Final

Environmental Impact Statement

for a

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

Volume IV

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

DOE/EIS-0250

February 2002

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.

CFR	Code of Federal Regulations
DOE	U.S. Department of Energy (also called *the Department*)
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
FR	*Federal Register*
LCF	latent cancer fatality
MTHM	metric tons of heavy metal
NEPA	National Environmental Policy Act, as amended
NRC	U.S. Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act, as amended
PM_{10}	particulate matter with an aerodynamic diameter of 10 micrometers or less
$PM_{2.5}$	particulate matter with an aerodynamic diameter of 2.5 micrometers or less
REMI	Regional Economic Models, Inc.
RMEI	reasonably maximally exposed individual
Stat.	United States Statutes
TSPA	Total System Performance Assessment
U.S.C.	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
$10^1 = 10 \times 1 = 10$
$10^2 = 10 \times 10 = 100$
and so on, therefore,
$10^6 = 1{,}000{,}000$ (or 1 million)

Negative Powers of 10
$10^{-1} = 1/10 = 0.1$
$10^{-2} = 1/100 = 0.01$
and so on, therefore,
$10^{-6} = 0.000001$ (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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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 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 (DIRS 147496-CRWMS M&O 2000, 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 intends to identify accidents and evaluate their impacts as required to support the License Application for the proposed repository that it would 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. To meet licensing requirements, the results from the accident analysis 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.

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 (DIRS 104699-Jackson et al. 1984, all; DIRS 100181-SNL 1987, all; DIRS 101930-Ma et al. 1992, all; DIRS 104693-Yook et al. 1984, all). More recent evaluations include DIRS 104695-CRWMS M&O (1996, all); DIRS 100204-CRWMS M&O (1996, all); DIRS 100217-CRWMS M&O (1997, all); DIRS 102702-CRWMS M&O (1997, all); DIRS 103237-CRWMS M&O (1998, all); DIRS 147496-CRWMS M&O (2000, all); DIRS 150276-CRWMS M&O (2000, all); DIRS 149759-CRWMS M&O (1999, all); and DIRS 137064-CRWMS M&O (1999, 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 (DIRS 101941-USN 1996, all; DIRS 103213-DOE 1996, all).

Radiological accidents involve an initiating event that could 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. In addition to these credible accidents, DOE evaluated a repository aircraft crash event. Even though such an event was determined to be not credible (annual probability less than one in 10 million), DOE decided to evaluate it because such an accident could have large impacts. The results of the evaluation are presented in Section H.2.1.5.1.

H.2.1.1 Internal Events – Waste Handling Building and Emplacement System

The most recent repository accident scenario analysis for internal and external events in the Waste Handling Building (DIRS 155734-DOE 2001, pp. 5-1 to 5-48) addressed Nuclear Regulatory Commission requirements in 10 CFR Part 63. The analysis was a comprehensive evaluation of repository operations to identify accident sequences that could lead to a radioactive release. Detailed analyses involving the use of event trees and fault trees were performed on the sequences to estimate accident frequencies. The frequency evaluation was used to identify Category 1 accidents (a frequency of once per 100 years or greater), Category 2 accidents (a frequency of between once in 100 years and once in 1 million years), or beyond-design-basis events (a frequency less than once in 1 million years).

A review of these evaluations indicated that they were valid for use in the EIS with a few exceptions and revisions (noted below).

The evaluation used to identify internal accidents did not evaluate criticality events (see Glossary for event description) quantitatively (DIRS 103237-CRWMS M&O 1998, p. 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 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 (DIRS 104699-Jackson et al. 1984, p. 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 of the spent nuclear fuel (DIRS 104441-YMP 1998, p. 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 (DIRS 103237-CRWMS M&O 1998, p. 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 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 (DIRS 103683-DOE 1994, Volume 1, Appendix D, p. F-85) and, thus, would limit the source term.

- During the monitoring and closure phase of operations, water would have to enter a waste package that contained fuel with sufficient fissionable material to cause a criticality. Water would have to

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 computed as the product of the frequency of an accident and the consequences that result.

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.

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.

- Evaluated criticality events (DIRS 147496-CRWMS M&O 2000, pp. 5-41 and 5-42) would be beyond-design-basis events with a frequency of less than once in 1 million years (probability of less than 0.000001 per year). Accordingly, DOE did not evaluate these events further as part of the safety assessment process to evaluate compliance with Nuclear Regulatory Commission safety regulations.

Considering these factors, the criticality event is not expected to be a large potential contributor to risk.

Table H-1 lists the accidents that DOE considered for analysis in this EIS. Section A of the table lists the Category 1 accidents as derived in DIRS 147496-CRWMS M&O 2000, p. 5-21, Section B lists the Category 2 accidents from the *Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation* (DIRS 147496-CRWMS M&O 2000, p. 5-22), and Section C lists the accidents retained for analysis from the Draft EIS. Some of these accidents were eliminated from further consideration based on evaluations discussed later in this section.

The No. column in Table H-1 provides a numerical identifier that corresponds to the identifier used in the source document. The Location column lists the repository location designator where the accident is assumed to occur. The Accident column describes the accident. The MAR column lists the material at risk; that is, the amount of radioactive material involved in the accident. The Frequency column lists an estimate of the annual probability of the accident. The EIS disposition column describes whether the accident was retained for further analysis, bounded by another accident in the table, or eliminated from further consideration based on other reasons such as design change or reduced probability estimates. The basis for these evaluations is provided in subsequent sections of this appendix.

DOE selected fuel from pressurized-water reactors for most of the accident analyses because it would be the predominant fuel handled at the proposed repository (Appendix A, p. A-15), and because this fuel would produce higher doses than boiling-water reactor fuel (see Section H.2.1.4.4) for equivalent accidents. The analysis retained one accident involving boiling-water reactor fuel (Table H-1, No. 13) to confirm this conclusion (see Section H.2.1.5).

The following paragraphs contain details of the postulated accident scenarios in each location.

H.2.1.1.1 *Cask/Carrier Transport and Handling Area*

DOE would handle incoming transportation casks in the Cask/Carrier Transport and Handling Area. The casks would be unloaded from carriers and impact limiters would be removed to facilitate handling of the casks. The Draft EIS conservatively assumed that damage to the casks would occur if they were dropped from heights greater than 2 meters (6.6 feet) after removal of the impact limiters. Accordingly, four accidents were defined (Numbers 1, 3, 5, and 7 from Table H-1) for analysis. However, DOE has determined that transportation casks would be unlikely to be damaged if dropped from the maximum heights (7.1 meters or 23 feet) to which the casks would be lifted during handling operations. A recent analysis of transportation cask response under accident conditions concluded (DIRS 152476-Sprung et al. 2000, p. 2-7) that truck cask seals are not compromised by impacts at any orientation onto an unyielding surface at speeds as high as at least 145 kilometers (90 miles) per hour even assuming that the impact limiters are fully crushed before the impact. For rail casks (DIRS 154930-NRC 2000, p. 2-8), seal leakage could occur at impact speeds as low as 97 kilometers (60 miles) per hour. At the proposed repository, the casks would be lifted a maximum of 7.1 meters (23 feet) according to the Draft EIS, Volume II, Appendix H, p. H-4. A drop from this distance would produce an impact velocity of 42

Table H-1. Internal-event-initiated accidents evaluated for further analysis.[a]

No.	Location[b]	Accident[c]	MAR[d] (PWR SFAs)		Frequency (events/year)	EIS disposition
		A. Category 1 Accidents (DIRS 155734-DOE 2001, Table 5-5)				
1-01	P	SFA drop on SFA	2		0.2	Bounded by 1-07[e]
1-02	P	SFA collision	1		0.04	Bounded by 1-07
1-03	P	SFA drop on empty basket	1		0.04	Bounded by 1-07
1-04	P	SFA drop on SFA in rack	2		0.2	Bounded by 1-07
1-05	P	Basket drop onto basket in rack	8		0.04	Bounded by 1-07
1-06	P	Basket drop onto basket in storage (transfer into pool storage)	8		0.04	Same as 1-07
1-07	P	Basket drop onto basket in pool (transfer out of pool storage)	8		0.04	Retained
1-08	P	Basket drop onto transfer cart/floor (transfer out of pool storage)	4		0.04	Bounded by 1-07
1-09	P	Basket drop into pool	4		0.04	Bounded by 1-07
1-10	C	Basket drop onto cell floor	4		0.04	Bounded by 1-11
1-11	C	Basket drop onto basket in dryer	8		0.04	Retained
1-12	C	SFA drop on another SFA in dryer	2		0.2	Bounded by 1-11
1-13	C	SFA drop on cell floor	1		0.2	Bounded by 1-11
1-14	C	SFA drop on SFA in DC	2		0.2	Bounded by 1-11
		B. Category 2 Accidents (DIRS 155734-DOE 2001, Table 5-6)				
2-01	P	Basket collision during transfer	4		0.007	Bounded by 1-07
2-02	P	Uncontrolled descent of transfer cart	4		0.007	Bounded by 1-07
2-03	P	Handling equipment drop on basket	4		0.002	Bounded by 1-07
2-04	C	Handling equipment drop on basket	4		0.00007	Bounded by 1-11
2-05	D	Unsealed DC collision	21		0.002	Bounded by 2-06
2-06	D	Unsealed DC drop	21		0.008	Retained
2-07	D	Handling equipment drop on DC	21		0.0001	Bounded by 2-06
2-08	C	Unsealed shipping cask drop	26		0.009	Retained
2-09	P	Unsealed shipping cask drop	26		0.009	Retained
		C. Accidents evaluated in Draft EIS				
Event	Location	Accident	MAR[d]	Filters	Frequency	Disposition
1	A	6.9-meter drop of shipping cask	61 BWR	No	0.00045	Eliminated
3	A	7.1-meter drop of shipping cask	26 PWR	No	0.00061	Eliminated
5	A	4.1-meter drop of shipping cask	61 BWR	No	0.0014	Eliminated
7	A	4.1-meter drop of shipping cask	26 PWR	No	0.0019	Eliminated
9	B	6.3-meter drop of multicanister overpack	N-Reactor fuel	Yes	0.00045	Eliminated
10	B	6.3-meter drop of multicanister overpack	N-Reactor fuel	No	0.00000022	Eliminated
11	C	5-meter drop of transfer basket (onto another basket)	8 PWR	Yes	0.011	Retained (same as 1-11)
12	C	5-meter drop of transfer basket (onto another basket)	8 PWR	No	0.00000028	Eliminated
13	C	7.6-meter drop of transfer basket (onto another basket)	16 BWR	Yes	0.0074	Retained
14	C	7.6-meter drop of transfer basket (onto another basket)	16 BWR	No	0.00000019	Eliminated
15	D	6-meter vertical drop of DC	21 PWR	Yes	0.0018	Retained (same as 2-06)
16	D	6-meter vertical drop of DC	21 PWR	No	0.00000086	Eliminated
19	E	Transporter runaway and derailment	21 PWR	Yes	0.00000012	Retained (without filters)

a. Source: Modified from DIRS 147496-CRWMS M&O (2000, pp. 5-21 and 5-22).

b. Location designators: A = Cask/Carrier Transport and Handling Area; B = Canister Transfer System; C = Assembly Transfer System Spent Fuel Handling; D = Disposal Container Handling System; E = Waste Emplacement and Subsurface Facility; P = Assembly Transfer System or Blending Inventory Pool.

c. To convert meters to feet, multiply by 3.2808.

d. MAR = material at risk; SFA = spent fuel assembly, BWR = boiling-water reactor, PWR = pressurized-water reactor, DC = disposal container.

e. Bounding is based on the highest material at risk independent of event frequency.

kilometers (26 miles) per hour (see Section H.2.1.4.2). Thus, shipping cask seal leakage would be unlikely from an accidental drop from the maximum lift heights during cask handling operations. This conclusion is consistent with DIRS 147496-CRWMS M&O (2000, all) because no accidents were identified in the Cask/Carrier Transport and Handling Area with the potential to release radioactive materials. Therefore, DOE eliminated accidents 1, 3, 5, and 7 from further consideration, as indicated in the EIS disposition column of Table H-1.

