ solid low-level waste drums per year} \\ &\quad \times 2 \text{ handling operations per drum} \\ &\quad \times 0.0001 \text{ failure per handling operation} \\ &= 0.59 \text{ solid low-level waste drum failures per year.} \end{aligned}$$

Failure of these drums would result in a release of radioactive material, which later sections evaluate further.

H.2.1.3 External Events

External events are either external to the repository (earthquakes, high winds, etc.) or are natural processes that occur over a long period of time (corrosion, erosion, etc.). DOE performed an evaluation to identify which of these events could initiate accidents at the repository with potential for release of radioactive material.

Because some external events evaluated as potential accident-initiating events would affect both the Waste Treatment and Waste Handling Buildings simultaneously [the buildings are physically connected (TRW 1999b, page 38)], this section considers potential accidents involving external event initiators, as appropriate, for the combined buildings.

Table H-2 lists generic external events developed as potential accident initiators for consideration at the proposed repository and indicates how each potential event could relate to repository operations based on an initial evaluation process. The list, from DOE (1996b, page 15), was developed by an extensive review of relevant sources and known or predicted geologic, seismologic, hydrologic, and other characteristics. The list includes external events from natural phenomena as well as man-caused events.

The center column in Table H-2 (relation to repository) represents the results of a preliminary evaluation to determine the applicability of the event to the repository operations, and is based in part on evaluations previously reported in DOE (1996b, all). Events were excluded for the following reasons:

- Not applicable because of site location (condition does not exist at the site)
- Not applicable because of site characteristics (potential initiator does not exist in the vicinity of the site)

Table H-2. External events evaluated as potential accident initiators.^a

Event	Relation to repository ^b	Comment
Aircraft crash	A	
Avalanche	C	
Coastal erosion	B	
Dam failure	C	
Debris avalanche	A	Caused by excessive rainfall
Dissolution	A	Chemical weathering of rock
Epeirogenic displacement (tilting of the Earth's crust)	D (earthquake)	Large-scale surface uplifting and subsidence
Erosion	D (flooding)	
Extreme wind	A	
Extreme weather	A	Includes extreme episodes of fog, frost, hail, ice cover, etc.
Fire (range)	A	
Flooding	A	
Denudation	E	Wearing away of ground surface by weathering
Fungus, bacteria, algae	E	A potential waste package long-term corrosion process not relevant during the repository operational period ^c
Glacial erosion	B	
High lake level	C	
High tide	B	
High river stage	C	
Hurricane	B	
Inadvertent future intrusion	E	To be addressed in postclosure Performance Assessment
Industrial activity	A	
Intentional future intrusion	E	
Lightning	A	
Loss of offsite or onsite power	A	
Low lake level	C	
Meteorite impact	A	
Military activity	A	
Orogenic diastrophism	D (earthquake)	Movement of Earth's crust by tectonic processes
Pipeline rupture	C	
Rainstorm	D (flooding)	
Sandstorm	A	
Sedimentation	B	
Seiche	B	Surface water waves in lakes, bays, or harbors
Seismic activity, uplift	D (earthquake)	
Seismic activity, earthquake	A	
Seismic activity, surface fault	D (earthquake)	
Seismic activity, subsurface fault	D (earthquake)	
Static fracture	D (earthquake)	Rock breakup caused by stress
Stream erosion	B	
Subsidence	D (earthquake)	Sinking of Earth's surface
Tornado	A	
Tsunami	B	Sea wave caused by ocean floor disturbance
Undetected past intrusions	E	
Undetected geologic features	D (earthquake, volcanism ash fall)	
Undetected geologic processes	D (erosion, earthquake, volcanism ash fall)	
Volcanic eruption	D (volcanism ash fall)	
Volcanism, magmatic	D (volcanism ash fall)	
Volcanism, ash flow	D (volcanism ash fall)	
Volcanism, ash fall	A	
Waves (aquatic)	B	

a. Source: DOE (1996b, page 15).

b. A = retained for further evaluation; B = not applicable because of site location; C = not applicable because of site characteristics (threat of event does not exist in the vicinity of the site); D = included in another event as noted; E = does not represent an accident-initiating event for proposed repository operations.

c. Source: TRW (1999a, all).

- Included in another event
- Does not represent an accident-initiating event for proposed repository operations

The second column of Table H-2 identifies the events excluded for these reasons. The preliminary evaluation retained the events identified in Table H-2 with "A" for further detailed evaluation. The results of this evaluation are as follows:

- 1. Aircraft Crash.** The EIS analysis evaluated the frequency of aircraft crashes on the proposed repository to determine if such events could be credible and, therefore, candidates for consequence analysis. This frequency determination used analytical methods recommended for aircraft crashes into hazardous facilities (DOE 1996c, all).

An earlier analysis assumed that the only reasonable aircraft crash threat would be from military aircraft operations originating from Nellis Air Force Base (Kimura, Sanzo, and Sharirli 1998, page 8), primarily because commercial and general aviation aircraft are restricted from flying over the Nevada Test Site. DOE considered this assumption valid and adopted it for the EIS analysis.

The formula used in the crash frequency analysis, taken from Kimura, Sanzo, and Sharirli (1998, pages 9 to 12) based on DOE (1996c, all), was:

$$F = (N_t \div A_t) \times A_{\text{eff}} \times \lambda \times (4 \div \pi) \times (R_{\text{eff}} + R_c)$$

where:

- F = the frequency per year of aircraft crashes on the repository
- N_t = total number of aircraft overflights per year
- A_t = total area of the overflight region
- A_{eff} = effective area of the repository (target area)
- λ = crash rate of the aircraft per mile of flight
- R_{eff} = effective radius of the repository (target area)
- R_c = radius of the crash area potentially affected by a distressed aircraft

The parameters in this formula were quantified as follows:

N_t The estimated total number of flights in the flight corridor in the vicinity of the repository would be 13,000 per year, with the repository located on the western edge of the corridor, which extends 49 kilometers (30 miles) to the east. Most flights would not be observed from the repository. However, this value was used in a recent crash assessment for a Nevada Test Site facility beneath the same airspace as the repository (Kimura, Sanzo, and Sharirli 1998, page 7). Future Nellis operations could result in increased overflights. The only known planned change in future activities involve the bed-down of F-22 fighter aircraft. This planned activity involves 17 aircraft that will be at Nellis by 2010. The additional aircraft would increase flight activities by only 2 to 3 percent over current activities (Myers 1997, page 2).

A_t The total area of the overflight area would be about 3,400 square kilometers (1,300 square miles) (Kimura, Sanzo, and Sharirli 1998, page 18).

A_{eff} The analysis estimated the repository target area by assuming that the roof of the Waste Handling Building would be the only vulnerable location at the repository with the potential for a large radionuclide release as a result of an aircraft impact. This is because the Waste Handling Building would be the only facility that would handle bare spent nuclear fuel assemblies. The shipping casks and the waste packages loaded with spent nuclear fuel or high-level radioactive waste would not be vulnerable to air crash impacts because both would have steel walls thick enough to prevent aircraft penetration. The Waste Treatment Building would not contain large amounts of radioactive material, so radionuclide releases from accidents involving this building would not produce large impacts (see Section H.2.1.4 for details). Further, the walls of the Waste Handling Building around areas for the handling of canisters and fuel assemblies would be 1.5 meters (5 feet) thick to a level of 9 meters (30 feet), and then 1 meter (3.3 feet) thick to the intersection with the roof (TRW 1999b, pages 31 to 37). The aircraft crash would not penetrate these walls because the concrete penetration capability for aircraft is limited to about 0.76 meter (2.5 feet) (see Appendix K for details). Therefore, the only likely vulnerable target area at the repository would be the roof of the Waste Handling Building, which would consist of concrete 20 to 25 centimeters (8 to 10 inches thick) (TRW 1999b, pages 31 to 37). The overall footprint of the Waste Handling Building would be about 163 meters by 165 meters (535 feet by 540 feet), which would produce a target area of approximately 27,000 square meters (290,000 square feet).

λ The crash rate for the small military aircraft involved in the overflights [primarily F-15s, F-16s, and A-10s (USAF 1999, pages 1-34 to 1-35)] would be 1.14×10^{-8} per kilometer (1.84×10^{-8} per mile) (Kimura, Sanzo, and Sharirli 1998, page 7). Large military aircraft fly over the area to some extent, but have a lower crash rate [1.17×10^{-9} per kilometer (1.9×10^{-9} per mile) (Kimura, Sanzo, and Sharirli 1998, page 7)]. Thus, the use of the small aircraft crash rate bounds the large aircraft crash rate.

R_{eff} The effective radius of the repository is the equivalent radius of the repository target effective area (A_{eff}), or R_{eff} is equal to the square root of the quotient 27,000 square meters divided by π , which is about 93 meters (310 feet).

R_c The radius of the crash area potentially affected by a distressed military aircraft represents the distance an aircraft could travel after engine failure (glide distance). This distance is the glide ratio of the aircraft times the elevation of the flight above the ground. The aircraft are required to fly a minimum of 4,300 meters (14,000 feet) above mean sea level while in the airspace over the repository (Kimura, Sanzo, and Sharirli 1998, page 5). The actual altitude flown varies from 4,600 to 7,000 meters (15,000 to 23,000 feet) (Tullman 1997, page 4). For this analysis, a mean altitude of 5,800 meters (19,000 feet) was assumed. Because the Waste Handling Building would be at about 1,100 meters (3,680 feet) (TRW 1998a, page I-6), the mean flight elevation for aircraft above the repository ground level would be about 4,700 meters (10,000 feet). The glide ratio for the aircraft involved in the overflights (F-15, F-16, and A-10) is 8 (Thompson 1998, all). Therefore, R_c would be 4,700 meters multiplied by 8, which is 38,000 meters or 38 kilometers (23 miles).

Substituting these values into the frequency equation yields:

$$\begin{aligned} F &= (13,000 \div 3,400) \times 0.027 \times 1.14 \times 10^{-8} \times (4 \div \pi) \times (38 + 0.093) \\ &= 5.6 \times 10^{-8} \text{ crash per year.} \end{aligned}$$

Thus, aircraft crashes on the vulnerable area of the repository are not credible because the probability would be below 1×10^{-7} per year, which is the credible limit specified by DOE (1993, page 28).

2. **Debris Avalanche.** This event, which can result from persistent rainfall, would involve the sudden and rapid movement of soil and rock down a steep slope. The nearest avalanche potential to the proposed location for the Waste Handling Building is Exile Hill (the location of the North Portal entrance). The base of Exile Hill is about 90 meters (300 feet) from the location of the Waste Handling Building. Since Exile Hill is only about 30 meters (100 feet) high (TRW 1997a, page 5.09), it would be unlikely that avalanche debris would reach the Waste Handling Building. Furthermore, the design for the Waste Handling Building includes concrete walls about 1.5 meters (5 feet) thick (TRW 1999b, page 38) that would provide considerable resistance to an impact or buildup of avalanche debris.
3. **Dissolution.** Chemical weathering could cause mineral and rock material to pass into solution. This process, called dissolution, has been identified as potentially applicable to Yucca Mountain (DOE 1996b, page 18). However, this is a very slow process, which would not represent an accident-initiating event during the preclosure period being considered in this appendix.
4. **Extreme Wind.** Extreme wind conditions could cause transporter derailment (TRW 1997b, page 72), the consequences of which would be bounded by a transporter runaway accident scenario. The runaway transporter accident scenario is discussed further in Section H.2.1.4.
5. **Extreme Weather.** This potential initiating event includes various weather-related phenomena including fog, frost, hail, drought, extreme temperatures, rapid thaws, ice cover, snow, etc. None of these events would have the potential to cause damage to the Waste Handling Building that would exceed the projected damage from the earthquake event discussed in this section. In addition, none of these events would compromise the integrity of waste packages exposed on the surface during transport operations. Thus, the earthquake event and other waste package damage accident scenarios considered in this appendix would bound all extreme weather events. It would also be expected that operations would be curtailed if extreme weather conditions were predicted.
6. **Fire.** There would be two potential external fire sources at the repository site—diesel fuel oil storage tank fires and range fires. Diesel fuel oil storage tanks would be some distance [more than 90 meters (300 feet)] from the Waste Handling Building and Waste Treatment Building (TRW 1999b, Attachment IV Figure 4). Therefore, a fire at those locations would be highly unlikely to result in any meaningful radiological consequences. Range fires could occur in the vicinity of the site, but would be unlikely to be important accident contributors due to the clearing of land around the repository facilities. Furthermore, the potential for early fire detection and, if necessary, active fire protection measures and curtailment of operations (TRW 1999b, Section 4.2) would minimize the potential for fire-initiated radiological accidents. DOE is performing detailed evaluations of fire-initiating events (Kappes 1998, page III-2), and will incorporate the results in the Final EIS as appropriate.
7. **Flooding.** Flash floods could occur in the vicinity of the repository (DOE 1996b, page 21). However, an earlier assessment (Kappes 1998, page 32) screened out severe weather events as potential accident-initiating events primarily by assuming that operational rules will preclude transport and emplacement operations whenever there are local forecasts of severe weather. A quantitative analysis of flood events (Jackson et al. 1984, page 34) concluded that the only radioactive material that extreme flooding would disperse to the environment would be decontamination sludge from the waste treatment complex. The doses resulting from such dispersion would be limited to workers, and would be very small (Jackson et al. 1984, page 53). A more recent study reached a similar conclusion (Ma et al. 1992, page 3-11).
8. **Industrial Activity.** This activity would involve both drift (tunnel) development activities at the repository and offsite activities that could impose hazards on the repository.

- a. **Emplacement Drift Development Activities** – Drift development would continue during waste package emplacement activities. However, physical barriers in the main drifts would isolate development activities from emplacement activities (TRW 1999a, page 4-52). Thus, events that could occur during drift development activities would be unlikely to affect the integrity of waste packages.
 - b. **External Industrial Activities** – The analysis examined anticipated activities in the vicinity of the proposed repository to determine if accident-initiating events could occur. Two such activities—the Kistler Aerospace activities and the Wahmonie rocket launch facility—could initiate accidents at the repository from rocket impacts. The Wahmonie activities, which involved rocket launches from a location several miles east of the repository site, have ended (Wade 1998, all), so this facility no longer poses a risk to the repository. The planned Kistler Aerospace activities would involve launching rockets from the Nevada Test Site to place satellites in orbit (DOE 1996d, Volume 1, page A-42). However, at present there is insufficient information to assess if this activity would pose a threat to the repository. As details become available, the Final EIS will evaluate the potential for this activity to become an external accident-initiating event. (Aircraft activity was discussed in item 1 above.)
- 9. Lightning.** This event has been identified as a potential design-basis event (DOE 1997b, pages 86 and 87). Therefore, the analysis assumed that the designs of appropriate repository structures and transport vehicles would include protection against lightning strikes. The lightning strike of principal concern would be the strike of a transporter train during operations between the Waste Handling Building and the North Portal (DOE 1997b, page 86). The estimated frequency of such an event would be 1.9×10^{-7} per year (Kappes 1998, page 33). DOE expects to provide lightning protection for the transporter (TRW 1998b, Volume 1, page 18) such that a lightning strike that resulted in enough damage to cause a release would be well below the credibility level of 1×10^{-7} per year (DOE 1993, page 28).
- 10. Loss of Offsite Power.** A preliminary evaluation (DOE 1997b, page 84) concluded that a radionuclide release from an accident sequence initiated by a loss of offsite power would be unlikely. Loss of offsite power events could result in a failure of the ventilation system and of the overhead crane system. However, there would be emergency power for safety systems at the site (TRW 1999b, page 45). Loss of offsite power was included as a contributor to the frequency of crane failure (Kappes 1998, page III-6), as listed in the event frequencies in Table H-1.
- 11. Meteorite Impact.** This event would not be credible based on a strike frequency of 2×10^{-8} per year for a damaging meteorite [based on data in Solomon, Erdmann, and Okrent (1975, page 68)]. This estimate accounts for the actual area of the Waste Handling Building roof given previously in item 1.
- 12. Military Activity.** Two different military activities would have the potential to affect repository operations. One is the possibility of an aircraft crash from overflights from Nellis Air Force Base. The analysis determined that this event would not be credible, as described above in this section. The second potential activity is the resumption of underground nuclear weapons testing, which the United States has suspended. The only impact such testing could impose on the repository would be ground motion associated with the energy released from the detonation of the weapon. The impact of such motion was the subject of a recent study that concluded that ground motions at Yucca Mountain from nuclear tests would not control seismic design criteria for the potential repository (Walck 1996, page i).

13. Sandstorm. Severe sandstorms could cause transporter derailments and sand buildup on structures. However, such events would be unlikely to initiate accidents with the potential for radiological release. Ma et al. (1992, page 3-11) reached a similar conclusion. Furthermore, it is assumed that DOE probably would curtail operations if local forecasts indicated the expected onset of high winds with potential to generate sandstorms (Kappes 1998, page 32). For these reasons, the analysis eliminated this event from further consideration.

14. Seismic Activity, Earthquake (including subsidence, surface faults, uplift, subsurface fault, and static fracture). DOE has selected the beyond-design-basis earthquake for detailed analysis. The seismic design basis for the repository specifies that structures (including the Waste Handling Building), systems, and components important to safety should be able to withstand the horizontal motion from an earthquake with a return frequency of once in 10,000 years (annual probability of occurrence of 0.0001) (Kappes 1998, page VII-3). A recent comprehensive evaluation of the seismic hazards associated with the site of the proposed repository (USGS 1998, all) concluded that a 0.0001-per-year earthquake would produce peak horizontal accelerations at the site of about 0.53g (mean value). Structures, systems, and components are typically designed with large margins over the seismic design basis to account for uncertainties in material properties, energy absorption, damping, and other factors. For nuclear powerplant structures, the methods for seismic design provide a factor of safety of 2.5 to 6 (Kennedy and Ravindra 1984, page R-53). In the absence of detailed design information, the analysis conservatively assumed that the Waste Handling Building would collapse at an acceleration level twice that associated with the design-basis earthquake, or 1.1g. Figure H-1 shows that this acceleration level would be likely to occur with a frequency of about 2×10^{-5} per year (mean value).

The Waste Treatment Building is not considered a safety-related structure. Accordingly, the seismic design basis for this building is to withstand an earthquake event with a return frequency of 1,000 years (annual exceedance probability of 1×10^{-3} per year) (TRW 1999b, page 14). Consistent with the assumption for the Waste Handling Building, it is assumed that the Waste Treatment Building would collapse during an earthquake that produced twice the design level acceleration. From Figure H-1, the design-basis acceleration for a 1×10^{-3} per year event is 0.18g. Thus, the building collapse is assumed to occur at an acceleration level of 0.36, which has an estimated return frequency of about 2×10^{-4} per year. The analysis retains these events as accident initiators, and evaluates the consequences in subsequent sections. The effects of other seismic-related phenomena included under this event (subsidence, surface faults, uplift, etc.) would be unlikely to produce greater consequences than those associated with the acceleration produced by the seismic event selected for analysis (complete collapse of the Waste Handling and Waste Treatment Buildings).

15. Tornado. The probability of a tornado striking the repository is estimated to be 3×10^{-7} (one chance in 10 million) based on an assessment of tornado strike probability for any point on the Nevada Test Site (DOE 1996d, page 4-146), which is adjacent to the proposed repository. This is slightly above the credibility level of 1×10^{-7} for accidents, as defined by DOE (DOE 1993, page 28). However, most tornadoes in the western United States have relatively modest wind speeds. For example, the probability of a tornado with wind speeds greater than 100 miles per hour is 0.1 or less (Ramsdell and Andrews 1986, page 41). Thus, winds strong enough to damage the Waste Handling Building are considered to be not credible.

Tornadoes can generate missiles that could penetrate structures at the repository, but radioactive material would be protected either by shipping casks, the Waste Handling Building with thick concrete walls, or the waste package. Therefore, tornado-driven missiles would not be a great hazard.

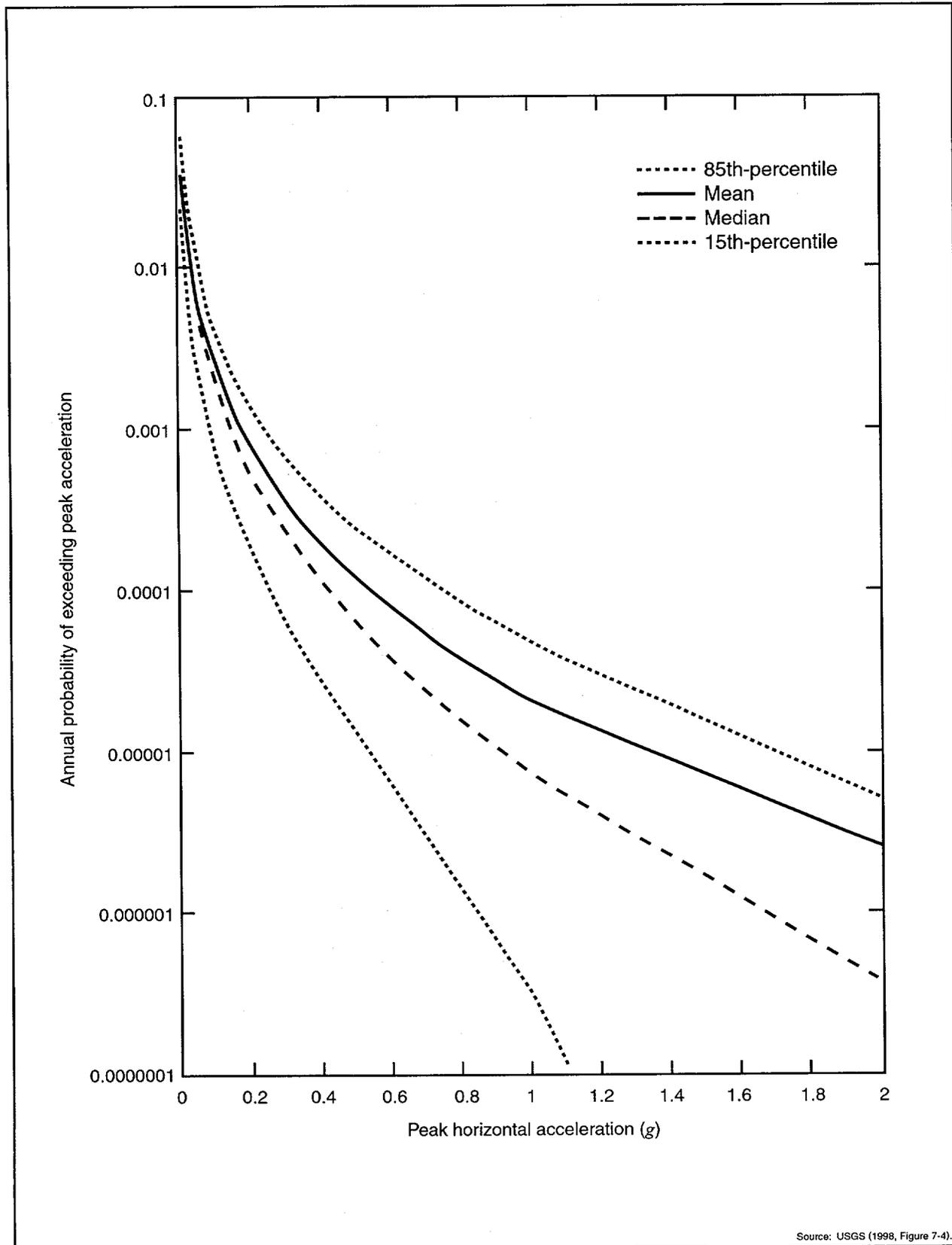


Figure H-1. Integrated seismic hazard results: summary hazard curves for peak horizontal acceleration.

- 16. Volcanism, Ash Fall.** The potential for volcanic activity at the proposed repository site has been studied extensively. A recent assessment (Geomatrix and TRW 1996, page 4-46) estimates that the mean annual frequency of a volcano event that would intersect the repository footprint would be 1.5×10^{-8} per year (with 5 percent and 95 percent bounds of 5×10^{-10} and 5×10^{-8} per year), which is below the frequency of a credible event (DOE 1993, page 28). This result is consistent with a previous study of volcano activity at the site (DOE 1998b, all). Impacts from a regional volcanic eruption would be more likely; such an event could produce ash fall on the repository, and would be similar to the sandstorm event discussed above. Ash fall could produce a very heavy loading on the roof of the Waste Handling Building. Studies have concluded, however, that the worst case event would be an ash fall of 3 centimeters (1.2 inches) and analyses to date indicate that repository structures would not be affected by a 3-centimeter ash fall (DOE 1998b, Volume 1, pages 2-9).
- 17. Sabotage.** The analysis separately considered sabotage (not listed in Table H-2) as a potential initiating event. This event would be unlikely to contribute to impacts from the repository. The repository would not represent an attractive target to potential saboteurs due to its remote location and the low population density in the area. Furthermore, security measures DOE would use to protect the waste material from intrusion and sabotage (TRW 1999b, pages 58 to 60) would make such attempts unlikely to succeed. At all times the waste material would be either in robust shipping or disposal containers or inside the Waste Handling Building, which would have thick concrete walls. On the basis of these considerations, DOE concluded that sabotage events would be unlikely at the repository. DOE expects that both the likelihood and consequences of sabotage events would be greater during transportation of the material to the repository (DOE 1997d, page 14). Appendix J presents the impacts of sabotage events during transportation.

Based on the external event assessment, DOE concluded that the only external event with a credible potential to release radionuclides of concern would be a large seismic event. This conclusion is supported by previous studies that screened out all external event accident initiators except seismic events (Ma et al. 1992, page 3-11; Jackson et al. 1984, pages 12 and 13). DOE is continuing to evaluate a few external events (Kappes 1998, page 33), and will examine the results of these evaluations to confirm the Draft EIS conclusions. If revisions are necessary, they will be provided in the Final EIS.