H.2.1.1.2 *Canister Transfer System*

Some spent nuclear fuel and high-level radioactive waste would arrive at the repository in canisters suitable for direct placement in disposal containers. The canister transfer system would unload these canisters from a transportation cask and load them in a disposal container in the Waste Handling Building confinement system. This system would include a filtration function that would ensure that any radioactive material that could be released would pass through high-efficiency particulate air filters before exhausting to the atmosphere. During these operations, canister drops could release radioactive material. Accident evaluations performed for the Draft EIS, Volume II, Appendix H, p. H-5 determined that the drop of a canister containing N-Reactor fuel could produce a radioactive release, and that this accident would bound other accidents involving canisters. Two such accidents, Numbers 9 and 10 as listed in Table H-1, were considered. However, since the publication of the Draft EIS, DOE has established waste acceptance criteria that specify (DIRS 110306-DOE 1999, p. 20) that waste canisters arriving at the proposed repository for emplacement (1) withstand drops from the maximum lift height during repository handling operations without a release, or (2) if a drop would result in a release, ensure that resulting doses would be within requirements established by the Nuclear Regulatory Commission assuming no filtration of released radionuclides by the Waste Handling Building ventilation system. As a result of these requirements, DOE did not evaluate impacts from canister drops. However, a drop of a defective canister could produce a release. The probability that a canister could be manufactured with a defect significant enough to produce a failure if dropped has been conservatively estimated to be 3×10^{-6} per canister (DIRS 154327-DOE 2000, p. 1). To determine the annual probability of a release, it is necessary to combine the number of canister lift operations per year with the probability of a drop and the probability of a defective canister. The estimated maximum number of DOE canister lifts per year would be 2,114 (DIRS 152151-CRWMS M&O 2000, p. 2-3), and the estimated probability of a drop per lift would be 1.4×10^{-5} (DIRS 103237-CRWMS M&O 1998, p. 14). Thus, the probability of a release involving a drop of a defective canister is:

2,114 canister lifts per year (maximum) $\times$ 1.4×10^{-5} canister drops per year $\times$ 10^{-6} defect per canister = 8.9×10^{-8} releases per year.

This probability is below the credibility limit established by DOE for environmental impact assessment (DIRS 104601-DOE 1993, p. 28) of once in 10 million years (1×10^{-7} per year). Therefore, DOE did not evaluate this accident scenario further.

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 casks in the cask unloading pool. Next, they would be moved to the assembly holding pool or the fuel blending inventory pools 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. In the cask preparation pit of the assembly transfer system, the lid would be removed from the shipping cask and the cask would be transferred to the cask unloading pool. During transfer of the shipping cask from the pit to the pool, the cask could be accidentally dropped onto the cask preparation pit floor or the transfer pool floor (DIRS 147496-CRWMS M&O 2000, p. 5-24). These accidents are listed as 2-08 and 2-09 in Table H-1. However, the number of fuel assemblies has been reduced from 26 to 24 for this accident. The 26 pressurized-water reactor fuel assembly case was selected for the preclosure safety assessment (DIRS 147496-CRWMS M&O 2000, p. 5-24) to represent an upper limit on the number of pressurized-water reactor fuel assemblies in a rail transportation cask. The most probable number of pressurized-water reactor assemblies in a rail transportation cask is 24, as discussed in Appendix J, Section J.1.4.2. The estimated frequency of these accidents would be 0.0087 per year (DIRS 147496-CRWMS M&O 2000, p. 5-22), based on the number of unsealed shipping cask handling operations expected at the proposed repository and the failure probability of the shipping cask handling crane (DIRS 150276-CRWMS M&O 2000, Attachment VII, pp. VII-1 through VII-20).

The cask preparation pit and unloading pool would be in the Waste Handling Building confinement system. This system would include a filtration function that would ensure that any radioactive material that could be released would pass through high-efficiency particulate air filters before exhausting to the atmosphere. Thus, for these two unsealed shipping cask drop accidents, any radioactive material released from the cask would be filtered by the Waste Handling Building confinement system before being released to the environment. For this EIS, DOE examined the probability of failure of the confinement filtration system in conjunction with these accidents. The filtration system failure probability for a 24-hour period would be 1.7×10^{-7} (DIRS 137064-CRWMS M&O 1999, all). Thus, the probability of filtration system failure in conjunction with an unsealed shipping cask drop would be 8.7×10^{-3} multiplied by $1.7 \times 10^{-7} = 1.5 \times 10^{-9}$ per year. This probability is well below the credibility limit established by DOE (DIRS 104601-DOE 1993, p. 28) of once in 10 million years (1×10^{-7} per year). Therefore, DOE did not evaluate this accident scenario further.

After the shipping casks were placed in the pool with lids removed, the spent fuel assemblies (either bare or canistered assemblies) would be removed and placed in storage racks or in transfer baskets. The transfer baskets could contain either four pressurized-water reactor assemblies or eight boiling-water reactor assemblies. A loaded transfer basket would be loaded into the transfer cart. All of these operations would take place underwater in the 15-meter- (50-foot)-deep pool. DOE evaluated accidental drops of individual spent fuel assemblies or of transfer baskets during these operations. Accidents involving these underwater operations are listed in Table H-1 as accidents 1-01 through 1-09 and 2-01 through 2-03, and 11 through 14. In examining these accidents, DOE determined that accident 1-06 or 1-07 would produce the maximum radiological impacts because the amount of radioactive material released would be directly proportional to the amount of spent nuclear fuel involved in the accident (MAR column in Table H-1). Therefore, DOE retained only accident 1-07 for further evaluation in the EIS, as indicated in the EIS disposition column in Table H-1. This accident, based on assumptions in Section H.2.1.4, would produce the maximum consequences (impacts) for all fuel-handling accidents in the pool and, therefore, would bound accidents 1-01 through 1-06. Furthermore, this accident would bound accidents 2-01, 2-02, and 2-03 because more material at risk would be involved in 1-07.

The next accidents considered in Table H-1 involve events that could occur after the spent fuel assemblies were removed from the pool and prepared for disposal container loading. Spent fuel assemblies would be brought to the assembly transfer system hot cell from the pool for drying by the transfer cart, which would hold one transfer basket. After the cart arrived in the cell, the basket would be lifted out of the cart and placed in the dryer. After drying, the assemblies would be lifted out of the dryer vessel and placed in the disposal container in the hot cell. During these operations, assemblies could be dropped to the hot cell floor, into the dryer, or into the disposal container. These accidents are listed in Table H-1 as events 1-10 through 1-14, 2-04, and 11 through 14. Because these accidents would occur in the Waste Handling Building confinement system, radioactive releases would be filtered by the confinement filtration system. As noted above, a recent assessment (DIRS 137064-CRWMS M&O 1999, all) estimated that the filtration system failure probability has been reduced to 1.7×10^{-7}. Thus, neither accident involving filter system failure in conjunction with a transfer basket drop (accidents 12 and 14) would be credible (probability of greater than once in 10 million years or 1×10^{-6} per year). Accident 12 would have a probability of $1.1 \times 10^{-2} \times 1.7 \times 10^{-7}$ or 1.9×10^{-9} per year and accident 14 would have a probability of $7.4 \times 10^{-3} \times 1.7 \times 10^{-7}$ or 1.3×10^{-9} per year. The remaining accidents would be bounded by accident 1-11, which would involve the highest radionuclide inventory (material at risk) and thus would provide the largest source term and impacts.

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. DOE examined the details of these operations and identified several accidents that could occur. These are accidents 2-05, 2-06, 2-07, and 15 and 16 in Table H-1. The first three accidents are bounded by accident 2-06 because this event would impart the most energy to the material at risk (21 pressurized-water reactor fuel assemblies) and thus would result in the most fuel damage leading to the highest release of radioactive material (see Section H.2.1.4). Accident 15 is the same as accident 2-06, and DOE eliminated accident 16 because the drop of a disposal container concurrent with a failure of the filtration system would be incredible based on a recent evaluation of the failure of the system (DIRS 137064-CRWMS M&O 1999, all) that, as noted above, estimated the failure probability as 1.7×10^{-7} for a 24-hour period. The combined probability in this case is $1.8 \times 10^{-3} \times 1.7 \times 10^{-7}$ or 3.1×10^{-10} per year, well below the credibility level of 1×10^{-7} per year.

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 shielded transporter car for waste package transportation. The transporter car would be moved into the waste emplacement area by an electric locomotive and the waste package would be placed in the emplacement drift. The only accident in Table H-1 that would involve subsurface emplacement operations is accident 19 from Section C (transporter runaway and derailment). DOE has retained this accident for evaluation but has modified it such that the release would not be filtered. This is because the current design concept (DIRS 153849-DOE 2001, all) does not contain an automatic subsurface filter system (DIRS 150941-CRWMS M&O 2000, p. 4-23), as did the design concept evaluated in the Draft EIS. The design concept does retain filtration capability (DIRS 150941-CRWMS M&O 2000, p. 4-23), but it would be a manual system that might not be available in time to provide filtration of the release from the transporter runaway accident. Final design details of the transporter system have not been established. A recent evaluation of transporter accident potential determined that several design features (five of the six evaluated) could reduce the probability of transporter runaway to less than 0.0000001 per year (DIRS 149105-CRWMS M&O 2000, all). If DOE selected any of these features, the transporter runaway accident retained for analysis in this evaluation could become not credible.

A recent evaluation of potential waste package accidents during emplacement activities considered a comprehensive evaluation of accident initiating events (DIRS 150198-CRWMS M&O 2000, all). This evaluation concluded that either the accident-initiating event would not be credible or would be within the design basis of the waste package. However, one event, a rockfall involving a rock weight of more than 6 metric tons (6.6 tons) (assumed to be large enough to fail the waste package), would have a probability of 5×10^{-7} per year. While this event would not be credible under Nuclear Regulatory

Commission safety regulations (DIRS 147496-CRWMS M&O 2000, p. 4-18), it would be credible based on DOE guidelines for environmental impact analysis (DIRS 104601-DOE 1993, p. 28) and, therefore, the Department evaluated it further. The evaluation of a failure of a waste package after emplacement (DIRS 150276-CRWMS M&O 2000, all) assumed that the waste package would fail from unspecified causes and that all of the pressurized-water reactor fuel rods in 21 fuel assemblies would rupture and release all radioactive gases in them. The calculated site boundary dose from this event would be 0.0027 rem (DIRS 150276-CRWMS M&O 2000, p. X-48). As discussed in Section H.2.1.5, this dose would be far less than that produced from the transporter runaway and derailment accident, which would damage the waste package being transported for emplacement. Therefore, the rockfall on a waste package event is bounded by the transporter runaway accident, and is not evaluated further.

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 (DIRS 104508-CRWMS M&O 1999, p. 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) (DIRS 104508-CRWMS M&O 1999, p. 42).

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 (DIRS 104508-CRWMS M&O 1999, p. 55). The evaporation process would reduce the volume of the liquid waste stream by 90 percent (DIRS 101816-DOE 1997, 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 (DIRS 104508-CRWMS M&O 1999, p. 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 (DIRS 101816-DOE 1997, all) and supporting documentation (DIRS 103688-Mueller et al. 1996, all) to aid in identifying potential accident scenarios and evaluating radionuclide source terms. DOE based the information in the Waste

Management Programmatic EIS on high- and low-level waste handling and treatment experience at various sites. At those sites, DOE has stored, packaged, treated, and transported these wastes for several decades and has compiled an extensive database of information relevant to accident assessments (for example, safety analysis reports, unusual occurrences). 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 (DIRS 103688-Mueller et al. 1996, p. 13). Drum-handling accidents are expected to dominate the risk of exposure to workers (DIRS 103688-Mueller et al. 1996, p. 93).

The estimated frequency of an accident involving drum failure is about 0.0001 failure per drum operation (DIRS 103688-Mueller et al. 1996, p. 39). The total number of drums containing grouted aqueous waste would be 2,280 per year (DIRS 100248-CRWMS M&O 1997, p. 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:

Frequency = 2,280 aqueous (grouted) low-level waste drums per year
× 2 handling operations per drum
× 0.0001 failure per handling operation
= 0.46 aqueous (grouted) low-level waste drum failures per year.

The number of solid-waste grouted drums produced would be 2,930 per year (DIRS 100248-CRWMS M&O 1997, p. 35). Assuming two handling operations and the same failure rate yields a frequency of drum failure of:

Frequency = 2,930 solid low-level waste drums per year
× 2 handling operations per drum
× 0.0001 failure per handling operation
= 0.59 solid low-level waste drum failures per year.

Failure of these drums would result in a release of radioactive material, which later sections (H.2.1.4.5, H.2.1.5) 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 (DIRS 104508-CRWMS M&O 1999, Attachment IV, Figure 6)], 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 (DIRS 100204-CRWMS M&O 1996, p. 15), was developed by an extensive review of relevant sources and known or predicted geologic, seismologic, hydrologic, and

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: DIRS 146897-CRWMS M&O (2000, Table 6-1).

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: DIRS 146897-CRWMS M&O (2000, p. 31).

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 an evaluation to determine the applicability of the event to the repository operations, and is based in part on evaluations previously reported in (DIRS 100204-CRWMS M&O 1996, all; DIRS 147496-CRWMS M&O 2000, Section 5; DIRS 104508-CRWMS M&O 1999, 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)
- 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.** This assessment evaluated the probability of an aircraft crash on the proposed Yucca Mountain Repository to see if such an event would be reasonably foreseeable and, therefore, a candidate for consequence analysis. Since the publication of the Draft EIS, new information and data have become available. The information and data include the following:

   a. The design concept of the Waste Handling Building has been updated. The flexible design concept includes thinner walls in the upper regions of the building, as well as a smaller footprint for areas of the building where the waste would be out of the storage pools. As a consequence, the target area for the aircraft impact has changed.

   b. A recent assessment of aircraft crash probability contains information useful for the reassessment (DIRS 154930-NRC 2000, all).

   c. Since March 1999, DOE has collected aircraft overflight data to evaluate the frequency of overflights in the region of the repository. Because this information was not available for the Draft EIS, that evaluation assumed a constant overflight density in the entire flight corridor [49 kilometers (30 miles)] that encompasses the repository. The overflight data indicate that the flight density over the repository site is less than the average for the flight corridor. (The repository site is at the extreme western edge of the flight corridor.) DOE used this recent overflight data in the assessment.

   d. The repository design could include a surface aging facility, which DOE is considering as an option to enable aging of commercial nuclear fuel prior to emplacement. The aging process would reduce the heat generation rate from spent nuclear fuel. Thus, aging could be used to control subsurface temperatures. DOE evaluated the aircraft crash probability and consequences for this facility.

   Aircraft Overflights. As noted in the Draft EIS (Appendix H, page H-10), the only aircraft that fly over the repository airspace are military aircraft from Nellis Air Force Base. This conclusion is also derived in a recent aircraft crash probability analysis (DIRS 108290-CRWMS M&O 1999, Section 7.1). The only information available on the frequency of military overflights at the time of the Draft EIS analysis was the total number of flights in the 47-kilometer- (29-mile)-wide flight corridor used by Nellis Air Force Base, which includes the repository at its western edge. The Draft EIS used the Uniform Overflight Density Model to estimate the frequency of aircraft crashes on the site. However, in March 1999, DOE began actual counting of aircraft overflying a 10-kilometer- (6-mile)-wide airway with the repository at the center. To date, overflight data have been processed on a quarterly basis. The results through June 30, 2001, are as follows (DIRS 155256-Morissette 2001, all; DIRS 155257-Morissette 2001, all; DIRS 156117-Morissette 2001, all; DIRS 154768-Monette 2001, all):

Fiscal year	Quarter	Number of overflights
1999	Third	361
1999	Fourth	274
2000	First	424
2000	Second	328
2000	Third	648
2000	Fourth	326
2001	First	490
2001	Second	370
2001	Third	769

The average number of quarterly overflights from these data was 443, giving an annual average of 1,773. This value is less than the number of flights that would be expected in the 10-kilometer- (6-mile)-wide airway if the 13,000 flights per year used in the Draft EIS were evenly distributed over the 49-kilometer- (30-mile)-wide corridor ($13{,}000/30 \times 6 = 2{,}690$). In other words, actual flightpaths are concentrated east of the repository. In the Draft EIS assessment, DOE used the Uniform Overflight Density Model because site-specific overflight information was not available. However, because repository-specific overflight data are now available, DOE decided to use the Nuclear Regulatory Commission Airway Model in the reassessment. This model was also used in DIRS 108290-CRWMS M&O (1999, p. 26), which noted that it gives somewhat higher crash estimates than the Uniform Overflight Density Model when applied to the 49-kilometer-wide corridor case. Therefore, the results in this appendix are conservative based on the selection of the model.

DOE also examined the potential for a change in overflight numbers at the time of repository operation due to aircraft operational changes contemplated by the Air Force. The only known planned change in future activities involves the addition of F-22 fighter aircraft at Nellis Air Force Base. The additional aircraft would increase flight activities by 2 to 3 percent over current activities (DIRS 104707-Myers 1997, p. 3).

Commercial air traffic is not allowed in the air space over the proposed repository location. An inadvertent commercial flight over the restricted repository air space followed by a crash into the repository would be significantly less probable than the military crash probability evaluated in this analysis.

Airway Model. The Airway Model from NUREG-0800 (DIRS 152082-NRC 1981, Section 3.5.1.6, p. 3.5.1.6-3) is:

$$P_{FA} = C \times N \times A/w$$

where:

P_{FA} is the probability per year of the aircraft crashing into the facility

C is the crash rate in crashes per mile flown
N is the number of flights per year along the airway
A is the effective area of the facility (square miles)
w is the width of the airway (miles).

This model was used by the analysis in the *Draft Environmental Impact Statement for the Construction and Operation of an Independent Spent Fuel Storage Installation on the Reservation of the Skull Valley Band Indians and the Related Transportation Facility in Tooele County, Utah* (DIRS 152001-NRC 2000, all), and was modified to account for the fact that Air Force fighter pilots would be likely to attempt to direct aircraft away from ground structures before ejecting if they could maintain flight control. The Nuclear Regulatory Commission accepted this modification (DIRS 154930-NRC 2000, p. 198). DOE considered this modification to be applicable to the repository crash analysis based on similar conditions, including overflights in high-altitude cruise mode, similar pilot training, and similar aircraft. The modification consisted of separating the crash probability into two components, P_1 and P_2, where the overall crash probability P_{FA} is the sum of P_1 and P_2. The P_1 component represents the probability of an aircraft crashing on the repository as a result of engine failure or other malfunction with the pilot retaining control of the aircraft. P_2 is the probability of an aircraft crashing on the repository due to engine failure or other malfunction with the pilot not retaining control of the aircraft. The analysis then reformulated the overall crash probability as follows:

$$P_{FA} = P_1 + P_2 = C \times N \times A/w \times R_1 + C \times N \times A/w \times R_2$$

where:

R_1 = probability that the crash is of the type such that the pilot retains control of the aircraft but is unable to guide the aircraft away from repository structures. This is the product of the probability that the pilot retains control of the aircraft for a time that is sufficient to guide the aircraft away from the facility (0.9) and the probability that the pilot will still not be able to guide the aircraft away from the structures (0.05). The assessment estimated the value of R_1 at 0.045 (DIRS 154930-NRC 2000, p. 197) based on crash data, pilot training and experience, and other factors.

R_2 = probability that the crash is of the type such that the pilot does not retain control of the aircraft and is, therefore, unable to guide the aircraft away from the repository before ejecting. The assessment estimated the value of R_2 as 0.1 (DIRS 154930-NRC 2000, p. 197). This value is based on crash data which indicate that a pilot would retain control of the aircraft with sufficient time to steer the plane away from surface structures for 90 percent of F-16 crashes (DIRS 154930-NRC 2000, p. 197).

Based on these considerations, the overall crash rate becomes:

$$P_{FA} = C \times N \times A/w \times (0.045) + C \times N \times A/w \times (0.1) = C \times N \times A/w \times (0.145).$$

Using this formula, DOE evaluated the crash rate for both the Waste Handling Building and a surface aging facility.

Crash Rate (C). The aircraft operating out of Nellis Air Force Base consist of more than 20 different types (DIRS 103472-USAF 1999, p. 1-35). However, the predominant aircraft types are F-16, F-15, and A-10 jets. These three types represent more than 75 percent of all aircraft operating out of Nellis, with the F-16 aircraft being the most prevalent, representing almost half (46 percent) of all aircraft operations (DIRS 103472-USAF 1999, pp. 1-35, 1-36). Estimates of the crash rates for these three aircraft are as follows (DIRS 108290-CRWMS M&O 1999, p. 18):

Aircraft	Crash rate/mile
F-16	3.86×10^{-8}
F-15	6.25×10^{-9}
A-10	3.14×10^{-8}

This analysis selected the F-16 crash rate to represent all aircraft operating out of Nellis Air Force Base. This selection was based on the fact that the F-16 aircraft, as noted, is the most numerous aircraft involved in Nellis operations, and it has the highest crash rate of the three most predominant aircraft and, therefore, results in a conservative evaluation. The rate is also somewhat conservative compared to a recent aircraft crash evaluation performed for the proposed Private Fuel Storage Facility in Utah (DIRS 154930-NRC 2000, Section 1.5.1.2.11). That analysis used an F-16 crash rate of 2.74×10^{-8} (DIRS 154930-NRC 2000, p. 193).

Effective Area of the Repository. According to the Nuclear Regulatory Commission (DIRS 152082-NRC 1981, p. 3.5.1.6-5), the effective area, A, to be used in the model should include the shadow area, the skid area, and the plant area. However, the equations for calculating these areas are not provided. Both DIRS 108290-CRWMS M&O (1999, p. 22) and DIRS 103687-Kimura, Sanzo, and Sharirli (1998, p. 9) use the formula recommended by DOE (DIRS 101810-DOE 1996, all). This formula is:

$$A = A_f + A_s$$

where

A_f is the effective fly-in area
A_s is the effective skid area.