H.2.1.4 Source Terms for Repository Accident Scenarios

Following the definition of the accident scenarios as provided in previous sections, the analysis then estimated a source term for each accident scenario retained for analysis. The source term specification needed to include several factors, including the quantity of radionuclides released, the elevation of the release, the chemical and physical forms of the released radionuclides, and the energy (if any) of the plume that would carry the radionuclides to the environment. These factors would be influenced by the state of the material involved in the accident and the extent and type of damage estimated for the accident sequence. The estimate of the source term also considered mitigation measures, either active (for example, filtration systems) or passive (for example, local deposition of radionuclides or containment), that would reduce the amount of radioactive material released to the environment.

The analysis developed the source term for each accident scenario retained for evaluation. These include the accident scenarios retained from the internal events as listed in Table H-1 and the seismic event retained from the external event evaluation. Because many of the internal event-initiated accidents would involve drops of commercial spent nuclear fuel, the analysis considered the source term for these accidents as a group. Accordingly, source terms were developed for the following accident scenarios: commercial spent nuclear fuel drops, transporter runaway and derailment, DOE spent nuclear fuel drop, seismic event, and low-level waste drum failure.

H.2.1.4.1 Commercial Spent Nuclear Fuel Drop Accident Scenario Source Term

Commercial spent nuclear fuel contains more than 100 radioactive isotopes (SNL 1987, Appendix A). Not all of these isotopes, however, would be important in terms of a potential to cause adverse health effects (radiotoxicity) if released, and many would have decayed by the time the material arrived at the repository. Based on the characteristics of the radioactivity associated with an isotope (including type and energy of radioactive emissions, amount produced during the fissioning process, half-life, physical and chemical form, and biological impact if inhaled or ingested by a human), particular isotopes could be meaningful contributors to health effects if released. To determine the important radionuclides for an accident scenario consequence analysis, DOE consulted several sources. The Nuclear Regulatory Commission has identified a minimum of eight radionuclides in commercial spent nuclear fuel that “must be analyzed for potential accident release” (NRC 1997, page 7-6). Repository accident scenario evaluations (SNL 1987, pages 5-3 and 5-4) identified 14 isotopes (five of which were also on the Nuclear Regulatory Commission list) that contribute to “99 percent of the total dose consequence.” A more recent analysis (DOE 1996a, pages 6 to 9) lists 24 radionuclides (10 of which were not included in either of the other two lists) that are important for consequence analysis (99.9-percent cumulative dose for at least one organ). The DOE analysis also included carbon-14. Appendix A contains a list of 53 radionuclides, which includes the important isotopes discussed above. DOE used this longer list in the development of the source term for the accident scenario analyses.

Commercial spent nuclear fuel includes two primary types—boiling-water reactor and pressurized-water reactor spent fuel. For these commercial fuels, the radionuclide inventory depends on burnup (power history of the fuel) and cooling time (time since removal from the reactor). The EIS accident scenario analysis used “typical” fuels for each type. These typical fuels are representative of the majority of the fuel DOE would receive at the repository (see Appendix A). Table H-3 lists the characteristics of typical commercial spent nuclear fuel types.

A recent sensitivity study examined the consequences from accident scenarios involving bounding fuel types and illustrated the adequacy of selecting typical fuel types for this accident scenario analysis. Table H-4 lists the radionuclide inventory selected for estimating the accident scenario consequences for the fuel types selected (typical boiling-water reactor and pressurized-water reactor).

Commercial spent nuclear fuel damaged in the accidents evaluated in this EIS could release radionuclides from three different sources. These sources, and a best estimate of the release potential, are as follows:

H.2.1.4.1.1 Crud. During reactor operation, crud (corrosion material) builds up on the outside of the fuel rod cladding and becomes radioactive from neutron activation. Five years after discharge from the reactor (the minimum age of any commercial spent nuclear fuel for acceptance at the repository), the dominant radioactive constituent in the crud is cobalt-60, which accounts for 98 percent of the activity (Sandoval et al. 1991, page 15). Cobalt-60 concentration measurements have been made on several boiling-water and pressurized-water reactor fuel rods; the results indicate that the maximum activity density is 0.0000094 curie per square centimeter for pressurized-water reactors and 0.000477 curie per square centimeter for boiling-water reactors (Sandoval et al. 1991, pages 14 and 15). The maximum values are about twice the average value over the length of the fuel rod (Sandoval et al. 1991, page 14). Accordingly, the values used in these source term determinations were 0.00005 for pressurized-water

Table H-3. Typical commercial spent nuclear fuel characteristics.^a

Fuel type ^b	Cooling time (years)	Burnup (GWd/MTHM) ^c
PWR typical	25.9	39.56
BWR typical	27.2	32.2

a. Source: Appendix A.

b. PWR = pressurized-water reactor; BWR = boiling-water reactor.

c. GWd/MTHM = gigawatt-days per metric ton of heavy metal.

Table H-4. Inventory used for typical reactor fuel (curies per assembly).^{a,b}

Isotope	Pressurized-water reactor	Boiling-water reactor
Hydrogen-3	9.8×10^1	3.4×10^1
Carbon-14	6.4×10^{-1}	3.0×10^{-1}
Chlorine-36	5.4×10^{-3}	2.2×10^{-3}
Cobalt-60 ^c	1.4×10^1	2.0×10^1
Nickel-59	1.3	3.5×10^{-1}
Nickel-63	1.8×10^2	4.6×10^1
Selenium-79	2.3×10^{-1}	7.9×10^{-2}
Krypton-85	9.3×10^2	2.9×10^2
Strontium-90	2.1×10^4	7.1×10^3
Zirconium-93	1.2	4.8×10^{-1}
Niobium-93m	8.2×10^{-1}	3.5×10^{-1}
Niobium-94	5.8×10^{-1}	1.9×10^{-2}
Technetium-99	7.1	2.5
Rhodium-102	1.2×10^{-3}	2.8×10^{-4}
Ruthenium-106	4.8×10^{-3}	6.7×10^{-4}
Palladium-107	6.3×10^{-2}	2.4×10^{-2}
Tin-126	4.4×10^{-1}	1.5×10^{-1}
Iodine-129	1.8×10^{-2}	6.3×10^{-3}
Cesium-134	1.6×10^1	3.4
Cesium-135	2.5×10^{-1}	1.0×10^{-1}
Cesium-137	3.1×10^4	1.1×10^4
Samarium-151	1.9×10^2	6.6×10^1
Lead-210	2.2×10^{-7}	9.4×10^{-8}
Radium-226	9.3×10^{-7}	3.7×10^{-7}
Radium-228	1.3×10^{-10}	4.7×10^{-11}
Actinium-227	7.8×10^{-6}	3.1×10^{-6}
Thorium-229	1.7×10^{-7}	6.1×10^{-8}
Thorium-230	1.5×10^{-4}	5.8×10^{-5}
Thorium-232	1.9×10^{-10}	6.9×10^{-11}
Protactinium-231	1.6×10^{-5}	6.0×10^{-6}
Uranium-232	1.9×10^{-2}	5.5×10^{-3}
Uranium-233	3.3×10^{-5}	1.1×10^{-5}
Uranium-234	6.6×10^{-1}	2.4×10^{-1}
Uranium-235	8.4×10^{-3}	3.0×10^{-3}
Uranium-236	1.4×10^{-1}	4.8×10^{-2}
Uranium-238	1.5×10^{-1}	6.2×10^{-2}
Neptunium-237	2.3×10^{-1}	7.3×10^{-2}
Plutonium-238	1.7×10^3	5.5×10^2
Plutonium-239	1.8×10^2	6.3×10^1
Plutonium-240	2.7×10^2	9.5×10^1
Plutonium-241	2.0×10^4	7.5×10^3
Plutonium-242	9.9×10^{-1}	4.0×10^{-1}
Americium-241	1.7×10^3	6.8×10^2
Americium-242/242m	1.1×10^1	4.6
Americium-243	1.3×10^1	4.9
Curium-242	8.7	3.8
Curium-243	8.3	3.1
Curium-244	7.0×10^2	2.5×10^2
Curium-245	1.8×10^{-1}	6.3×10^{-2}
Curium-246	3.8×10^{-2}	1.3×10^{-2}
Curium-247	1.3×10^{-7}	4.3×10^{-8}
Curium-248	3.9×10^{-7}	1.2×10^{-7}
Californium-252	3.1×10^{-8}	6.0×10^{-9}

a. Source: Appendix A, except cobalt-60.

b. Inventory numbers have been rounded to two significant figures.

c. Cobalt-60 inventory in crud, as calculated in this appendix.

reactors and 0.00025 for boiling-water reactors. Using the fuel rod dimensions and the number of rods per fuel assembly from Appendix A, these concentrations produce the following total inventory of cobalt-60 for a pressurized-water reactor fuel assembly at discharge:

$$\begin{aligned} \text{Cobalt-60 inventory} &= \text{fuel rod surface area per assembly} \times \text{cobalt-60 concentration} \\ \text{(per assembly)} &= \text{fuel rod diameter} \times \pi \\ &\quad \times \text{fuel rod length} \times \text{number of fuel rods per assembly} \\ &\quad \times \text{cobalt 60 concentration} \end{aligned}$$

For pressurized-water reactor assemblies, the corresponding values are (from Appendix A):

$$\begin{aligned} \text{Pressurized-water} &= 0.95 \text{ centimeters} \times 3.14 \\ \text{reactor cobalt-60} &\quad \times 366 \text{ centimeters} \times 264 \text{ rods} \\ \text{inventory} &\quad \times 0.00005 \text{ curie per square centimeter} \\ \text{(per assembly)} &\equiv 14 \text{ curies per pressurized-water reactor fuel assembly} \\ &\quad \text{(at reactor discharge)} \end{aligned}$$

For boiling-water reactor assemblies, the corresponding values are (from Appendix A):

$$\begin{aligned} \text{Boiling-water reactor} &= 1.25 \text{ centimeters} \times 3.14 \\ \text{cobalt-60 inventory} &\quad \times 366 \text{ centimeters} \times 55 \text{ rods} \\ \text{(per assembly)} &\quad \times 0.00025 \text{ curie per square centimeter} \\ &\equiv 20 \text{ curies per boiling-water reactor fuel assembly} \\ &\quad \text{(at reactor discharge)} \end{aligned}$$

The analysis used these concentrations, decayed to appropriate levels (25.9 years for pressurized-water reactor fuel and 27.2 years for boiling-water reactor fuel, from Table H-3), to obtain the final cobalt-60 inventory used in the source term determination.

The amount of crud that would be released from the surface of the fuel rod cladding is uncertain because there are very few data for the accident conditions of interest, and the physical condition of the crud can be highly variable (Sandoval et al. 1991, page 18). Two sources (NRC 1997, Table 7-1; NRC 1998, Table 4-1) recommend a release fraction of 1.0 (100 percent of the cobalt-60) for accident conditions; therefore, the EIS analysis assumed this value.

Following their release from the cladding, some crud particles would be retained by deposition on the surrounding surfaces (the fuel assembly cladding, spacer grids and structural hardware). The estimated fraction of released particles deposited on these surfaces would be 0.9 (SNL 1987, page 5-27), resulting in an escape fraction of 0.1. In accidents involving casks or canisters, additional surfaces represented by these components would offer surfaces for further plateout.

The inhalation radiation dose from cobalt-60 (or any radioactive particle) depends on the amount of particulate material inhaled into and remaining in the lungs (called the respirable fraction). The analysis assumed that the respirable fraction would be 0.05 (based on Wilmot 1981, page B-3). Therefore, the analysis assumed that the total cobalt-60 respirable airborne release fraction would be 0.005 (the escape fraction of 0.1 multiplied by the respirable fraction of 0.05) for accident scenarios involving commercial spent nuclear fuel assemblies.

H.2.1.4.1.2 Fuel Rod Gap. The space between the fuel rod cladding and the fuel pellets (called the *gap*) contains fission products released from the fuel pellets during reactor operation. The only

potentially important radionuclides in the gap are the gases tritium (hydrogen-3) and krypton-85, and the volatile radionuclides strontium-90, cesium-134, cesium-137, ruthenium-106, and iodine-129 (NRC 1997, page 7-6). The Nuclear Regulatory Commission recommends fuel rod release fractions (the fraction of the total fuel rod inventory) of 0.3 for tritium and krypton-85, 0.000023 for the strontium and cesium components, 0.000015 for ruthenium-106, and 0.1 for iodine under accident conditions that rupture the cladding (NRC 1997, page 7-6). The release fraction for the gases (tritium and krypton), as expected, would be rather high because most of the gas would be in the fuel rod gap and under pressure inside the fuel rod. The analysis also considered the fraction of the rods damaged in a given accident scenario. SNL (1987, page 6-19 *et seq.*) assumed that the fraction of damaged fuel pins in each assembly involved in a collision or drop accident scenario would be 20 percent. Another assessment (Kappes 1998, page 18) assumed that any drop of the fuel rods in a fuel assembly or basket of assemblies would result in failure of 10 percent of the fuel rods, regardless of the drop distance. Because neither value seems to have a strong basis, the EIS analysis assumed the more conservative 20-percent figure. For the particulate species released from the gap, the analysis applied a retention factor of 0.9 (escape factor of 0.1) to account for local deposition of the particles on the fuel assembly structures, consistent with SNL (1987, page 5-27). SNL (1987, page 5-28) also applies a similar factor to account for retention on the failed shipping cask structures for accident scenarios involving cask failure. However, the EIS analysis judged that this factor does not have a strong basis, especially because the actual mode of cask failure is unknown. For accident scenarios that could rupture the cask, surfaces on the cask structure might not be in the path of the released material and, therefore, would not be a potential deposition site. Furthermore, particulate material, which would escape local deposition on the fuel assembly surfaces, probably would be less susceptible to deposition on surfaces it encountered subsequently. Therefore, the analysis assumed no retention factor for cask structures. The final consideration is the fraction of remaining airborne particulates that would be respirable. No specific reference could be found to the volatile materials in the gap. The analysis conservatively assumed, therefore, that the respirable fraction would be 1.0.