Further,

$$A_f = \left[(Ws + R) \times H\cot\Phi\right] + \left[\frac{2L \times W \times Ws}{R}\right] + L \times W$$

$$A_s = (R+Ws)S$$

where

Ws	=	aircraft wingspan
H	=	facility height (feet)
$\cot\Phi$	=	mean of the cotangent of the aircraft impact angle
L	=	length of the facility (feet)
W	=	width of the facility (feet)
S	=	aircraft skid distance (feet)
R	=	length of the diagonal of the facility = $(L^2 + W^2)^{1/2}$.

The value of $\cot\Phi$ is 8.4 for in-flight crashes for small military aircraft (DIRS 101810-DOE 1996, p. B-29). The skid area is based on a skid distance (S). The analysis used a skid distance of 75 meters (246 feet) for small military aircraft under in-flight crash conditions based on mishap reports (DIRS 101810-DOE 1996, p. B-29). The wingspan recommended for high-performance jet fighters is 24 meters (78 feet) (DIRS 101810-DOE 1996, p. B-28). The remaining parameters (W, L, R, and H) are target (facility) specific.

- Waste Handling Building. The width of the Waste Handling Building would be about 116 meters (380 feet) (DIRS 152010-CRWMS M&O 2000, Figure 9, p. IV-11). This width includes all areas where spent nuclear fuel assemblies and high-level radioactive waste would be handled out of the storage pools. The spent nuclear fuel in the storage pools would not be vulnerable because it would be covered with 15 meters (50 feet) of water (DIRS 152010-CRWMS M&O 2000, Figure 13, p. IV-15). Even if the aircraft penetrated the walls around the pools, sank into the pool, and damaged the fuel, the release would be minimal because the pool water would retain most radionuclides (DIRS 150276-CRWMS M&O 2000, p. 20). Because the storage pool areas would be below grade, the aircraft could not enter the side of the pool and cause drainage in conjunction with spent nuclear fuel damage.

  The estimated length (L) of the facility vulnerable to aircrash impact would be 165 meters (542 feet) (DIRS 152010-CRWMS M&O 2000, Figure 9, p. IV-11).

  The length and width values include the disposal container transporter loading areas and handling cells for both the assembly and canister transfer systems. They also include the assembly dryer cells, the canister transfer cells, and the shipping cask preparation and transfer areas. The values for length and width are conservative because they encompass areas that are not vulnerable to radioactive release from air crashes, such as the heating, ventilation, and air conditioning areas, electrical equipment room, and hallways and corridors.

  The height of the facility would be 22 meters (73 feet) (DIRS 152010-CRWMS M&O 2000, Figure 13, p. IV-15). This would encompass the areas where radioactive material would be handled.

  The effective area, A, then becomes (in square feet):

$$A = [(Ws + R) \times H\cot\Phi] + \left[\frac{2L \times W \times Ws}{R}\right] + L \times W + (R + Ws)S$$

$$A = \{78 + [(542)^2 + (380)^2]^{1/2}\} \times (73)(8.40) + \{(2)542 \times 380 \times 78/[(542)^2 + (380)^2]^{1/2}\} + 542 \times 380 + [\{(542)^2 + (380)^2\}^{1/2} + Ws]S$$

$$= (78 + 662) \times 613 + 32{,}129{,}760/662 + 205{,}960 + (662 + 78)246$$

$$= 453{,}620 + 48{,}534 + 205{,}960 + 182{,}040 = 890{,}154 \text{ ft.}^2 = 0.032 \text{ mi.}^2$$

  Substituting the derived values into the aircraft crash probability equation yields the following for the annual probability of an aircraft crash on repository structures resulting in the release of radioactive material:

$$P_{FA} = C \times N \times A/w \times 0.145 = 3.86 \times 10^{-8} \times 1{,}773 \times 0.032/6 \times 0.145 = 5.2 \times 10^{-8}$$

  This probability is below once in 10 million (1×10^{-7}) per year, which is the probability level DOE has established (DIRS 104601-DOE 1993, p. 28) for consideration of accidents. Although the probability of this accident is outside the range normally presented in DOE EISs. DOE has chosen to present the potential consequences in Section H.2.1.5.1.

- Surface Aging Facility. Using an analysis consistent with the evaluation of the probability of a military aircraft crash into the Waste Handling Building, DOE evaluated the probability of a crash on the surface aging facility. The effective area of this facility, based on dimensions contained in DIRS 155043-CRWMS M&O (2001, all) was determined to be 0.49 square kilometer (0.19 square mile). Thus, the probability of an aircraft crash on the surface aging facility would be:

$$P_{FA} = C \times N \times A/w \times 0.145 = 3.86 \times 10^{-8} \times 1{,}773 \times 0.19/6 \times 0.145 = 3.14 \times 10^{-7}/\text{yr}.$$

The probability is slightly above the level that DOE has used in previous EISs. Section H.2.1.3.1 discusses the results of this analysis.

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 (DIRS 103813-DOE 1997, p. 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 (DIRS 152010-CRWMS M&O 2000, p. 30) 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 (DIRS 100204-CRWMS M&O 1996, p. 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 (DIRS 102702-CRWMS M&O 1997, p. 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 fire sources external to waste handling areas 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 (DIRS 104508-CRWMS M&O 1999, Section 4.2). 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 (DIRS 153849-DOE 2001, p. 2-69) would minimize the potential for fire-initiated radiological accidents.

7. **Flooding.** Flash floods could occur in the vicinity of the repository (DIRS 100204-CRWMS M&O 1996, p. 21). However, an earlier assessment (DIRS 103237-CRWMS M&O 1998, p. 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 (DIRS 104699-Jackson et al. 1984, p. 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 (DIRS 104699-Jackson et al. 1984, p. 53). A more recent study reached a similar conclusion (DIRS 101930-Ma et al. 1992, p. 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 (DIRS 153849-DOE 2001, Section 2.3.3.3). 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 (DIRS 104722-Wade 1998, all), so this facility poses no risk to the repository. The Kistler Aerospace activities would involve launching rockets from the Nevada Test Site to place satellites in orbit (DIRS 101811-DOE 1996, Volume 1, p. A-42). However, the Kistler Aerospace activity is currently on hold (DIRS 152582-Davis 2000, all), and there is insufficient information to assess if this activity would pose a threat to the repository. If the project moves forward, DOE will evaluate its potential to become an external accident-initiating event. (Aircraft activity is discussed in item 1 above.) No other industrial activities were found that could initiate accidents (DIRS 149759-CRWMS M&O 1999, all).

9. **Lightning.** This event has been identified as a potential design-basis event (DIRS 102702-CRWMS M&O 1997, pp. 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 (DIRS 102702-CRWMS M&0 1997, p. 86). The estimated frequency of such an event would be 1.9×10^{-7} per year (DIRS 103237-CRWMS M&O 1998, p. 33). DOE expects to provide lightning protection for the transporter (DIRS 100277-CRWMS M&O 1998, Volume 1, p. 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 (DIRS 104601-DOE 1993, p. 28).

10. **Loss of Offsite Power.** A preliminary evaluation (DIRS 102702-CRWMS M&O 1997, p. 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 loss of power to the ventilation system and of the overhead crane system. However, there would be emergency power for safety systems at the site (DIRS 104508-CRWMS M&O 1999, p. 45), and structures, systems, and components important to safety are designed to prevent load drops during loss of offsite power (DIRS 153849-DOE 2001, p. 5-12).

11. **Meteorite Impact.** The potential for a meteorite strike on the Waste Handling Building was examined and found to be an incredible event. This is based on the following analysis: Small meteorites dissipate their energy in the upper atmosphere and have no direct effect on the ground below. Only when the incoming projectile is larger than about 10 meters (33 feet) in diameter does it begin to pose some hazard to humans. A meteorite in the range of 10 meters in diameter strikes the Earth about once per decade, or a probability of 0.1 per year (DIRS 156370-NASA 2001, Section 2.2). Since the radius of the Earth is 6,383 kilometers (3,963 miles), the surface area of the Earth is 5.11×10^8 square kilometers (2.0×10^9 square miles). Thus, the probability of a hazardous meteorite strike on a specific square kilometer of area is $0.1/5.11 \times 10^8 = 1.96 \times 10^{-9}$ per year. Because the Waste Handling Building design footprint dimensions are (overall outside dimensions, ignoring included open spaces) 214 meters × 181 meters (704 feet × 593 feet) (DIRS 152010-CRWMS M&O 2000, Figure 13, p. IV-15), the target area would be 0.038 square kilometer (0.02 square mile). Therefore, the estimated probability of a hazardous meteor strike is $1.96 \times 10^{-9} \times 0.038 = 7.5 \times 10^{-11}$ per year, well below the credibility threshold of 1×10^{-7} per year (once in 10,000,000 per year) established by DOE (DIRS 104601-DOE 1993, p. 26). For the surface aging facility, the probability would also be below the credibility threshold. This is based on a facility area of 0.49 square kilometer (0.19 square mile) from Item 1 preceding. This area would result in an impact probability of $1.96 \times 0.49 = 9.6 \times 10^{-10}$.

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 (DIRS 103273-Walck 1996, p. 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. (DIRS 101930-Ma et al. 1992, p. 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 (DIRS 103237-CRWMS M&O 1998, p. 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) (DIRS 103237-CRWMS M&O 1998, p. VII-1). A recent comprehensive evaluation of the seismic hazards associated with the site of the proposed repository (DIRS 100354-USGS 1998, all) concluded that a 0.0001-per-year earthquake would produce peak horizontal accelerations at the site of about 0.53*g* (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 (DIRS 102182-Kennedy and Ravindra 1984, p. 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 1.1*g*. 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 designed to withstand an earthquake event with a return frequency of 1,000 years (annual exceedance probability of 1×10^{-3} per year) (DIRS 104508-CRWMS M&O 1999, p. 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 0.18*g*. 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} (three in 10 million) based on an assessment of tornado strike probability for any point on the Nevada Test Site (DIRS 101811-DOE 1996, p. 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 (DIRS 104601-DOE 1993, p. 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 (DIRS 103693-Ramsdell and Andrews 1986, p. 41). Thus, winds strong enough to damage the Waste Handling Building are considered to be not credible.

    Tornadoes can generate missiles that could affect structures at the repository, but radioactive material would be protected either by shipping casks, the Waste Handling Building with thick concrete walls, or the transporter. Structures, systems, and components that could be vulnerable to tornado missile impacts would either be protected from the missiles, designed to withstand a missile impact, or shown to not interact with a missile by a probabilistic analysis (DIRS 153849-DOE 2001, p. 5-15). Therefore, tornado-driven missiles would not be a credible hazard.