H.2.1.4.1.3 Fuel Pellet. During reactor operation, the fuel pellets undergo cracking from thermal and mechanical stresses. This produces a small amount of pellet particulate material that contains radionuclides. The analysis assumed that the radionuclides are distributed evenly in the fuel pellets so that the fractional release of the pellet particulates is equivalent to the same fractional release of the total inventory of the appropriate radionuclides in the fuel. If the fuel cladding failed during an accident, a fraction of these particulates would be small enough (diameter less than 10 micrometers) for release to the atmosphere and would be respirable (small enough to remain in the lungs if inhaled). Sandia National Laboratories estimates this fraction to be 0.000001 (SNL 1987, page 5-26) based on experiments performed at Oak Ridge National Laboratory. The EIS used this value to develop source terms for the accident scenarios considered. Additional particulates could be produced by pulverization due to mechanical stresses imposed on the fuel pellets from the accident conditions. This pulverization factor has been evaluated in SNL (1987, page 5-17) and applied in Kappes (1998, page I-3). Based on experimental results involving bare fuel pellets, the analysis determined that the fraction likely to be pulverized into respirable particles would be proportional to the drop height (which is directly proportional to energy input) and would be:

$$2.0 \times 10^{-7} \times \text{energy partition factor} \times \text{unimpeded drop height (centimeters)} \quad (\text{Kappes 1998, page I-3}).$$

The energy partition factor is the fraction of the impact energy that is available for pellet pulverization. A large fraction of the impact energy is expended in deforming the fuel assembly structures and rupturing the fuel rod cladding. It has been estimated (SNL 1987, page 5-25) that the energy partition factor is 0.2.

As indicated above, some of the dispersible pellet particulates released in the accident could deposit on surfaces in the vicinity of the damaged fuel. Consistent with the particulate material considered above, the estimated fraction that would not deposit locally and would remain airborne would be 0.1 based on

SNL (1987, page 5-26). Based on these considerations, the respirable airborne release fraction produced from pulverization of the fuel pellets would be:

$$\begin{aligned}
 \text{Respirable airborne release fraction} &= 2 \times 10^{-7} \times \text{drop height (centimeters)} \\
 &\quad \times \text{energy partition factor} \times \text{fraction not deposited} \\
 &\quad \times \text{fuel rod damage fraction} \\
 &= 2 \times 10^{-7} \times \text{drop height} \\
 &\quad \times 0.2 \times 0.1 \\
 &\quad \times 0.2 \\
 &= 8 \times 10^{-10} \times \text{drop height}
 \end{aligned}$$

This result is reasonably consistent with the value of 8×10^{-7} from SAIC (1998, page 3-9), which is characterized as a bounding value for the respirable airborne release fraction for accident scenarios that would impose mechanical stress on fuel pellets for a range of energy densities (drop heights). This value would correspond to a drop from 1,000 centimeters (10 meters or 33 feet) based on the formulation above.

H.2.1.4.1.4 Conclusions. Table H-5 summarizes the source term parameters for commercial spent nuclear fuel drop accident scenarios, as discussed above.

Table H-5. Source term parameters for commercial spent nuclear fuel drop accident scenarios.

Radionuclide ^a	Location	Damage fraction	Release fraction	Fraction not deposited	Respirable fraction	Respirable airborne release fraction
Co-60	Clad surface	1.0	1.0	0.1	0.05	0.005
H-3, Kr-85, C-14	Gap	0.2	0.3	1.0	1.0	0.06
I-129	Gap	0.2	0.1	1.0	1.0	0.02
Cs-137, Sr-90	Gap	0.2	2.3×10^{-5}	0.1	1.0	4.6×10^{-7}
Ru-106	Gap	0.2	1.5×10^{-5}	0.1	1.0	3.0×10^{-7}
All solids	Gap (existing fuel fines)	0.2	1.0×10^{-6}	0.1	1.0	2.0×10^{-8}
All solids	Pellet-pulverization	0.2	$4.0 \times 10^{-8} \times h^b$	0.1	1.0	$8.0 \times 10^{-10} \times h^b$

a. Abbreviations: Co = cobalt; H = hydrogen (H-3 = tritium); Kr = krypton; C = carbon; I = iodine; Cs = cesium; Sr = strontium; Ru = ruthenium.

b. h = drop height in centimeters.

H.2.1.4.2 Transporter Runaway and Derailment Accident Source Term

This accident, as noted in Section H.2.1.3, would involve the runaway and derailment of the waste package transporter. It assumes the ejection of the waste package from the transporter during the event; the waste package would be split open by impact on the access tunnel wall. The calculated maximum impact speed would be 18 meters per second (38 miles per hour) (DOE 1997b, page 98). This analysis assumed that the source term from the damage to the 21 pressurized-water reactor fuel assemblies in the waste package is equivalent to a drop height that would produce the same impact velocity (equivalent to the same energy input). The equivalent drop height was computed from basic equations for the motion of a body falling under the influence of gravity:

$$\begin{aligned}
 \text{velocity} &= \text{acceleration} \times \text{time} \\
 \text{and,} \\
 \text{distance} &= \frac{1}{2} \times \text{acceleration} \times \text{time squared}
 \end{aligned}$$

where: velocity = velocity of the impact (18 meters per second)
time = time required for the fall
acceleration = acceleration due to gravity (9.8 meters per second squared)

By substitution,

$$\begin{aligned} \text{distance} &= \frac{1}{2} \times \text{acceleration} \times (\text{velocity} \div \text{acceleration})^2 \\ &= (\text{velocity})^2 \div (\text{acceleration} \times 2) \\ &= (18)^2 \div (9.8 \times 2) \\ &= 16 \text{ meters} \end{aligned}$$

Thus, the calculation of the source term for this accident scenario assumed a drop height of 16 meters and used the parameters in Table H-5 for the various nuclide groups.

H.2.1.4.3 DOE Spent Nuclear Fuel Drop Accident Source Term

Appendix A lists the various types of DOE spent nuclear fuel and high-level radioactive waste that the Department would place in the proposed repository. A review of the inventory indicates that the spent nuclear fuel from the Hanford Site (N-Reactor fuel) represents a large percentage of DOE spent nuclear fuel. The N-Reactor fuel also has one of the highest radionuclide inventories of any of the DOE spent fuels. Although a canister of naval spent nuclear fuel would have a higher radionuclide inventory than a canister of N-Reactor fuel (Appendix A, Table A-18), the amount of radioactive material that would be released from a naval canister during this hypothetical accident scenario would be less than the amount released from an N-Reactor fuel canister due to the highly robust design of naval fuel (Appendix A, Section A.2.2.5.3) (USN 1996, all). Therefore, DOE selected N-Reactor spent nuclear fuel material as the bounding form to represent the source term for accidents that would involve DOE material. The analysis derived the source term for accidents involving a drop of N-Reactor fuel from DOE (1995, page 5-88), which lists the estimated source term for a drop of a cask containing 1,000 kilograms (2,200 pounds) of N-Reactor fuel from a height of 4.6 meters (15 feet). For the repository accident scenario involving N-Reactor fuel, a total of 4,800 kilograms (10,600 pounds) of fuel would be involved in a multi-canister overpack drop (Appendix A) from a height of 6.3 meters (21 feet), as noted above. The analysis adjusted the DOE (1995, page 5-88) source term upward by a factor of 4.8 to account for the increased amount of material involved (4,800 kilograms as opposed to 1,000 kilograms), and by a factor of 1.37 to account for the increased drop height (6.3/4.6) because the analysis assumed the source term would be proportional to the energy input, which is proportional to the drop height. These two factors were applied to the DOE (1995, page 5-88) source term and the result is listed in Table H-6. The behavior of N-Reactor fuel during an accident is uncertain (Kappes 1998, page 15) and the Final EIS analysis might utilize a revised source term estimate based on the results of further studies of this fuel. Furthermore, DOE has not developed the requirements for receipt of the fuel at the repository. These requirements could influence the source term, as could the corresponding requirements for processing the fuel prior to shipment.

H.2.1.4.4 Seismic Accident Scenario Source Term

Waste Handling Building. In this event, as noted in Section H.2.1.3, the Waste Handling Building could collapse from a beyond-design-basis earthquake. Bare fuel assemblies being transferred during the event would be likely to drop to the floor and concrete from the ceiling could fall on the fuel assemblies, causing damage that could result in radioactive release, which would discharge to the atmosphere through the damaged roof. In addition, other radioactive material stored or being handled in the Waste Handling Building could be vulnerable to damage. To estimate the source term, the analysis evaluated the extent of damage to the fuel rods and pellets for the assemblies being transferred and then examined the other material that could be vulnerable.

Table H-6. Source term used for N-Reactor Mark IV fuel drop accident scenario analysis (curies).^a

Radionuclide	Total release	Radionuclide	Total release	Radionuclide	Total release
Tritium (H ₃)	1.7×10 ⁻²	Tin-119m	1.7×10 ⁻⁸	Europium-154	8.3×10 ⁻²
Carbon-14	2.6×10 ⁻⁴	Tin-121m	3.0×10 ⁻⁵	Uranium-234	1.7×10 ⁻⁴
Iron-55	1.3×10 ⁻³	Tin-126	5.6×10 ⁻⁵	Uranium-235	5.7×10 ⁻⁶
Nickel-59	1.4×10 ⁻⁵	Stibium-125 (antimony)	2.4×10 ⁻²	Uranium-236	3.3×10 ⁻⁵
Nickel-63	1.7×10 ⁻³	Stibium-126	7.9×10 ⁻⁶	Uranium-238	1.4×10 ⁻⁴
Cobalt-60	5.4×10 ⁻²	Stibium-126m	5.6×10 ⁻⁵	Neptunium-237	2.6×10 ⁻⁵
Selenium-79	2.9×10 ⁻⁵	Tellurium-125m	6.7×10 ⁻³	Plutonium-238	7.9×10 ⁻²
Krypton-85	2.4×10 ⁻²	Iodine-129	2.3×10 ⁻⁶	Plutonium-239	7.3×10 ⁻²
Strontium-90	3.6	Cesium-134	2.3×10 ⁻²	Plutonium-240	5.9×10 ⁻²
Yttrium-90	3.6	Cesium-135	2.6×10 ⁻⁵	Plutonium-241	4.3
Niobium-93m	7.2×10 ⁻⁵	Cesium-137	4.9	Plutonium-242	4.9×10 ⁻⁵
Zirconium-93	1.3×10 ⁻⁴	Cerium-144	8.9×10 ⁻⁵	Americium-241	1.7×10 ⁻¹
Technetium-99	9.7×10 ⁻⁴	Praseodymium-144	8.9×10 ⁻⁵	Americium-242	3.9×10 ⁻⁴
Ruthenium-106	8.0×10 ⁻⁴	Praseodymium-144m	1.1×10 ⁻⁶	Americium-242m	3.9×10 ⁻⁴
Palladium-107	6.7×10 ⁻⁶	Promethium-147	2.4×10 ⁻¹	Americium-243	5.4×10 ⁻⁵
Silver-110m	1.3×10 ⁻⁸	Samarium-151	4.6×10 ⁻²	Curium-242	3.2×10 ⁻⁴
Cadmium-113m	1.6×10 ⁻³	Europium-152	4.9×10 ⁻⁴	Curium-244	2.4×10 ⁻²

a. Source: DOE (1995, page 5-88), with adjustments as noted above.

The ceiling of the transfer cell, which would consist of concrete 20 to 25 centimeters (8 to 10 inches) thick, would be about 15 meters (50 feet) high (TRW 1999b, Attachment IV, Figure 13). Typical pressurized-water reactor fuel assemblies weigh 660 kilograms (1,500 pounds) each (see Appendix A). The assemblies are about 21 centimeters (8.3 inches) wide by about 410 centimeters (160 inches) long, for an effective cross-sectional area (horizontal) of 1 square meter (11 square feet) (SNL 1987, page 5-2). The weight of a single fuel assembly is roughly equivalent to a 25-centimeter-thick concrete block with a 1-square-meter cross-section [about 750 kilograms (1,700 pounds) based on a density of 2.85 grams per cubic centimeter (180 pounds per cubic foot) (CRC 1997, page 15-28)]. Thus, as a first approximation, the analysis assumed that the concrete blocks falling from the ceiling onto the fuel assemblies would produce about the same energy as the fuel assemblies falling from the same height.

Some of the energy imparted to the fuel assemblies from the falling debris would be absorbed in deforming the fuel assembly structures and, thus, would not be available to pulverize the fuel pellets. As evaluated above for falling fuel assemblies, this energy absorption factor would result in an estimated 20 percent of the energy being imparted to the pellets and the rest absorbed by the structure (SNL 1987, page 5-25). Finally, as noted above, the analysis used a 0.1 release factor (0.9 retention) to represent the retention of the released fuel particles by deposition on the cladding and other fuel assembly structures (SNL 1987, page 5-27). In addition, it assumed that additional retention would be associated with the concrete and other rubble that would be on top, or in the vicinity, of the fuel assemblies. It assumed this release factor would be 0.1 (0.9 retention) consistent with that used by SNL (1987, page 5-28) for retention by deposition on the cask and canister materials that surround the fuel assemblies during accident scenarios. It also assumed a fuel pellet pulverization factor of $8 \times 10^{-10} \times h$, the same as that used for fuel assembly drop accident scenarios. Thus, the overall pellet respirable airborne release fraction for the fuel pellet particulates is:

$$\begin{aligned}
 \text{Respirable airborne release fraction} &= 8 \times 10^{-10} \times \text{drop height (centimeters)} \times \text{rubble retention} \\
 &= 8 \times 10^{-10} \times 1,500 \times 0.1 \\
 &= 1.2 \times 10^{-6}
 \end{aligned}$$

Other radioactive materials either stored or being handled in the Waste Handling Building could also be at risk. For material in casks and canisters and waste packages, the analysis assumed that the damage

potential from falling debris would not be great enough to cause a large radionuclide release. This is based on the fact that canisters and casks are quite robust and that, even if the containers were breached by the energy of the impact, there would be very little energy remaining to cause fuel pellet pulverization. There could be, however, bare fuel assemblies exposed in the dryers and in disposal containers awaiting lid attachment. An estimated 375 bare pressurized-water reactor fuel assemblies could be exposed to falling debris (Montague 1999, page 1). The location of this material would be as follows:

- Assembly transfer system dryers: 25 pressurized-water reactor assemblies
- Disposal canister handling system welding stations: 346 pressurized-water reactor assemblies
- Transfer operations: four pressurized-water reactor assemblies

Because the concrete roof heights over these areas would be roughly the same as the assembly transfer system area in the Waste Handling Building [15 meters (50 feet)] where the analysis assumed the four bare pressurized-water reactor assemblies would be involved, the analysis assumed the pellet pulverization contribution to the source term to be equivalent to that for the fuel assemblies being transferred. The overall source term, then, was determined by assuming 375 typical pressurized-water reactor assemblies with the release fractions listed in Table H-5.

Boiling-water reactor fuel assemblies could be exposed at these areas, but the analysis evaluated only pressurized-water reactor fuel assemblies because they would result in a slightly higher source term under equivalent accident conditions and would be more likely to be involved because they would comprise a larger amount of material (see Appendix A) to be received at the repository. Thus, the source term for the seismic event would be 375 typical pressurized-water reactor fuel assemblies (Table H-4) with release fractions based on Table H-5.

Waste Treatment Building. It is assumed that the radionuclide concentration for the dry compactable waste in the Waste Treatment Building would be similar to that for power reactors (McFeely 1998, page 2). This material would consist of paper, plastic, and cloth with a specific activity of 0.025 curie per cubic meter (0.7 millicurie per cubic foot) (McFeely 1998, page 2). This activity would consist primarily of cobalt isotopes (primarily cobalt-60) representing 67 percent of the total activity, and cesium, which would contribute 28 percent of the total (McFeely 1999, all).

The Waste Treatment Building would operate a single shift per day, and would continuously process waste such that no large accumulation would occur. Because Waste Handling Building operations would be likely to involve three shifts per day (TRW 1999b, Section 6.2), the analysis assumed that three shifts of solid waste would accumulate before the Waste Treatment Building began its single-shift operation. The generation rate of solid compactible waste would be about 1,500 cubic meters (53,000 cubic feet) per year (DOE 1997a, page 32) or about 0.17 cubic meter (5.8 cubic feet) per hour. Thus, three shifts (24 hours) of Waste Handling Building operation would produce about 4.0 cubic meters (140 cubic feet) of solid compactible waste. The total radionuclide inventory in this waste would be:

$$\begin{aligned} \text{Cobalt-60} &= 4.0 \text{ cubic meters} \times 0.025 \text{ curie per cubic meters} \times 0.67 \text{ (cobalt-60 fraction)} \\ &\cong 0.07 \text{ curie} \end{aligned}$$

$$\begin{aligned} \text{Cesium-137} &= 4.0 \text{ cubic meters} \times 0.025 \text{ curie per cubic meters} \times 0.28 \text{ (cesium-137 fractions)} \\ &\cong 0.03 \text{ curie} \end{aligned}$$

The respirable airborne release fraction for a fire involving combustible low-level waste has been conservatively estimated at 0.4 (Mueller et al. 1996, page D-21). Thus, the respirable airborne release source term for the fire accident scenario would be:

$$\begin{aligned}\text{Cobalt-60} &= 0.07 \text{ curie} \times 0.4 = 0.028 \text{ curie} \\ \text{Cesium-137} &= 0.03 \text{ curie} \times 0.4 = 0.012 \text{ curie}\end{aligned}$$

The assumed release height for the accident scenario is 2 meters (6.6 feet). This is the minimum release height for the consequences analysis and represents a ground-level release.

H.2.1.4.5 Low-Level Waste Drum Failure Source Term

As indicated in Section H.2.1.2, the most meaningful accident scenarios involving exposure to workers would be those related to puncture or rupture of waste drums that contained low-level waste. Such events could occur during handling operations and probably would involve the puncture of a drum by a forklift, or the drop of the drum during stacking and loading operations.

Two types of waste drums would contain the processed waste. Concentrated liquid waste would be mixed with cement and poured into 0.21-cubic-meter (55-gallon) drums. Compacted and noncompacted solid waste would also be placed in the same drums, which would, in turn, be placed in 0.32-cubic-meter (85-gallon) drums with the space between the two drums grouted. The probability of a drum failure was analyzed for these two drum types.

Following a drum failure, some fraction of the radionuclides in the waste would be released and workers in the immediate vicinity could be exposed to the material. The amount released would depend on the radionuclide concentration in the low-level waste material, the fraction of low-level waste released from the drum on its failure, and the respirable airborne release fraction from the released waste.

For liquid waste, the concentration of radionuclides is expected to be (McFeely 1998, page 3):

$$\begin{aligned}\text{Cobalt-60} &= 0.001 \text{ curie per cubic meter} \\ \text{Cesium-137} &= 0.0015 \text{ curie per cubic meter}\end{aligned}$$

As noted in Section H.2.1.2, the evaporator would concentrate the liquid waste down to 10 percent of the original generated so the concentration of radionuclides in the waste would be increased to:

$$\begin{aligned}\text{Cobalt 60} &= 0.01 \text{ curie per cubic meter} \\ \text{Cesium-137} &= 0.015 \text{ curie per cubic meter}\end{aligned}$$

The grouting operation would dilute this concentration somewhat by adding cement, but this dilution has been ignored for conservatism.

The total activity in a 0.21-cubic meter (55-gallon) drum would become:

$$\begin{aligned}\text{Cobalt-60} &= 0.01 \text{ curie per cubic meter} \times 0.21 \text{ cubic meter} \\ &\cong 0.0021 \text{ curie per drum} \\ \text{Cesium-137} &= 0.015 \text{ curie per cubic meter} \times 0.21 \text{ cubic meter} \\ &\cong 0.0032 \text{ curie per drum}\end{aligned}$$

For dry compacted waste, the total inventory in a 0.21-cubic-meter (55-gallon) drum would be

$$\begin{aligned} \text{Cobalt-60} &= 0.21 \text{ cubic meter} \times 0.025 \text{ curie per cubic meter} \times 0.67 \text{ (cobalt-60 fraction)} \\ &\cong 0.0035 \text{ curie} \end{aligned}$$

$$\begin{aligned} \text{Cesium-137} &= 0.21 \text{ cubic meter} \times 0.025 \text{ curie per cubic meter} \times 0.28 \text{ (cesium-137 fraction)} \\ &\cong 0.0015 \text{ curie} \end{aligned}$$

The estimated amount of material released from drums containing solid waste is 25 percent of the contents based on Mueller et al. (1996, page 94). Values from Mueller et al. (1996, all) were used for the respirable airborne release fraction. For dry waste, the recommended respirable airborne release fraction is 0.001. For grouted liquid waste, this fraction is determined by the following equation:

$$\text{Respirable airborne release fraction} = A \times D \times G \times H$$

where:

$$\begin{aligned} A &= \text{constant } (2.0 \times 10^{-11}) \text{ (Mueller et al. 1996, page D-25)} \\ D &= \text{material density [3.14 grams per cubic centimeter (196 pounds per cubic foot)} \\ &\quad \text{(McFeely 1998, all)} \\ G &= \text{gravitational acceleration [980 centimeters (32.2 feet) per second squared]} \\ H &= \text{height of fall of the drum in the accident scenario} \end{aligned}$$

The assumed height of the fall is 2 meters (6.6 feet), which would be the approximate maximum lift height when the drum was stacked on another drum or placed on a carrier for offsite transportation. This same formula applies to drum puncture accident scenarios (Mueller et al. 1996, page D-30), and the 2-meter drop event would be equivalent in damage potential to a forklift impact at about 4.5 meters per second (10 miles per hour). The respirable airborne release fraction for this case then becomes:

$$\begin{aligned} \text{Respirable airborne release fraction} &= 2.0 \times 10^{-11} \times 3.14 \times 980 \times 200 \\ &\cong 1.23 \times 10^{-5} \end{aligned}$$

Based on these results, the worker risk would be dominated by accidents involving drums that contained dry waste because both the frequency of the event [0.59 versus 0.46 (Section H.2.1.2)] and the release fraction [1×10^{-3} versus 1.23×10^{-5} (derived above)] would be greater. The total amount of airborne respirable material release (source term) for the risk-dominant dry waste accident scenario would be:

$$\begin{aligned} \text{Cobalt-60} &= 0.0035 \text{ curie (total drum inventory)} \times 0.25 \text{ (fraction released)} \\ &\quad \times 0.001 \text{ (respirable airborne release fraction)} \\ &\cong 8.5 \times 10^{-7} \text{ curies} \end{aligned}$$

$$\begin{aligned} \text{Cesium-137} &= 0.0015 \text{ curie (total drum inventory)} \times 0.25 \text{ (fraction released)} \\ &\quad \times 0.001 \text{ (respirable airborne release fraction)} \\ &\cong 3.8 \times 10^{-7} \text{ curies} \end{aligned}$$

The analysis assumed that, following normal industrial practice, workers would not be in the area beneath suspended objects. Accordingly, the nearest worker was assumed to be 5 meters (16 feet) from the impact area. Therefore, the volume assumed for dispersion of the material prior to reaching the worker would be 125 cubic meters (4,400 cubic feet), which represents the immediate vicinity of the accident

location [a volume approximately 5 meters (16 feet) by 5 meters by 5 meters]. The breathing rate of the worker would be 0.00035 cubic meter (about 0.012 cubic foot) per second (ICRP 1975, page 346).

H.2.1.5 Assessment of Accident Scenario Consequences

Accident scenario consequences were calculated as individual doses (rem), collective doses (person-rem), and latent cancer fatalities. The receptors considered were (1) the maximally exposed offsite individual, defined as a hypothetical member of the public at the point on the proposed repository land withdrawal boundary who would receive the largest dose from the assumed accident scenario (a minimum distance of 11 kilometers (7 miles), (2) the maximally exposed involved worker, the hypothetical worker who would be nearest the spent nuclear fuel or high-level radioactive waste when the accident occurred, (3) the noninvolved worker, the hypothetical worker near the accident but not involved in handling the material, assumed to be 100 meters (about 330 feet) from the accident, and (4) the members of the public who reside within about 80 kilometers (50 miles) of the proposed repository.

For radiation doses below about 20 rem and low dose rates (below 10 rem per hour), potential health effects would be those associated with a chronic exposure or an increase in the risk of fatal cancer (ICRP 1991, Chapter 3) (see the discussion in Appendix F, Section F.1). The International Committee on Radiation Protection has recommended the use of a conversion factor of 0.0005 fatal cancer per person-rem for the general population for low doses, and a value of 0.0004 fatal cancer per person-rem for workers for chronic exposures. The higher value for the general population accounts in part for the fact that the general population contains young people, who are more susceptible to the effects of radiation. These conversion factors were used in the EIS consequence analysis. The latent cancer fatality caused by radiation exposure could occur at any time during the remaining lifetime of the exposed individual. As dose increases above about 15 rem over a short period (acute exposures), observable physical effects can occur, including temporary male sterility (ICRP 1991, page 15). At even higher acute doses (above about 500 rem), death within a few weeks is probable (ICRP 1991, page 16).