16. **Volcanism, Ash Fall.** The potential for volcanic activity at the proposed repository site has been studied extensively. A recent assessment (DIRS 151945-CRWMS M&O 2000, p. 12.2-4) estimates that the mean annual frequency of a volcanic event that would intersect the repository footprint would be 1.6×10^{-8} per year (with 5-percent and 95-percent bounds of 7.6×10^{-10} and 5×10^{-8} per year, respectively), which is below the frequency of a reasonably foreseeable event for evaluation as an accident. Igneous activity scenarios are, however, evaluated as part of long-term performance (Chapter 5, Section 5.7.2). This result is consistent with a previous study of volcanic activity at the site (DIRS 101779-DOE 1998, 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, if thick enough, 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 depth of 3 centimeters (1.2 inches), and analyses to date indicate that repository structures would not be affected by a 3-centimeter ash fall (DIRS 101779-DOE 1998, Volume 2, pp. 2-9). Furthermore, the extreme consequence of excessive ashfall on the Waste Handling Building would be collapse of the building from excessive weight. Therefore, this event is bounded by the seismic event that caused collapse. The potential of a volcanic event affecting postclosure repository performace is discussed in Chapter 5, Section 5.7.2.

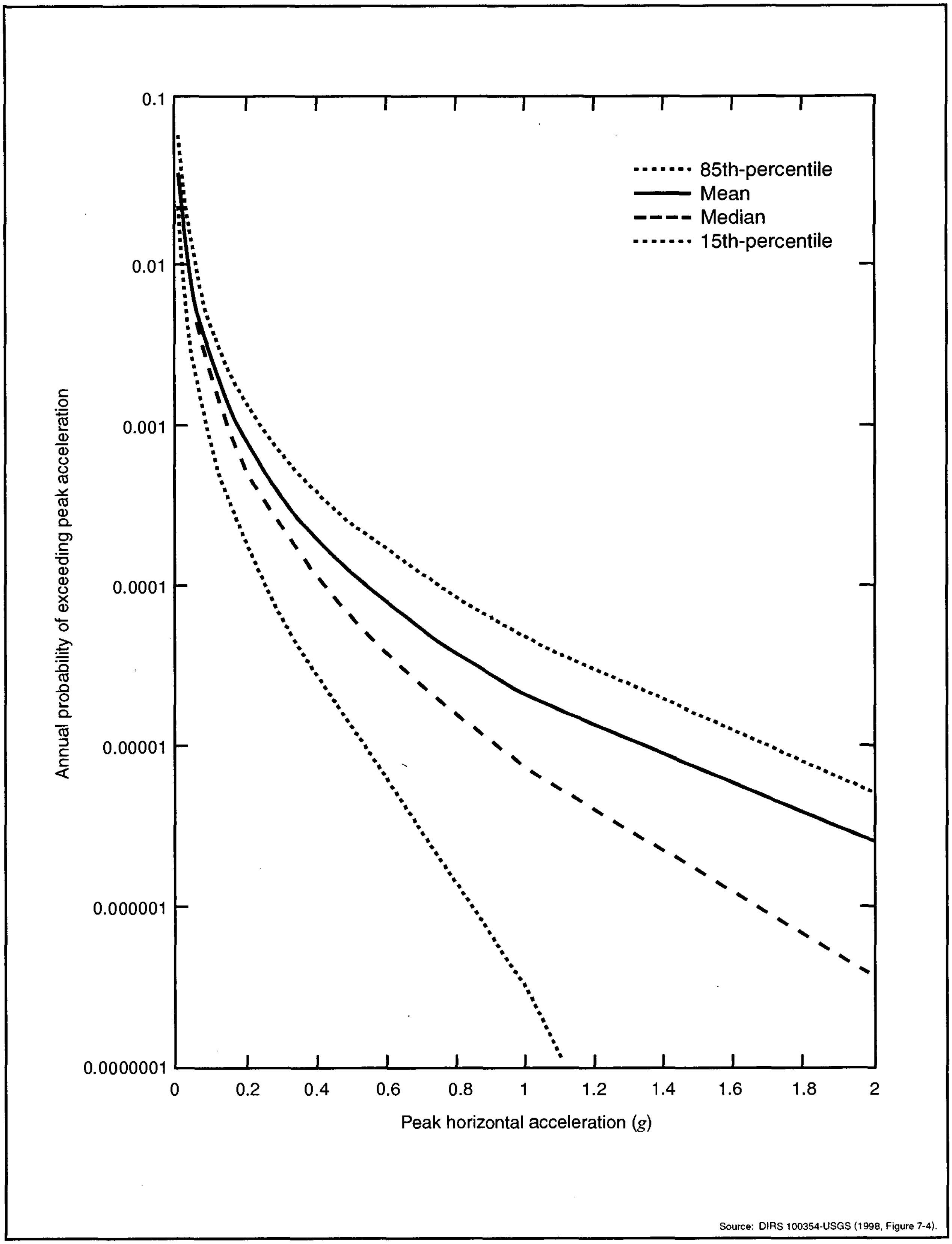

Figure H-1. Integrated seismic hazard results: summary hazard curves for peak horizontal acceleration.

17. Sabotage. In the aftermath of the tragic events of September 11, DOE is continuing to assess measures that it could take to minimize the risk or potential consequences of radiological sabotage or terrorist attacks against our Nation's proposed monitored geologic repository.

Over the long term (after closure), deep geologic disposal of spent nuclear fuel and high-level radioactive waste would provide optimal security by emplacing the material in a geologic formation that would provide protection from inadvertent and advertent human intrusion, including potential terrorist activities. The use of robust metal waste packages to contain the spent nuclear fuel and high-level radioactive waste more than 200 meters (660 feet) below the surface would offer significant impediments to any attempt to retrieve or otherwise disturb the emplaced materials.

In the short term (prior to closure), the proposed repository at Yucca Mountain would offer certain unique features from a safeguards perspective: a remote location, restricted access afforded by Federal land ownership and proximity to the Nevada Test Site, restricted airspace above the site, and access to a highly effective rapid-response security force.

Current Nuclear Regulatory Commission regulations (10 CFR 63.21 and 10 CFR 73.51) specify a repository performance objective that provides "high assurance that activities involving spent nuclear fuel and high-level waste do not constitute an unreasonable risk to public health and safety." The regulations require that spent nuclear fuel and high-level radioactive waste be stored in a protected area such that:

- Access to the material requires passage through or penetration of two physical barriers. The outer barrier must have isolation zones on each side to facilitate observation and threat assessment, be continually monitored, and be protected by an active alarm system.
- Adequate illumination must be provided for observation and threat assessment.
- The area must be monitored by random patrol.
- Access must be controlled by a lock system, and personnel identification must be used to limit access to authorized persons.

A trained, equipped, and qualified security force is required to conduct surveillance, assessment, access control, and communications to ensure adequate response to any security threat. Liaison with a response force is required to permit timely response to unauthorized entry or activities. In addition, the Nuclear Regulatory Commission requires (10 CFR Part 63, by reference to 10 CFR Part 72) that comprehensive receipt, periodic inventory, and disposal records be kept for spent nuclear fuel and high-level radioactive waste in storage. A duplicate set of these records must be kept at a separate location.

DOE believes that the safeguards applied to the proposed repository should involve a dynamic process of enhancement to meet threats, which could change over time. Repository planning activities would continue to identify safeguards and security measures that would further protect fixed facilities from terrorist attack and other forms of sabotage. Additional measures that DOE could adopt include:

- Facilities with thicker reinforced walls and roofs designed to mitigate the potential consequences of the impact of airborne objects

- Underground or surface bermed structures to lessen the severity of damage in cases of aircraft crashes
- Additional doors, airlocks, and other features to delay unauthorized intrusion
- Additional site perimeter barriers to provide enhanced physical protection of site facilities
- Active denial systems to disable any adversaries, thereby preventing access to the facility

Although it is not possible to predict if sabotage events would occur, and the nature of such events if they did occur, DOE examined various accident scenarios in this Appendix that approximate the types of consequences that could occur.

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 (DIRS 101930-Ma et al. 1992, p. 3-11; DIRS 104699-Jackson et al. 1984, pp. 12 and 13). As mentioned in the discussion of an accidental aircraft crash, DOE has chosen to evaluate the consequences of such an event even though the estimated frequency is below the threshold for credible events. This analysis is included in Section H.2.1.5.1.

H.2.1.3.1 *Surface Aging Facility*

As indicated previously, DOE is considering a surface aging facility as an option to enable aging of commercial spent nuclear fuel prior to emplacement. The aging process would reduce the heat generation rate from the spent nuclear fuel, which could be used to control subsurface temperatures. The design of the surface aging facility is described in detail in DIRS 155043-CRWMS M&O (2001, all). The storage facility could include up to 40,000 metric tons of heavy metal of spent nuclear fuel in individual storage modules on concrete storage pads. Spent nuclear fuel to be aged would be loaded in an overpack cask in the Waste Handling Building and moved to the surface aging facility and placed in a shielded storage cask. For this analysis, DOE assumed that the components used in the storage modules would be the same as those proposed for the Private Fuel Storage facility in Utah (DIRS 154930-NRC 2000, all), which is designed for the interim storage of commercial spent nuclear fuel. That facility has design characteristics and operation parameters similiar to those DOE would use for the surface aging facility at the proposed repository. The surface aging facility design would conform to the same safety requirements as that for the Private Fuel Storage facility.

In evaluating potential accidents at the surface aging facility, DOE assumed that the results of the Private Fuel Storage facility safety analysis would generally apply (DIRS 154930-NRC 2000, all). On the basis of that safety analysis and site-specific characteristics of the proposed repository, DOE determined that only two accidents, both external events, would have the potential to release radioactivity to the environment. These accidents are a beyond-design-basis earthquake event and an aircraft crash into storage modules.

The surface aging facility would be designed to withstand the design-basis earthquake without tipover of the storage modules, in compliance with Nuclear Regulatory Commission requirements. A beyond-design-basis earthquake, however, could be a credible event, so DOE evaluated it. The most significant consequences of a beyond-design-basis earthquake would be tipover of the storage modules containing the overpack cask and storage canister. Such an event would not result in a release because tipover of the storage overpack cask would not impair the ability of the cask to maintain confinement of the stored fuel (DIRS 154930-NRC 2000, p.165).

For the aircraft crash event, DOE determined, as evaluated in Section H.2.1.3, that a crash involving a military aircraft from Nellis Air Force Base could be a reasonably forseeable event if the entire storage capacity was being used. As a consequence, an analysis of the penetration capability of a crashing aircraft determined that the limiting aircraft missiles from Nellis Air Force Base aircraft would not penetrate the storage modules (DIRS 157108-Jason 2001, all). This result was based on analysis of the Private Fuel Storage module design (DIRS 154930-NRC 2000, all), which includes an inner storage canister, an overpack cask with thick steel walls, and a shielded outer cask consisting of steel shells enclosing a concrete annulus 70 centimeters (28 inches) thick. Other designs that DOE could select for the surface aging facility would have similar characteristics to meet applicable requirements.

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 is an estimate of the amount of material released, which is used in estimating radiological impacts from accidents. 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, seismic event, and low-level waste drum failure. The source term for the accidental aircraft crash into the repository surface facilities is described in Section H.2.1.5.1.

For accident releases that would be filtered through high-efficiency particulate air filters by the heating, ventilation, and air conditioning system, the analysis assumed a retention factor of 0.99 for particulates, consistent with DIRS 150276-CRWMS M&O (2000, p. 21).

H.2.1.4.1 *Commercial Spent Nuclear Fuel Drop Accident Scenario Source Term*

Commercial spent nuclear fuel contains nearly 400 radionuclides (DIRS 100181-SNL 1987, Appendix A). Not all of these radionuclides, 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 a radionuclide (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 radionuclides 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" (DIRS 101903-NRC 1997, p. 7-6). Repository accident scenario evaluations (DIRS 100181-SNL 1987, pp. 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 accident consequence evaluation (DIRS 150276-CRWMS M&O 2000, Attachment VIII) used a total of 51 radionuclides that included all of those discussed above. DOE used this same list for the EIS accident impact evaluations (see Appendix A, Section A.2.1.5.2).

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 "representative" fuels for each type. These fuels were defined on the basis of a relative hazard evaluation (see Appendix A, Section A.2.1.5). Table H-3 lists the characteristics of representative commercial spent nuclear fuel types. Table H-4 lists the radionuclide inventory selected for estimating the accident scenario consequences for the fuel types selected (representative boiling-water reactor and pressurized-water reactor).

Table H-3. Representative commercial spent nuclear fuel characteristics.[a]

Fuel type[b]	Cooling time (years)	Burnup (GWd/MTHM)[c]
PWR representative	15	50
BWR representative	14	40

a. Source: Appendix A, Section A.2.1.5.
b. PWR = pressurized-water reactor; BWR = boiling-water reactor.
c. GWd/MTHM = gigawatt-days per metric ton of heavy metal.

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 assembly surfaces and becomes radioactive from neutron activation. Appendix A, Section A.2.1.5.2, describes the inventory of this material, which amounts to a total of 9 curies per assembly of cobalt-60 for representative pressurized-water reactor fuel and 16 curies per assembly for representative boiling-water reactor fuel.

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 (DIRS 103696-Sandoval et al. 1991, p. 18). A recent comprehensive assessment (DIRS 152476-Sprung et al. 2000, Section 7) of crud release potential under accident conditions involving commercial spent nuclear fuel estimated that 10 percent of the crud would flake off during events involving mechanical impacts to the fuel assemblies (DIRS 152476-Sprung et al. 2000, p. 7-49). DOE used this value for repository accident analyses for events involving mechanical impact to the assemblies.

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 (DIRS 100181-SNL 1987, p. 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

Table H-4. Inventory used for representative commercial spent nuclear fuel (curies per assembly).[a,b,c]

Isotope	Location	Pressurized-water reactor	Boiling-water reactor
Hydrogen-3	Fuel clad gap	2.0×10^2	66
Carbon-14	Fuel clad gap	0.31	0.16
Chlorine-36	Fuel clad gap	6.3×10^{-3}	2.6×10^{-3}
Iron-55	Nonfuel structures	40	16
Cobalt-60	Nonfuel structures	1.1×10^{-3}	1.7×10^2
Cobalt-60	Assembly surface (crud)	8.8	16
Nickel-59	Nonfuel structures	1.9	0.45
Nickel-63	Nonfuel structures	2.5×10^2	57
Selenium-79	Fuel pellet	4.6×10^{-2}	1.4×10^{-2}
Krypton-85	Fuel clad gap	2.2×10^3	7.0×10^2
Strontium-90	Fuel pellet, gap	3.6×10^4	1.1×10^4
Yttrium-90	Fuel pellet, gap	3.6×10^4	1.1×10^4
Zirconium-93	Fuel pellet	0.98	0.3
Niobium-93m	Fuel pellet	19	0.5
Niobium-94	Fuel pellet	0.81	1.7×10^{-2}
Technetium-99	Fuel pellet	9.1	2.9
Ruthenium-106	Fuel pellet	11	4.9
Palladium-107	Fuel pellet	7.8×10^{-2}	2.4×10^{-2}
Cadmium-113m	Fuel pellet	12	3.5
Antimony-125	Fuel pellet	1.2×10^2	43
Tin-126	Fuel pellet	0.37	0.11
Iodine-129	Fuel clad gap	2.2×10^{-2}	6.7×10^{-3}
Cesium-134	Fuel pellet, gap	7.2×10^2	2.3×10^2
Cesium-135	Fuel pellet, gap	0.38	0.13
Cesium-137	Fuel pellet, gap	5.2×10^4	1.6×10^4
Barium-137m	Fuel pellet, gap	5.2×10^4	1.6×10^4
Promethium-147	Fuel pellet	1.7×10^3	6.6×10^2
Samarium-151	Fuel pellet	2.4×10^2	53
Europium-154	Fuel pellet	1.5×10^3	3.9×10^2
Europium-155	Fuel pellet	2.2×10^2	75
Actinium-227	Fuel pellet	1.3×10^{-5}	0
Thorium-230	Fuel pellet	9.9×10^{-5}	3.3×10^{-5}
Protactinium-231	Fuel pellet	3.3×10^{-5}	1.2×10^{-5}
Uranium-232	Fuel pellet	2.4×10^{-2}	4.6×10^{-3}
Uranium-233	Fuel pellet	3.2×10^{-5}	0
Uranium-234	Fuel pellet	6.7×10^{-1}	0.21
Uranium-235	Fuel pellet	8.8×10^{-3}	2.4×10^{-3}
Uranium-236	Fuel pellet	0.19	5.6×10^{-2}
Uranium-238	Fuel pellet	0.14	5.7×10^{-2}
Neptunium-237	Fuel pellet	0.25	6.0×10^{-2}
Plutonium-238	Fuel pellet	2.6×10^3	5.7×10^2
Plutonium-239	Fuel pellet	1.8×10^2	48
Plutonium-240	Fuel pellet	3.1×10^2	1.0×10^3
Plutonium-241	Fuel pellet	3.9×10^4	1.0×10^{-4}
Plutonium-242	Fuel pellet	1.5	0.46
Americium-241	Fuel pellet	1.5×10^3	3.7×10^2
Americium-242m	Fuel pellet	7.2	2.1
Americium-243	Fuel pellet	20	4.8
Curium-242	Fuel pellet	5.9	1.7
Curium-243	Fuel pellet	13	2.9
Curium-244	Fuel pellet	1.8×10^3	3.5×10^2
Curium-245	Fuel pellet	0.29	3.6×10^2
Curium-246	Fuel pellet	9.1×10^{-2}	1.3×10^{-2}

a. Source: Appendix A.
b. Inventory numbers have been rounded to two significant figures.
c. The analysis included yttrium-90 and barium-137m and assumed them to be in equilibrium with strontium-90 and cesium-137, respectively.

assumed that the respirable fraction would be 0.05 (based on DIRS 104724-Wilmot 1981, p. B-3). Therefore, the analysis assumed that the total cobalt-60 respirable airborne release fraction would be 0.0005 (the flake off fraction of 0.1 multiplied by the amount not deposited on fuel assembly surfaces of 0.1 multiplied by the respirable fraction of 0.05) for accident scenarios involving commercial spent nuclear fuel.

H.2.1.4.1.2 Fuel Rod Gap. The space between the fuel rod cladding and the fuel pellets (called the *gap*) contains radionuclides 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 (DIRS 101903-NRC 1997, p. 7-6). In addition, the analysis considered carbon-14, which it assumed to reside in the gaps as a gas. 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 (DIRS 101903-NRC 1997, p. 7-6). The carbon-14 release fraction was assumed to be the same as the radioactive gases in the gap, tritium, and krypton-85 (release fraction of 0.3). Chlorine-36 was assumed to be combined with cesium and, therefore, would have the same release fraction (2.3×10^{-5}). These assumptions are consistent with releases assumed for transportation accidents (see Appendix J, Section J.1.4.2). 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. DIRS 100181-SNL (1987, p. 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 (DIRS 103237-CRWMS M&O 1998, p. 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 (DIRS 100181-SNL 1987, p. 5-27). DIRS 100181-SNL (1987, p. 5-28) also applies a similar factor to account for retention on the failed shipping cask structures for accident scenarios involving cask failure. 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 existing pellet particulates is equivalent to the same fractional release of the total inventory of the appropriate radionuclides in the fuel pellets. 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 (DIRS 100181-SNL 1987, p. 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 (DIRS 100181-SNL 1987, p. 5-17) and applied in (DIRS 103237-CRWMS M&O 1998, p. 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:

> 2.0×10^{-7} × energy partition factor × unimpeded drop height (centimeters) (DIRS 103237-CRWMS M&O 1998, p. 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 (DIRS 100181-SNL 1987, p. 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 (DIRS 100181-SNL 1987, p. 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 (DIRS 103695-SAIC 1998, p. 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	0.1	0.1	0.05	0.0005
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, Cl-36	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 particulates)	0.2	1.0×10^{-6}	0.1	1.0	2.0×10^{-8}
All solids	Pellet-pulverization	0.2	$4.0\times10^{-8}\times h$[b]	0.1	1.0	$8.0\times10^{-10}\times h$[b]

a. Abbreviations: Co = cobalt; H = hydrogen (H-3 = tritium); Kr = krypton; C = carbon; I = iodine; Cs = cesium; Sr = strontium; Cl = chlorine; Ru = ruthenium.

b. h = drop height in centimeters; depends on specific accident scenarios.

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) (DIRS 102702-CRWMS M&O 1997, p. 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:

$$\text{velocity} = \text{acceleration} \times \text{time}$$

and,

$$\text{distance} = \tfrac{1}{2} \times \text{acceleration} \times \text{time squared}$$

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,

$$\begin{aligned} \text{distance} &= \tfrac{1}{2} \times \text{acceleration} \times (\text{velocity} \div \text{acceleration})^2 \\ &= (\text{velocity})^2 \div (\text{acceleration} \times 2) \\ &= (18)^2 \div (9.8 \times 2) \\ &= 16 \text{ meters} \end{aligned}$$

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*

Because the analysis identified no repository accidents that could result in releases of radionuclides from DOE spent nuclear fuel that exceeded limits established by the Nuclear Regulatory Commission, the source term for such accidents is not important in this environmental impact analysis. Furthermore, as indicated in the Draft EIS (Appendix H, p. H-7), the maximum consequences for credible accidents involving bounding DOE spent nuclear fuel would be much less than equivalent accidents involving commercial spent nuclear fuel.

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.

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 (DIRS 104508-CRWMS M&O 1999, Attachment IV, Figure 13). Typical pressurized-water reactor fuel assemblies weigh 660 kilograms (1,500 pounds) each (see Appendix A, Section A.2.1.5.5). 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) (DIRS 100181-SNL 1987, p. 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) (DIRS 103178-Lide and Frederikse 1997, p. 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 (DIRS 100181-SNL 1987, p. 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 (DIRS 100181-SNL 1987, p. 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 (DIRS 100181-SNL 1987, p. 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:

$$\begin{aligned}\text{Respirable airborne release fraction} &= 8 \times 10^{-10} \times \text{drop height (centimeters)} \times \text{rubble release factor}\\ &= 8 \times 10^{-10} \times 1{,}500 \times 0.1\\ &= 1.2 \times 10^{-7}\end{aligned}$$

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 294 bare pressurized-water reactor fuel assemblies could be exposed to falling debris (DIRS 152579-Montague 2000, p. 1). The location of this material would be as follows:

- Assembly transfer system dryers: 84 pressurized-water reactor assemblies
- Disposal canister handling system welding stations: 168 pressurized-water reactor assemblies
- Assembly transfer system load port: 42 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 294 representative 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, Section A.2.2.1) to be received at the repository.