DOE used the MACCS2 computer program (Rollstin, Chanin, and Jow 1990, all; Chanin and Young 1998, all) and the radionuclide source terms for the identified accident scenarios in Section H.2.1.4 to calculate consequences to receptors. This program, developed by the U.S. Nuclear Regulatory Commission and DOE, has been widely used to compute radiological impacts from accident scenarios involving releases of radionuclides from nuclear fuel and radioactive waste. DOE used this program for offsite members of the public, the maximally exposed offsite individual, and the noninvolved worker. The MACCS2 program calculates radiological doses based on a sampling of the distribution of weather conditions for a year of site-specific weather data. Meteorological data were compiled at the proposed repository site from 1993 through 1997. This analysis used the weather conditions for 1993. The selection of 1993 was based on a sensitivity analysis that showed that, on the average, the weather conditions for 1993 produced somewhat higher consequences than those for the other years for most receptors, although the variation from year to year was small.

For exposure to inhaled radioactive material, it was assumed (in accordance with U.S. Environmental Protection Agency guidance) that doses would accumulate in the body for a total of 50 years after the accident (Eckerman, Wollbarst, and Richardson 1988, page 7). For external exposure (from ground contamination and contaminated food consumption), the dose was assumed to accumulate for 70 years (DOE 1993, page 21).

The MACCS2 program provides doses to selected receptors for a contiguous spectrum of site-specific weather conditions. Two weather cases were selected for the EIS: (1) a median weather case (designated at 50 percent) that represents the weather conditions that would produce median consequences to the

receptors, and (2) a 95 percent weather case that provides higher consequences that would only be exceeded 5 percent of the time.

The MACCS2 program is not suitable for calculating doses to receptors near the release point of radioactive particles [within about 100 meters (330 feet)]. For such cases, the analysis calculated involved worker dose estimates using a breathing rate of 0.00035 cubic meter (0.012 cubic foot) per second (ICRP 1975, page 346). For involved worker dose calculations from accident scenarios in the cask transfer and handling area, the analysis assumed that the worker would be a minimum of 4.6 meters (15 feet) from the location of the cask impact with the floor during the accident (normal industrial practice would preclude workers from being in the immediate vicinity of areas where heavy objects could strike the floor during lifting operations). Because of the perceived hazard following a breached cask, the analysis assumed that the worker would immediately vacate the area after observing that the cask had ruptured. Accordingly, the analysis assumed that the worker would breathe air containing airborne radioactive material from the ruptured cask for 10 seconds.

For involved worker doses from the drum handling accident scenario, the analysis assumed that the worker (a forklift operator) would be 3 meters (10 feet) from the drum rupture location, and would breathe air containing radioactive material from the ruptured drum for 30 seconds.

The involved worker dose estimates used the same dose conversion factors as those used by the MACCS2 program for inhalation exposure.

The analysis assumed that the population around the repository would be that projected for the year 2000 (see Appendix G, Table G-44). The exposed population would consist of individuals living within about 80 kilometers (50 miles) of the repository, including pockets of people who would reside just beyond the 80-kilometer distance. The dose calculations included impacts from the consumption of food contaminated by the radionuclide releases. The contaminated food consumption analysis used site-specific data on food production and consumption for the region around the proposed site (TRW 1997b, all). For conservatism, the analysis assumed no mitigation measures, such as post-accident evacuation or interdiction of contaminated foodstuffs. However, DOE would take appropriate mitigation actions in the event of an actual release.

The results of the consequence analysis are listed in Tables H-7 (for 50-percent weather) and H-8 (for 95-percent weather). These tables list doses in rem for individual receptors and in person-rem (collective dose to all exposed persons) for the 80-kilometer (50-mile) population around the site. For selected receptors, as noted, the tables list estimated latent cancer fatalities predicted to occur over the lifetime of the exposed receptors as a result of the calculated doses using the conversion factors described in this section. These estimates do not consider the accident frequency. For comparison, in 1995 the lifetime incidence of fatal cancer from all causes for Nevada residents was 0.24 (CDC 1998, page 215). Thus, the estimated latent cancer fatalities for the individual receptors from accidents would be very small in comparison to the cancer incidence from other causes. For the 28,000 persons living within 80 kilometers of the site (see Appendix G), 6,720 ($28,000 \times 0.24$) would be likely to die eventually of cancer. The accident of most concern for the 95-percent weather conditions (earthquake, Table H-8, number 14) would result only in an estimated 0.0072 latent cancer fatality for this same population.

H.2.2 NONRADIOLOGICAL ACCIDENT SCENARIOS

A potential release of hazardous or toxic materials during postulated operational accident scenarios at the repository would be very unlikely. Because of the large quantities of radioactive material, radiological considerations would outweigh nonradiological concerns. The repository would not accept hazardous waste as defined by the Resource Conservation and Recovery Act (40 CFR Parts 260 to 299). Some

Table H-7. Radiological consequences of repository operations accidents for median (50th-percentile) meteorological conditions.

Accident scenario ^{a,b,c}	Maximally exposed offsite individual					Noninvolved worker		Involved worker	
	Frequency (per year) ^a	Dose (rem)	LCFi ^d	Dose (person-rem)	LCFp ^d	Dose (rem)	LCFi	Dose (rem)	LCFi
1. 6.9-meter drop of shipping cask in CTHA-61 BWR assemblies-no filtration	4.5×10 ⁻⁴	1.9×10 ⁻³	1.0×10 ⁻⁶	5.5×10 ⁻²	2.7×10 ⁻⁵	9.4×10 ⁻¹	3.8×10 ⁻⁴	76	3.0×10 ⁻²
2. 7.1-meter drop of shipping cask in CTHA-26 PWR assemblies-no filtration	6.1×10 ⁻⁴	2.3×10 ⁻³	1.2×10 ⁻⁶	6.6×10 ⁻²	3.3×10 ⁻⁵	1.1	4.4×10 ⁻⁴	90	3.6×10 ⁻²
3. 4.1-meter drop of shipping cask in CTHA-61 BWR assemblies- no filtration	1.4×10 ⁻³	1.3×10 ⁻³	6.5×10 ⁻⁷	3.9×10 ⁻²	2.0×10 ⁻⁵	5.7×10 ⁻¹	2.3×10 ⁻⁴	46	1.8×10 ⁻²
4. 4.1-meter drop of shipping cask in CTHA-26 PWR assemblies-no filtration	1.9×10 ⁻³	1.4×10 ⁻³	7.0×10 ⁻⁷	4.6×10 ⁻²	2.3×10 ⁻⁵	6.6×10 ⁻¹	2.6×10 ⁻⁴	53	2.1×10 ⁻²
5. 6.3-meter drop of MCO in CTS-10 N-Reactor fuel canisters-filtration	4.5×10 ⁻⁴	3.7×10 ⁻⁷	1.9×10 ⁻¹⁰	1.1×10 ⁻⁵	5.3×10 ⁻⁹	1.1×10 ⁻⁴	4.4×10 ⁻⁸	(e)	(e)
6. 6.3-meter drop of MCO in CTS-10 N-reactor fuel canisters-no filtration	2.2×10 ⁻⁷	1.2×10 ⁻³	6.0×10 ⁻⁷	3.4×10 ⁻²	1.7×10 ⁻⁵	3.6×10 ⁻¹	1.4×10 ⁻⁴	(e)	(e)
7. 5-meter drop of transfer basket in ATS-8 PWR assemblies-filtration	1.1×10 ⁻²	6.6×10 ⁻⁷	3.3×10 ⁻¹⁰	4.0×10 ⁻⁴	2.0×10 ⁻⁷	1.7×10 ⁻⁴	6.8×10 ⁻⁸	(e)	(e)
8. 5-meter drop of transfer basket in ATS-8 PWR assemblies-no filtration	2.8×10 ⁻⁷	5.6×10 ⁻⁴	2.8×10 ⁻⁷	1.7×10 ⁻²	8.6×10 ⁻⁶	1.6×10 ⁻¹	6.4×10 ⁻⁵	(e)	(e)
9. 7.6-meter drop of transfer basket in ATS-16 BWR assemblies-filtration	7.4×10 ⁻³	5.1×10 ⁻⁷	2.6×10 ⁻¹⁰	2.9×10 ⁻⁴	1.5×10 ⁻⁷	1.3×10 ⁻⁴	5.2×10 ⁻⁸	(e)	(e)
10. 7.6-meter drop of transfer basket in ATS-16 BWR fuel assemblies-no filtration	1.9×10 ⁻⁷	6.1×10 ⁻⁴	3.1×10 ⁻⁷	1.6×10 ⁻²	8.2×10 ⁻⁶	1.8×10 ⁻¹	7.2×10 ⁻⁵	(e)	(e)
11. 6-meter drop of disposal container in DCHS-21 PWR assemblies-filtration	1.8×10 ⁻³	1.8×10 ⁻⁶	9.0×10 ⁻¹⁰	1.0×10 ⁻³	5.2×10 ⁻⁷	5.0×10 ⁻⁴	2.0×10 ⁻⁷	(e)	(e)
12. 6-meter drop of disposal container in DCHS-21 PWR fuel assemblies-no filtration	8.6×10 ⁻⁷	1.7×10 ⁻³	8.5×10 ⁻⁷	5.1×10 ⁻²	2.5×10 ⁻⁵	5.1×10 ⁻¹	2.0×10 ⁻⁴	(e)	(e)
13. Transporter runaway and derailment in access tunnel-21 PWR assemblies-filtration-16-meter drop height equivalent	1.2×10 ⁻⁷	4.3×10 ⁻³	2.2×10 ⁻⁶	1.1×10 ⁻¹	5.4×10 ⁻⁵	1.5	6.0×10 ⁻⁴	(f)	(f)
14. Earthquake - 375 PWR assemblies	2.0×10 ⁻⁵	9.1×10 ⁻³	4.6×10 ⁻⁶	3.6×10 ⁻¹	1.8×10 ⁻⁴	8.3	3.3×10 ⁻³	(f)	(f)
15. Earthquake w/fire in WTB	2.0×10 ⁻⁵	1.8×10 ⁻⁵	9.0×10 ⁻⁹	6.3×10 ⁻⁴	3.2×10 ⁻⁷	5.2×10 ⁻³	2.1×10 ⁻⁶	(f)	(f)
16. LLW drum rupture in WTB	0.59	6.1×10 ⁻¹⁰	3.1×10 ⁻¹³	2.1×10 ⁻⁸	1.1×10 ⁻¹¹	1.4×10 ⁻⁷	5.6×10 ⁻¹¹	7.0×10 ⁻⁵	2.8×10 ⁻⁸

- a. Source: Kappes (1998, all). These frequency estimates are highly uncertain due to the preliminary nature of the repository design and are provided only to show potential accident sequence credibility. They represent conservative estimates based on the approach taken in Kappes (1998, all).
- b. CTHA = Cask Transfer/Handling Area, CTS = Canister Transfer System, ATS = Assembly Transfer System, DCHS = Disposal Container Handling System, WTB = Waste Treatment Building.
- c. To convert meters to feet, multiply by 3.2808.
- d. LCFi is the likelihood of a latent cancer fatality for an individual who receives the calculated dose. LCFp is the number of cancers probable in the exposed population from the collective population dose (person-rem). These values were computed based on a conversion of dose in rem to latent cancers as recommended by the International Council on Radiation Protection as discussed in this section.
- e. For these cases, the involved workers are not expected to be vulnerable to exposure during an accident because operations are done remotely. Thus, involved worker impacts were not evaluated.
- f. For these events, involved workers would likely be severely injured or killed by the event; thus, no radiological impacts were evaluated. For the seismic event, as many as 39 people could be injured or killed in the Waste Handling Building, and as many as 36 in the Waste Treatment Building based on current staffing projections (TRW 1998c, pages 17 and 18).

Table H-8. Radiological consequences of repository operations accidents for unfavorable (95th-percentile) meteorological conditions.