Bare spent nuclear fuel assemblies stored in the blending inventory pools or the assembly holding pool could be vulnerable to damage from the postulated earthquake. However, the Waste Handling Building enclosure over the pool areas would be a steel frame structure that would not have a thick concrete slab roof. Therefore, there would be no heavy concrete blocks to fall into the pools and cause extensive damage to the stored fuel assemblies. The 15-meter (50-foot) depth of the pools would also limit the velocity of impact (and therefore impact damage) of any debris that might enter the pool from the postulated earthquake. Further, if a radionuclide release were to occur from damage to spent fuel assemblies in a pool, the release would be very small because the radionuclides contained in the fuel pellet particles would be retained in the pool water, and releases would therefore be minimal (DIRS 147496-CRWMS M&O 2000, p. 51). Because the pools would be below ground level, would be constructed of reinforced concrete, and would have steel liners, rapid draining of the pools would not be expected from earthquake damage.

Waste Treatment Building. It is assumed that the radionuclide concentration for the dry compactible waste in the Waste Treatment Building would be similar to that for power reactors (DIRS 104701-McFeely 1998, p. 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) (DIRS 104701-McFeely 1998, p. 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 (DIRS 104702-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 (DIRS 104508-CRWMS M&O 1999, 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 (DIRS 100217-CRWMS M&O 1997, p. 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:

$$\text{Cobalt-60} = 4.0 \text{ cubic meters} \times 0.025 \text{ curie per cubic meters} \times 0.67 \text{ (cobalt-60 fraction)} \cong 0.07 \text{ curie}$$

$$\text{Cesium-137} = 4.0 \text{ cubic meters} \times 0.025 \text{ curie per cubic meters} \times 0.28 \text{ (cesium-137 fractions)} \cong 0.03 \text{ curie}$$

The respirable airborne release fraction for a fire involving combustible low-level waste has been conservatively estimated at 0.4 (DIRS 103688-Mueller et al. 1996, p. D-21). Thus, the respirable airborne release source term for the fire accident scenario would be:

$$\text{Cobalt-60} = 0.07 \text{ curie} \times 0.4 = 0.028 \text{ curie}$$
$$\text{Cesium-137} = 0.03 \text{ curie} \times 0.4 = 0.012 \text{ curie}$$

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 (DIRS 104701-McFeely 1998, p. 3):

Cobalt-60 = 0.001 curie per cubic meter
Cesium-137 = 0.0015 curie per cubic meter

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:

Cobalt-60 = 0.01 curie per cubic meter
Cesium-137 = 0.015 curie per cubic meter

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:

Cobalt-60 = 0.01 curie per cubic meter × 0.21 cubic meter
≅ 0.0021 curie per drum

Cesium-137 = 0.015 curie per cubic meter × 0.21 cubic meter
≅ 0.0032 curie per drum

For dry compacted waste, the total inventory in a 0.21-cubic-meter (55-gallon) drum would be

Cobalt-60 = 0.21 cubic meter × 0.025 curie per cubic meter × 0.67 (cobalt-60 fraction)
≅ 0.0035 curie

Cesium-137 = 0.21 cubic meter × 0.025 curie per cubic meter × 0.28 (cesium-137 fraction)
≅ 0.0015 curie

The estimated amount of material released from drums containing solid waste is 25 percent of the contents based on (DIRS 103688-Mueller et al. 1996, p. 94). Values from (DIRS 103688-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:

A	=	constant (2.0×10^{-11}) (DIRS 103688-Mueller et al. 1996, p. D-25)
D	=	material density [3.14 grams per cubic centimeter (196 pounds per cubic foot)] (DIRS 104701-McFeely 1998, all)
G	=	gravitational acceleration [980 centimeters (32.2 feet) per second squared]
H	=	height of fall of the drum in the accident scenario

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 (DIRS 103688-Mueller et al. 1996, p. 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:

$$\text{Respirable airborne release fraction} = 2.0 \times 10^{-11} \times 3.14 \times 980 \times 200 \cong 1.23 \times 10^{-5}$$

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:

Cobalt-60 = 0.0035 curie (total drum inventory) × 0.25 (fraction released) × 0.001 (respirable airborne release fraction)
≅ 8.5×10^{-7} curies

Cesium-137 = 0.0015 curie (total drum inventory) × 0.25 (fraction released) × 0.001 (respirable airborne release fraction)
≅ 3.8×10^{-7} curies

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 (DIRS 101074-ICRP 1975, p. 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 individuals 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 8 kilometers (5 miles) (DIRS 150276-CRWMS M&O 2000, p. 14)], (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.

If the total radiation dose is less than 20 rem, or the dose rate is less than 10 rem per hour, potential health effects would be chronic rather than acute. Chronic health effects could result in an increase in the risk of fatal cancer (DIRS 101836-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 (DIRS 101836-ICRP 1991, p. 15). At even higher acute doses (above about 500 rem), death within a few weeks is probable (DIRS 101836-ICRP 1991, p. 16).

DOE used the MACCS2 computer program (DIRS 101897-Jow et al. 1990, all; DIRS 103168-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 individuals and populations. 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 (DIRS 101069-Eckerman, Wollbarst, and Richardson 1988, p. 7). For external exposure (from ground contamination and contaminated food consumption), the dose was assumed to accumulate for 30 years (DIRS 104601-DOE 1993, p. 21).

The MACCS2 program provides doses to selected individuals and populations 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, 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 individuals 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 0.00033 cubic meter (0.011 cubic foot) per second (DIRS 101074-ICRP 1975, p. 346).

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 2035 (see Appendix G, Table G-48). 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 (DIRS 150276-CRWMS M&O 2000, Attachment IV, pp. IV-1 through IV-20). 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-6 (for 50-percent weather) and H-7 (for 95-percent weather). These tables include the accidents retained for analysis based on the internal events evaluation described in Section H.2.1.1, the earthquake events resulting from the external events analysis in Section H.2.1.3, and an accident involving low-level waste in the Waste Treatment Building based on the evaluation in Section H.2.1.2. The tables list doses in rem for individuals and in person-rem (collective dose to all exposed persons) for the 80-kilometer (50-mile) population around the site. For selected individuals and populations, as noted, the tables list estimated latent cancer fatalities predicted to occur over the lifetime of the exposed individuals 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 1998 the likelihood of fatal cancer from all causes for Nevada residents was about 0.24 (DIRS 153066-Murphy 2000, p. 83). Thus, the estimated latent cancer fatalities for the individuals from accidents would be very small in comparison to the cancer incidence from other causes. For the 76,000 persons expected to be living within 80 kilometers of the site in 2035 (see Appendix G), 18,240 (76,000 × 0.24) would be likely to die eventually of cancer not related to the repository. The accident of most concern for the 95-percent weather conditions (earthquake, Table H-7, number 8) would result in an estimated 0.011 latent cancer fatality for this same population. The results illustrate, by comparison of accidents 6 and 7, that accidents involving pressurized-water reactor fuel assemblies in the Waste Handling Building would produce larger impacts than equivalent accidents involving boiling-water reactor fuel assemblies.

DOE has not evaluated in detail the potential cleanup costs associated with accidents involving releases of radioactive material at the proposed repository. However, cleanup costs for transportation accidents involving material to be transported to the repository are considered in Appendix J, Section J.1.4.2.5. Such costs are highly uncertain, and depend on the type of land involved, the type of remediation action employed, and the extent of cleanup based on requirements that could exist at the time of the accident. As noted in Section J.1.4.2.5, the costs could range from about $1 million to $10 billion for severe, maximum reasonably foreseeable transportation accidents. For the repository accidents evaluated in this Appendix, DOE expects costs to be below the lower end of this range because the releases would be very small and the land near the repository would be Federally controlled, undeveloped, and uninhabited. In any event, liability for, and recovery of, costs of such accidents would be covered under provisions of the Price-Anderson Act, which currently provides for costs as high as $9.43 billion.

H.2.1.5.1 *Assessment of Consequences from Hypothetical Aircraft Crash Event*

In response to public comments and to provide further information about accident risks, DOE analyzed an accident scenario in which a large, commercial jet aircraft impacts and penetrates the Waste Handling Building, resulting in a fire. The probability of this accident is below the threshold considered reasonably foreseeable (1 in 10 million); however, if the accident occurred, the estimated consequences would include a dose of 4.5 rem to the maximally exposed offsite individual and a corresponding likelihood of 0.0023 that this individual would incur a fatal cancer as a result of the exposure. The consequences to the population for this event would be 78 person-rem and an estimated 0.039 latent cancer fatalities.

Table H-6. Radiological consequences of repository operations accidents for median (50th-percentile) meteorological conditions.

		Maximally exposed offsite individual[d]		Population		Noninvolved worker		Involved worker	
Accident scenario[a,b,c]	Frequency (per year)[a]	Dose (rem)	LCFi[e]	Dose (person-rem)	LCFp[e]	Dose (rem)	LCFi	Dose (rem)	LCFi
1. Basket drop onto another basket in pool (PWR fuel)	0.04	8.2×10^{-7}	4.1×10^{-10}	4.9×10^{-4}	2.4×10^{-7}	3.6×10^{-4}	1.4×10^{-7}	(f)	(f)
2. 5-meter basket drop onto another basket in dryer (PWR fuel)	0.04	8.7×10^{-6}	4.4×10^{-9}	8.9×10^{-4}	4.4×10^{-7}	4.5×10^{-3}	1.8×10^{-6}	(f)	(f)
3. 7.6-meter drop of transfer basket onto another basket in dryer (BWR fuel)	7.4×10^{-3}	6.4×10^{-6}	3.2×10^{-9}	6.0×10^{-4}	3.0×10^{-7}	3.1×10^{-5}	1.2×10^{-8}	(f)	(f)
4. 6-meter unsealed DC drop and slapdown in cell (PWR fuel)	8.0×10^{-3}	2.6×10^{-5}	1.3×10^{-8}	2.5×10^{-3}	1.2×10^{-6}	1.3×10^{-2}	5.2×10^{-6}	(f)	(f)
5. 7.1-meter unsealed shipping cask drop in CPP (PWR fuel)	9.0×10^{-3}	3.4×10^{-5}	1.8×10^{-8}	3.0×10^{-3}	1.5×10^{-6}	1.8×10^{-2}	7.4×10^{-6}	(f)	(f)
6. Unsealed shipping cask drop in pool (PWR fuel)	9.0×10^{-3}	2.5×10^{-6}	1.3×10^{-9}	1.5×10^{-3}	7.3×10^{-7}	1.0×10^{-3}	4.1×10^{-7}	(f)	(f)
7. Transporter runaway and derailment (PWR fuel)	1.2×10^{-7}	1.0×10^{-2}	5.0×10^{-6}	0.14	7.3×10^{-5}	3.2	1.3×10^{-3}	(g)	(g)
8. Beyond design basis earthquake in WHB (PWR fuel)	2.0×10^{-5}	1.2×10^{-2}	6.0×10^{-6}	0.63	3.2×10^{-4}	4.9	2.0×10^{-3}	(g)	(g)
9. Earthquake with fire in WTB	2.0×10^{-5}	1.6×10^{-5}	8.0×10^{-9}	8.9×10^{-4}	4.4×10^{-7}	8.2×10^{-4}	3.3×10^{-7}	(g)	(g)
10. Low level waste drum rupture in WTB	0.59	5.7×10^{-10}	2.9×10^{-13}	3.0×10^{-8}	1.4×10^{-11}	2.5×10^{-8}	1.0×10^{-11}	8.8×10^{-5}	3.5×10^{-8}

a. 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 (DIRS 150276-CRWMS M&O 2000, all).

b. DC = Disposal Container, CPP = Cask Preparation Pit, PWR = Pressurized Water Reactor, BWR = Boiling Water Reactor, WHB = Waste Handling Building, WTB = Waste Treatment Building.

c. To convert meters to feet, multiply by 3.2808.

d. Assumed to be at the nearest land withdrawal boundary, which would be 11 kilometers (7 miles) for all accidents except 7. For these accidents, the distance would be 8 kilometers (5 miles).

e. LCFi is the estimated likelihood of a latent cancer fatality for an individual who receives the calculated dose. LCFp is the number of cancers estimated 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.

f. 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.

g. 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 WHB, and as many as 36 in the WTB based on staffing projections (DIRS 104718-CRWMS M&O 1998, pp. 17 and 18).