Accident scenario ^{a,b,c}	Frequency (per year) ^a	Maximally exposed offsite individual		Population		Noninvolved worker		Involved worker	
		Dose (rem)	LCFi ^d	Dose (person-rem)	LCFp ^d	Dose (rem)	LCFi	Dose (rem)	LCFi
1. 6.9-meter drop of shipping cask in CTHA-61 BWR assemblies-no filtration	4.5×10 ⁻⁴	7.2×10 ⁻³	3.5×10 ⁻⁶	1.7	8.6×10 ⁻⁴	5.1	2.0×10 ⁻³	76	3.0×10 ⁻²
2. 7.1-meter drop of shipping cask in CTHA-26 PWR assemblies-no filtration	6.1×10 ⁻⁴	8.0×10 ⁻³	4.0×10 ⁻⁶	2.1	1.1×10 ⁻³	5.9	2.4×10 ⁻³	90	3.6×10 ⁻²
3. 4.1-meter drop of shipping cask in CTHA-61 BWR assemblies-no filtration	1.4×10 ⁻³	4.3×10 ⁻³	2.2×10 ⁻⁶	1.3	6.5×10 ⁻⁴	3.1	1.2×10 ⁻³	46	1.8×10 ⁻²
4. 4.1-meter drop of shipping cask in CTHA-26 PWR assemblies-no filtration	1.9×10 ⁻³	5.2×10 ⁻³	2.6×10 ⁻⁶	1.5	7.8×10 ⁻⁴	3.5	1.4×10 ⁻³	53	2.1×10 ⁻²
5. 6.3-meter drop of MCO in CTS-10 N-Reactor fuel canisters-filtration	4.5×10 ⁻⁴	1.2×10 ⁻⁶	6.0×10 ⁻¹⁰	2.6×10 ⁻⁴	1.3×10 ⁻⁷	3.3×10 ⁻⁷	1.3×10 ⁻⁷	(e)	(e)
6. 6.3-meter drop of MCO in CTS-10 N-reactor fuel canisters-no filtration	2.2×10 ⁻⁷	4.3×10 ⁻³	2.2×10 ⁻⁶	8.6×10 ⁻¹	4.3×10 ⁻⁴	1.1	4.4×10 ⁻⁴	(e)	(e)
7. 5-meter drop of transfer basket in ATS-8 PWR assemblies- filtration	1.1×10 ⁻²	2.5×10 ⁻⁶	1.3×10 ⁻⁹	3.3×10 ⁻²	1.6×10 ⁻⁵	4.6×10 ⁻⁷	1.8×10 ⁻⁷	(e)	(e)
8. 5-meter drop of transfer basket in ATS-8 PWR assemblies-no filtration	2.8×10 ⁻⁷	2.1×10 ⁻³	1.1×10 ⁻⁶	5.6×10 ⁻¹	2.8×10 ⁻⁴	4.6×10 ⁻⁷	1.8×10 ⁻⁴	(e)	(e)
9. 7.6-meter drop of transfer basket in ATS-16 BWR assemblies-filtration	7.4×10 ⁻³	2.1×10 ⁻⁶	1.1×10 ⁻⁹	2.4×10 ⁻²	1.2×10 ⁻⁵	3.8×10 ⁻⁷	1.5×10 ⁻⁷	(e)	(e)
10. 7.6-meter drop of transfer basket in ATS-16 BWR fuel assemblies-no filtration	1.9×10 ⁻⁷	2.2×10 ⁻³	1.1×10 ⁻⁶	5.1×10 ⁻¹	2.6×10 ⁻⁴	5.1×10 ⁻⁷	2.0×10 ⁻⁴	(e)	(e)
11. 6-meter drop of disposal container in DCHS-21 PWR assemblies-filtration	1.8×10 ⁻³	7.3×10 ⁻⁶	3.7×10 ⁻⁹	8.6×10 ⁻²	4.3×10 ⁻⁵	1.3×10 ⁻⁷	5.2×10 ⁻⁷	(e)	(e)
12. 6-meter drop of disposal container in DCHS-21 PWR fuel assemblies-no filtration	8.6×10 ⁻⁷	6.1×10 ⁻³	3.1×10 ⁻⁶	1.6	8.0×10 ⁻⁴	1.3	5.2×10 ⁻⁴	(e)	(e)
13. Transporter runaway and derailment in access tunnel-21 PWR assemblies-filtration-16-meter drop height equivalent	1.2×10 ⁻⁷	1.3×10 ⁻²	6.5×10 ⁻⁶	3.2	1.6×10 ⁻³	3.9	1.6×10 ⁻³	(f)	(f)
14. Earthquake - 375 PWR assemblies	2.0×10 ⁻⁵	3.2×10 ⁻²	1.6×10 ⁻⁵	14	7.2×10 ⁻³	7.0	2.8×10 ⁻²	(f)	(f)
15. Earthquake w/fire in WTB	2.0×10 ⁻⁴	5.8×10 ⁻⁵	2.9×10 ⁻⁸	2.1	1.1×10 ⁻⁵	5.2×10 ⁻⁶	2.1×10 ⁻⁶	(f)	(f)
16. LLW drum rupture in WTB	0.59	1.9×10 ⁻⁹	9.5×10 ⁻¹³	7.5×10 ⁻⁷	3.7×10 ⁻¹⁰	1.4×10 ⁻¹¹	5.6×10 ⁻¹¹	7.0×10 ⁻⁵	2.8×10 ⁻⁸

- a. Source: Kappes (1998, all). These frequency estimates are highly uncertain due to the preliminary nature of the repository design and are provided only to show potential accident sequence credibility. They represent conservative estimates based on the approach taken in Kappes (1998, all).
- b. CTHA = Cask Transfer/Handling Area, CTS = Canister Transfer System, ATS = Assembly Transfer System, DCHS = Disposal Container Handling System, WTB = Waste Treatment Building.
- c. To convert meters to feet, multiply by 3.2808.
- d. LCFi is the likelihood of a latent cancer fatality for an individual who receives the calculated dose. LCFp is the number of cancers probable in the exposed population from the collective population dose (person-rem). These values were computed based on a conversion of dose in rem to latent cancers as recommended by the International Council on Radiation Protection, as discussed in this section.
- e. For these cases, the involved workers are not expected to be vulnerable to exposure during an accident since operations are done remotely. Thus, involved worker impacts were not evaluated.
- f. For these events, involved workers would likely be severely injured or killed by the event; thus, no radiological impacts were evaluated. For the seismic event, as many as 39 people could be injured or killed in the Waste Handling Building, and as many as 36 in the Waste Treatment Building based on current staffing projections (TRW 1998c, pages 17 and 18).

potentially hazardous metals such as arsenic or mercury could be present in the high-level radioactive waste. However, they would be in a solid glass matrix that would make the exposure of workers or members of the public from operational accidents highly unlikely. Appendix A contains more information on the inventory of potentially hazardous materials.

Some potentially nonradioactive hazardous or toxic substances would be present in limited quantities at the repository as part of operational requirements. Such substances would include liquid chemicals such as cleaning solvents, sodium hydroxide, sulfuric acid, and various solid chemicals. These substances are in common use at other DOE sites. Potential impacts to workers from normal industrial hazards in the workplace including workplace accidents were derived from DOE accident experience at other sites. These impacts include those from accident scenarios involving the handling of hazardous materials and toxic substances as part of typical DOE operations. Thus, the industrial health and safety impacts to workers include impacts to workers from accidents involving such substances.

Impacts to members of the public would be unlikely because the hazardous materials would be mostly liquid and solid so that a release would be confined locally. (For example, chlorine used at the site for water treatment would be in powder form, so a gaseous release of chlorine would be unlikely. Furthermore, the repository would not use propane as a heating fuel, so no potential exists for propane explosions or fires.) The potential for hazardous chemicals to reach surface water during the Proposed Action would be limited to spills or leaks followed immediately by a rare precipitation or snow melt event large enough to generate runoff. Throughout the project, DOE would install engineered measures to minimize the potential for spills or releases of hazardous chemicals and would comply with written plans and procedures to ensure that, if a spill did occur, it would be properly managed and remediated. The Spill Prevention Control and Countermeasures Plan that would be in place for Yucca Mountain activities is an example of the plans DOE would follow under the Proposed Action.

The construction phase could generate as many as 3,500 drums [about 730 cubic meters (26,000 cubic feet)] of solid hazardous waste, and emplacement operations could generate as much as 100 cubic meters (3,500 cubic feet) per year (TRW 1999b, Section 6.1). Maintenance operations and closure would generate similar or smaller waste volumes. DOE would accumulate this waste in onsite staging areas in accordance with the regulations of the Resource Conservation and Recovery Act. Emplacement and maintenance operation could generate as many as 2,700 liters (1,700 gallons) of liquid hazardous waste annually (TRW 1999b, Section 6.1). The construction and closure phases would not generate liquid hazardous waste. 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. Although a specific accident scenario analysis was not performed for these activities, the analysis of human health and safety (see Chapter 4, Section 4.1.7.3) included these impacts to workers implicitly through the use of a data base that includes impacts from accidents involving hazardous and toxic materials. Impacts to members of the public would be unlikely.

H.3 Accident Scenarios During Retrieval

During retrieval operations, activities at the repository would be essentially the reverse of waste package emplacement, except operations in the Waste Handling Building would not be necessary because the waste packages would not be opened. The waste packages would be retrieved remotely from the emplacement drifts, transported to the surface, and transferred to a Waste Retrieval Storage Facility (TRW 1999b, Attachment I). This facility would include a Waste Retrieval Transfer Building where the waste packages would be unloaded from the transporter, transferred to a concrete storage unit, and moved to a concrete storage pad. The storage pad would be a 24- by 24-meter (80- by 80-foot) pad, about 1 meter (3.3 feet) thick, which probably would be about 3 kilometers (2 miles) over flat terrain from the

North Portal. Each storage pad would contain 14 waste packages. The number of pads required would depend on how many waste packages would be retrieved.

Because retrieval operations would be essentially the reverse of emplacement operations, accidents involving the disposal container during emplacement bound the retrieval operation. The bounding accident scenario during emplacement of the disposal container would be transporter runaway and derailment in the access tunnel (see Section H.2.1.4). This accident scenario would also bound accident scenarios during retrieval.

During storage, no credible accidents resulting in radioactive release of any measurable consequence would be expected to occur. This prediction is based on an evaluation of above-ground dry storage accident scenarios at the commercial sites under similar conditions, as evaluated in Appendix K.

In view of these considerations, DOE has concluded that the waste transporter derailment and the rockfall accident scenarios analyzed in Section H.2 would bound accident impacts during retrieval.

H.4 Accident Scenarios During Monitoring and Closure

During monitoring and closure activities, DOE would not move the waste packages, with the possible exception of removing a container from an emplacement drift for examination or drift maintenance. Such operations could result in a transporter runaway and derailment accident, but the frequency of release from such an event would be extremely low, as would the consequences, resulting in minimal risk. Thus, DOE expects the radiological impacts from operations during monitoring and closure to be very small.

H.5 Accident Scenarios for Inventory Modules 1 and 2

Inventory Modules 1 and 2 are alternative inventory options that the EIS considers. These modules involve the consideration of additional waste material for emplacement in the repository. They would involve the same waste and handling activities as those for the Proposed Action, but the quantity of materials received 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, DOE expects the accident impacts evaluated for the Proposed Action to bound those that could occur for Inventory Modules 1 and 2 because the same set of operations would be involved.

REFERENCES

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|-----------------------|---|
| BMI 1984 | BMI (Battelle Memorial Institute), 1984, <i>Repository Preclosure Accident Scenarios</i> , BMI/ONWI-551, Columbus, Ohio. [NNA.19900405.0032] |
| CDC 1998 | CDC (Centers for Disease Control and Prevention), 1998, <i>Chronic Diseases and Their Risk Factors: The Nation's Leading Causes of Death, A Report With Expanded State-by-State Information</i> , U.S. Department of Health and Human Services, Washington, D.C. [244026] |
| Chanin and Young 1998 | Chanin, D., and M. L. Young, 1998, <i>Code Manual for MACCS2: Preprocessor Codes for COMIDA2, FGRDCF, IDCF2</i> , NUREG/CR-6613, SAND97-0594, Volume 2, U.S. Nuclear Regulatory Commission, Washington, D.C. [243881] |

- CRC 1997 CRC Press, 1997, *CRC Handbook of Chemistry and Physics – A Ready-Reference Book of Chemical and Physical Data*, 78th edition, D. R. Lide, Editor, H.P.R. Frederikse, Associate Editor, Boca Raton, New York. [243741]
- DOE 1993 DOE (U.S. Department of Energy), 1993, *Recommendations for the Preparation of Environmental Assessments and Environmental Impact Statements*, Office of National Environmental Policy Act Oversight, Washington, D.C. [HQX.19930623.0005]
- DOE 1994 DOE (U.S. Department of Energy), 1994, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Draft Environmental Impact Statement*, Volume 1, Appendix D, Part B, page F-85, DOE/EIS-0203-D, Office of Environmental Management, Idaho Operations Office, Idaho Falls, Idaho. [211232]
- DOE 1995 DOE (U.S. Department of Energy), 1995, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs: Final Environmental Impact Statement*, DOE/EIS-0203-F, Office of Environmental Management, Idaho Operations Office, Idaho Falls, Idaho. [102617]
- DOE 1996a DOE (U.S. Department of Energy), 1996a, *Source Terms for Design Basis Event Analyses*, BBA000000-01717-0200-00019, Revision 00, Office of Civilian Radioactive Waste Management, Yucca Mountain Project Office, Las Vegas, Nevada. [MOL.19970203.0121]
- DOE 1996b DOE (U.S. Department of Energy), 1996b, *Preliminary MGDS Hazards Analysis*, B00000000-01717-0200-00130, Revision 00, Office of Civilian Radioactive Waste Management, Las Vegas, Nevada. [MOL.19961230.0011]
- DOE 1996c DOE (U.S. Department of Energy), 1996c, *DOE Standard: Accident Analysis for Aircraft Crash into Hazardous Facilities Area Saft*, DOE-STD-3014-96, Washington, D.C. [231519]
- DOE 1996d DOE (U.S. Department of Energy), 1996d, *Final Environmental Impact Statement for the Nevada Test Site and Off-Site Locations in the State of Nevada*, DOE/EIS-0243-F, Nevada Operations Office, Las Vegas, Nevada. [239895]
- DOE 1997a DOE (U.S. Department of Energy), 1997a, *Canister Transfer System Design Analysis*, BCB000000-01717-0200-000008, Revision 00, Office of Civilian Radioactive Waste Management, Yucca Mountain Project Office, Las Vegas, Nevada. [MOL.19980108.0054]
- DOE 1997b DOE (U.S. Department of Energy), 1997b, *DBE/Scenario Analysis for Preclosure Repository Subsurface Facilities*, BCA000000-01717-0200-00017, Revision 00, Office of Civilian Radioactive Waste Management, Las Vegas, Nevada. [MOL.19980218.0237]

- DOE 1997c DOE (U.S. Department of Energy), 1997c, *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste*, DOE/EIS-0200-F, Office of Environmental Management, Washington, D.C. [232988]
- DOE 1997d DOE (U.S. Department of Energy), 1997d, *Record of Decision for the Storage and Disposition of Weapons-Usable Fissile Materials, Final Programmatic Environmental Impact Statement*, Washington, D.C. [239425]
- DOE 1998a DOE (U.S. Department of Energy), 1998a, *Disposal Criticality Analysis Methodology Topical Report*, YMP/TR-004Q, Revision 0, Office of Civilian Radioactive Waste Management, Yucca Mountain Project Office, Las Vegas, Nevada. [MOL.19990308.0035]
- DOE 1998b DOE (U.S. Department of Energy), 1998b, *Viability Assessment of a Repository at Yucca Mountain*, DOE/RW-0508, Office of Civilian Radioactive Waste Management, Washington, D.C. [U.S. Government Printing Office, MOL.19981007.0027, Overview; MOL.19981007.0028, Volume 1; MOL.19981007.0029, Volume 2; MOL.19981007.0030, Volume 3; MOL.19981007.0031, Volume 4; MOL.19981007.0032, Volume 5]
- Eckerman, Wolbarst, and Richardson 1988 Eckerman, K. F., A. B. Wolbarst, and A. C. B. Richardson, 1988, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, Federal Guidance Report No. 11, EPA-520/1-88-020, U.S. Environmental Protection Agency, Office of Radiation Programs, Oak Ridge National Laboratory, Oak Ridge, Tennessee. [203350]
- Geomatrix and TRW 1996 Geomatrix and TRW (Geomatrix Consultants, Inc., and TRW Environmental Safety Systems Inc.), 1996, *Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada*, BA0000000-01717-2200-00082, Revision 0, San Francisco, California. [MOL.19961119.0034]
- ICRP 1975 ICRP (International Commission on Radiological Protection), 1975, *Report of the Task Group on Reference Man; a report prepared by a task group of Committee 2 of the International Commission on Radiological Protection*, Publication 23, Pergamon Press, Oxford, Great Britain. [237218]
- ICRP 1991 ICRP (International Commission on Radiological Protection), 1991, *1990 Recommendations of the International Commission on Radiological Protection*, Publication 60, Volume 21, Numbers 1-3, Pergamon Press, Elmsford, New York. [235864]
- Jackson et al. 1984 Jackson, J. L., H. F. Gram, K. J. Hong, H. S. Ng, and A. M. Pendergrass, 1984, *Preliminary Safety Assessment Study for the Conceptual Design of a Repository in Tuff at Yucca Mountain*, SAND83-1504, Sandia National Laboratories, Albuquerque, New Mexico and Los Alamos Technical Associates, Inc., Los Alamos, New Mexico. [NNA.19870407.0044]

- Kappes 1998 Kappes, J. A., 1998, *Preliminary Preclosure Design Basis Event Calculations for the Monitored Geologic Repository*, BC0000000-01717-0200-0001, Revision 00A, TRW Environmental Safety Systems Inc., Las Vegas, Nevada. [MOL.19981002.0001]
- Kennedy and Ravindra 1984 Kennedy, R. P., and M. K. Ravindra, 1984, "Seismic Fragilities for Nuclear Power Plant Risk Studies," *Nuclear Engineering and Design*, 79 (1984) 47-68, pp. R43-R64, North-Holland Physics Publishing Division, Elsevier Science Publishers B.V., Switzerland. [243985]
- Kimura, Sanzo, and Sharirli 1998 Kimura, C. Y., D. L. Sanzo, and M. Sharirli, 1998, *Crash Hit Frequency Analysis of Aircraft Overflights of the Nevada Test Site (NTS) and the Device Assembly Facility (DAF)*, UCRL-ID-131259, Lawrence Livermore National Laboratory, Livermore, California. [243218]
- Ma et al. 1992 Ma, C. W., R. C. Sit, S. J. Zavoshy, and L. J. Jardine, 1992, *Preclosure Radiological Safety Analysis for Accident Conditions of the Potential Yucca Mountain Repository: Underground Facilities*, SAND88-7061, Bechtel National, Inc., San Francisco, California. [NNA.19920522.0039]
- McFeely 1998 McFeely, S., 1998, "Radiological Activity in LLW," memorandum to J. Jessen (Jason Technologies Corporation), September 3, Fluor Daniel, Las Vegas, Nevada. [MOL.19990513.0045]
- McFeely 1999 McFeely, S. H., 1999, "Revised DAW Activity," personal communication with P. R. Davis (Jason Technologies Corporation), Fluor Daniel, Las Vegas, Nevada. [MOL.19990511.0393]
- Montague 1999 Montague, K., 1999, "WHB Inventory," personal communication with P. R. Davis (Jason Technologies Corporation), April 13, Duke Engineering Services, Las Vegas, Nevada. [MOL.19990615.0240]
- Mueller et al. 1996 Mueller, C., B. Nabelssi, J. Roglans-Ribas, S. Folga, A. Policastro, W. Freeman, R. Jackson, J. Mishima, and S. Turner, 1996, *Analysis of Accident Sequences and Source Terms at Treatment and Storage Facilities for Waste Generated by U. S. Department of Energy Waste Management Operations*, ANL/EAD/TM-29, Environmental Assessment Division, Argonne National Laboratory, Argonne, Illinois. [243561]
- Myers 1997 Myers, W. A., 1997, "Environmental Impact Statement (EIS) for the F-22 Follow-on Operational Testing and Evaluation and Weapons School Beddown, Nellis AFB, Nevada," memorandum with attachment to W. Dixon (Yucca Mountain Site Characterization Office, U. S. Department of Energy), received April 1997, Chief, Environmental Planning Division, Environmental Conservation & Planning Directorate, U. S. Air Force, Headquarters, Air Force Center for Environmental Excellence, Brooks Air Force Base, Texas. [MOL.19990602.0182]
- NRC 1997 NRC (U.S. Nuclear Regulatory Commission), 1997, *Standard Review Plan for Dry Cask Storage Systems, Final Report*, NUREG-1536, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, Washington, D.C. [232373]

- NRC 1998 NRC (U.S. Nuclear Regulatory Commission), 1998, *Standard Review Plan for Transportation Packages for Spent Nuclear Fuel*, NUREG-1617, Draft Report for Comment, Spent Fuel Project Office, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C. [242481]
- Ramsdell and Andrews 1986 Ramsdell, J. V., and G. L. Andrews, 1986, *Tornado Climatology of the Contiguous United States*, NUREG/CR-4461, PNL-5679, Pacific Northwest Laboratory, Richland, Washington, [236705]
- Rollstin, Chanin, and Jow 1990 Rollstin, J. A., D. I. Chanin, and H-N Jow, 1990, *MELCOR Accident Consequence Code System (MACCS), Model Description*, NUREG/CR-4691, SAND86-1562, Sandia National Laboratories, Albuquerque, New Mexico. [236740]
- SAIC 1998 SAIC (Science Applications International Corporation), 1998, *Nuclear Fuel Cycle Facility Accident Analysis Handbook*, NUREG/CR-6410, Reston, Virginia. [240909]
- Sandoval et al. 1991 Sandoval, R. P., R. E. Einziger, H. Jordan, A. P. Malinauskas, and W. J. Mings, 1991, *Estimate of CRUD Contribution to Shipping Cask Containment Requirements*, SAND88-1358, Sandia National Laboratories, Albuquerque, New Mexico. [223920]
- SNL 1987 SNL (Sandia National Laboratories), 1987, *Nevada Nuclear Waste Storage Investigations Project, Site Characterization Plan Conceptual Design Report*, SAND84-2641, Sandia National Laboratories, Albuquerque, New Mexico. [203922, Volume 1; 203538, Volume 2; 206486, Volume 3; 206487, Volume 4; 206488, Volume 5]
- Solomon, Erdmann, and Okrent 1975 Solomon, K. A., R. C. Erdmann, and D. Okrent, 1975, "Estimate of the Hazards to a Nuclear Reactor from the Random Impact of Meteorites," *Nuclear Technology*, Volume 25, pp. 68-71, American Nuclear Society, LaGrange Park, Illinois. [241714]
- Thompson 1998 Thompson, R. A., 1998, "F-15, F-16, and A-10 glide ratios," personal communication with P. R. Davis (Jason Technologies Corporation), September 1, Science Applications International Corporation, Las Vegas, Nevada. [MOL.19990511.0285]
- TRW 1997a TRW (TRW Environmental Safety Systems Inc.), 1997a, *Yucca Mountain Site Characterization Project Atlas 1997*, Las Vegas, Nevada. [MOL.19980623.0385]
- TRW 1997b TRW (TRW Environmental Safety Systems Inc.), 1997b, *Project Integrated Safety Assessment*, Chapter 7, "Radiological Safety Assessment of the Repository Through Preclosure," Draft C, Las Vegas, Nevada. [MOL.19980220.0047]
- TRW 1998a TRW (TRW Environmental Safety Systems Inc.), 1998a, *Repository Surface Design Site Layout Analysis*, BCB000000-01717-0200-00007, Revision 02, Las Vegas, Nevada. [MOL.19980410.0136]
- TRW 1998b TRW (TRW Environmental Safety Systems Inc.), 1998b, *Waste Emplacement System Description Document*, BCA000000-01717-1705-00017, Revision 00, Las Vegas, Nevada. [MOL.19980519.0234]

- TRW 1998c TRW (TRW Environmental Safety Systems Inc.), 1998c, *Monitored Geologic Repository Operations Staffing Report*, BC0000000-01717-5705-00021, Revision 00, Las Vegas, Nevada. [MOL.19981211.0036]
- TRW 1999a TRW (TRW Environmental Safety Systems Inc.), 1999a, *Engineering File – Subsurface Repository*, BCA000000-01717-5705-00005, Revision 02 with DCN1, Las Vegas, Nevada. [MOL.19990622.0202, document; MOL.19990621.0157, DCN1]
- TRW 1999b TRW (TRW Environmental Safety Systems Inc.), 1999b, *Repository Surface Design Engineering Files Report*, BCB000000-01717-5705-00009, Revision 03, Las Vegas, Nevada. [MOL.19990615.0238]
- Tullman 1997 Tullman, E. J., Lieutenant Colonel, USAF, 1997, “Nellis Airspace and Crash Data for Yucca Mountain Hazard Analysis,” letter with enclosure to W. E. Barnes (Yucca Mountain Site Characterization Office), U. S. Department of Energy, June 5, USAF/DOE Liaison Office, U.S. Air Force, U.S. Department of the Air Force, U. S. Department of Defense, Las Vegas, Nevada. [MOL.19970806.0389]
- USAF 1999 USAF (U.S. Air Force), 1999, *Renewal of the Nellis Air Force Range Land Withdrawal: Legislative Environmental Impact Statement*, Air Combat Command, U.S. Department of the Air Force, U. S. Department of Defense, Nellis Air Force Base, Nevada. [243264]
- USGS 1998 USGS (U.S. Geological Survey), 1998, *Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada, Final Report*, U.S. Department of the Interior, Oakland, California. [MOL.19980619.0640]
- USN 1996 USN (U.S. Navy), 1996, *Department of the Navy Final Environmental Impact Statement for a Container System for the Management of Naval Spent Nuclear Fuel*, DOE/EIS-0251, in cooperation with the U.S. Department of Energy, Naval Nuclear Propulsion Program, U.S. Department of the Navy, U.S. Department of Defense, Arlington, Virginia. [227671]
- Wade 1998 Wade, 1998, personal communication with P. R. Davis (Jason Technologies Corporation), Yucca Mountain Site Characterization Office, U.S. Department of Energy, Las Vegas, Nevada. [MOL.19990511.0284]
- Walck 1996 Walck, M. C., 1996, *Summary of Ground Motion Prediction Results for Nevada Test Site Underground Nuclear Explosions Related to the Yucca Mountain Project*, SAND95-1938, Sandia National Laboratories, Albuquerque, New Mexico. [MOL.19970102.0001]
- Wilmot 1981 Wilmot, E. L., 1981, *Transportation Accident Scenarios for Commercial Spent Fuel*, SAND80-2124, TTC-0156, Transportation Technology Center, Sandia National Laboratories, Albuquerque, New Mexico. [HQO.19871023.0215]