The following locations were considered in the analysis:

1. Transportation casks staged at the repository
2. Waste Handling Building at the repository
3. Waste packages, either in transit at the repository, or in subsurface emplacement drifts
4. Repository surface aging facility storage modules

Table H-7. 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[d] Dose (rem)	LCFi[e]	Population Dose (person-rem)	LCFp[e]	Noninvolved worker Dose (rem)	LCFi	Involved worker Dose (rem)	LCFi
1. Basket drop onto another basket in pool (PWR fuel)	0.04	3.3×10^{-6}	1.7×10^{-9}	4.0×10^{-2}	2.0×10^{-5}	2.0×10^{-3}	8.0×10^{-7}	(f)	(f)
2. 5-meter basket drop onto another basket in dryer (PWR fuel)	0.04	3.2×10^{-5}	1.6×10^{-8}	4.7×10^{-2}	2.3×10^{-5}	2.3×10^{-2}	9.2×10^{-6}	(f)	(f)
3. 7.6 meter drop of transfer basket onto another basket in dryer (BWR fuel)	7.4×10^{-3}	2.3×10^{-5}	1.2×10^{-8}	3.0×10^{-2}	1.4×10^{-5}	1.6×10^{-4}	6.4×10^{-8}	(f)	(f)
4. 6-meter unsealed DC drop and slapdown in cell (PWR fuel)	8.0×10^{-3}	9.3×10^{-5}	4.7×10^{-8}	0.12	6.2×10^{-5}	7.4×10^{-2}	3.0×10^{-5}	(f)	(f)
5. 7.1-meter unsealed shipping cask drop in CPP (PWR fuel)	9.0×10^{-3}	1.1×10^{-4}	5.5×10^{-8}	0.14	7.2×10^{-5}	0.10	4.1×10^{-5}	(f)	(f)
6. Unsealed shipping cask drop in pool (PWR fuel)	9.0×10^{-3}	1.0×10^{-5}	5.0×10^{-9}	0.12	6.0×10^{-5}	6.0×10^{-3}	2.4×10^{-6}	(f)	(f)
7. Transporter runaway and derailment (PWR fuel)	1.2×10^{-7}	3.8×10^{-2}	1.9×10^{-5}	4.3	2.2×10^{-3}	16	6.4×10^{-3}	(f)	(f)
8. Beyond design basis earthquake in WHB (PWR fuel)	2.0×10^{-5}	3.8×10^{-2}	1.9×10^{-5}	21	1.1×10^{-2}	25	9.8×10^{-3}	(g)	(g)
9. Earthquake with fire in WTB	2.0×10^{-5}	5.4×10^{-5}	2.7×10^{-8}	3.1×10^{-2}	1.5×10^{-5}	6.5×10^{-3}	2.6×10^{-6}	(g)	(g)
10. Low level waste drum rupture in WTB	0.59	1.6×10^{-9}	8.0×10^{-13}	1.1×10^{-6}	5.3×10^{-10}	2.0×10^{-7}	8.0×10^{-11}	8.8×10^{-5}	3.5×10^{-8}

a. 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 (DIRS 150276-CRWMS M&O 2000, all).

b. DC = Disposal Container; CPP = Cask Preparation Pit; PWR = Pressurized-Water Reactor; BWR = Boiling-Water Reactor; WHB = Waste Handling Building; WTB = Waste Treatment Building.

c. To convert meters to feet, multiply by 3.2808.

d. Assumed to be at the nearest land withdrawal boundary, which would be 11 kilometers (7 miles) for all accidents except 7. For these accidents, the distance would be 8 kilometers (5 miles).

e. LCFi is the estimated likelihood of a latent cancer fatality for an individual who receives the calculated dose. LCFp is the number of cancers estimated 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.

f. 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.

g. 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 WHB, and as many as 36 in the WTB based on staffing projections (DIRS 104718-CRWMS M&O 1998, pp. 17 and 18).

DOE determined that an aircraft crash into the Waste Handling Building would bound the impacts from the list of locations considered. This is because the Waste Handling Building would be expected to contain the largest amount of vulnerable radioactive waste and could also be penetrated by an aircraft. The amount of waste that would be contained in the Waste Handling Building during normal operations for this assessment is assumed to be 294 pressurized-water reactor fuel assemblies, consistent with the material assumed for the seismic accident event analyzed in Section H.2.1.4.4. Transportation casks would contain up to 26 pressurized-water reactor assemblies. The analysis of an aircraft crash into a transportation cask is addressed in Chapter 6, Section 6.2.4.2. The repository spent nuclear fuel surface aging facility storage modules would be composed of thick concrete shielding with concentric steel cylinders and would contain up to 21 pressurized-water reactor fuel assemblies. The analysis of an aircraft crash into the surface aging facility determined that the aircraft would not penetrate the storage modules and determined that no release would be anticipated (DIRS 157108-Jason 2001, all).

The waste packages, which would be transported one at a time from the Waste Handling Building to the emplacement drifts, would contain only 21 pressurized-water reactor assemblies. Thus, the inventory of a waste package in transit from the Waste Handling Building to the North Portal would be far less than that of the Waste Handling Building. Waste packages in the emplacement drifts would be protected by an average of about 300 meters (1,000 feet) of overburden, plus a ground support system that would reinforce the emplacement drift tunnels. Consequently, the emplaced waste packages would not be vulnerable to impact from an aircraft.

The Waste Handling Building design includes blending and staging pools that would contain large amounts of commercial spent nuclear fuel. However, these pools would be below ground level, and contain water 15 meters (50 feet) deep. Thus, the fuel that would be contained in these pools is not considered vulnerable to an aircraft crash. The aircraft could cause damage to the pools from a high angle impact, but pool drainage would be expected to be slow due to the proximity of the surrounding earth. Furthermore, the water would limit the impact velocity, and therefore the damage potential, of incoming debris from the crash. As noted previously, the pool water would also limit release of radionuclides, and protect the fuel assemblies from an aircraft fuel fire that, as shown below, could enhance radionuclide release.

The vulnerable portion of the Waste Handling Building would include the assembly transfer areas which, as noted previously, are assumed to contain 294 pressurized-water reactor fuel assemblies. While these areas would be enclosed in thick concrete walls that could resist penetration of impacting aircraft, the concrete roof of the building would be only 20 to 25 centimeters (8 to 10 inches) thick, which was determined to be insufficient to resist penetration by an impacting aircraft.

The radionuclide release from such an event would result from two sources: (1) mechanical damage to the fuel assemblies, which could rupture the zirconium alloy cladding and pulverize a portion of the fuel pellets into particles, some of which would be small enough to be transported to the nearest individual and be inhaled, and (2) a large fire involving jet fuel carried by the aircraft. In the EIS No-Action Alternative aircraft crash assessment (Scenario 2) (Appendix K, Section K.2.5.1), it was conservatively assumed that all of the fuel pellets involved in the fire following the aircraft crash would be converted from uranium dioxide to U_3O_8, producing a powder containing radionuclides. This same assumption is made for the analysis herein. Thus, because all of the fuel pellet material in the 294 pressurized-water reactor fuel assemblies is assumed to be converted to a powder form, the particulates formed by mechanical damage would not contribute further to the source term. The fire source term in the No-Action assessment assumed that 12 percent of the U_3O_8 particles would become airborne, and approximately 1 percent of the airborne particles would be small enough to be available for inhalation into the lungs of downwind individuals. The basis for these assumptions is provided in Section K.2.5.1. Therefore, the fuel pellet respirable particulate source term is assumed to be 0.0012 of all of the fuel contained in 294 pressurized-water reactor fuel assemblies in the Waste Handling Building. The radionuclide inventory in the assemblies was assumed to be the same as the representative fuel assemblies used for repository accident analysis in the EIS.

In addition to the fuel particulate source term, other sources of radionuclides would be available for release. These sources include the crud on the outside of the zirconium alloy cladding and radioactive gases (hydrogen-3, krypton-85, carbon-14, and iodine-129) in the fuel gaps. Since the zirconium alloy is expected to burn in air at 800°C (1,472°F) (DIRS 156981-NRC 2001, p. A1-1), all crud on the zircaloy is assumed to be released, and the respirable fraction is 0.05, consistent with the seismic accident analyzed in the EIS. All of the radioactive gases are assumed to be released. Based on this discussion, the release fractions listed in Table H-8 were assumed.

These release fractions were applied to the 294 commercial spent nuclear fuel pressurized-water reactor representative fuel assemblies assumed to be in the Waste Handling Building out of the pools. The resulting radionuclide source term was input to the MACCS2 program (DIRS 103168-Chanin and Young 1998, all) and doses were calculated for the nearest offsite individual and the 80-kilometer (50-mile) population for an average weather condition. A plume rise model was also used in the analysis to account

Table H-8. Assumed release fractions of crud and radioactive gases.

Radionuclide[a]	Release fraction	Fraction respirable	RARF[a]
Crud (Co-60)	100%	0.05	0.05
H-3, Kr-85, C-14, I-129	100%	1.0	1.0
All solids	0.12	0.01	1.2×10^{-3}

a. Co = cobalt; H = hydrogen (H-3 = tritium); Kr = krypton; C = carbon; I = iodine.
b. RARF = Respirable Airborne Release Fraction.

for the plume lofting from the jet fuel fire. Since the release would be large compared to other accidents analyzed in this section, it was assumed that DOE and other Federal agencies would evacuate exposed individuals after the plume passed and also interdict consumption of contaminated food and water. Accordingly, the dose associated with immersion in and inhalation of the radioactive plume from the event was computed. The dose calculations also assume that the exposed individual remained on the contaminated land for 1 day following the event, after which they are assumed to be evacuated. Table H-9 lists the results.

Table H-9. Doses from immersion in or inhalation of radioactive plume from hypothetical aircraft crash.

Receptor	Dose	LCF[a]
Maximally exposed offsite individual	4.5 rem	0.0023
80-kilometer (50-mile) population	78 person-rem	0.039

a. LCF = likelihood of a latent cancer fatality for the maximally exposed offsite individual and estimated number of latent cancer fatalities in the exposed 80-kilometer (50-mile) 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 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 rather than gaseous 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 (DIRS 104508-CRWMS M&O 1999, 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 (DIRS 104508-CRWMS M&O 1999, Table 6-2). 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 (DIRS 104508-CRWMS M&O 1999, 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 located 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 conclusion is based on the analysis of accidents for the surface aging facility evaluated in this section and is also consistent with dry storage accident evaluations at commercial sites under similar conditions, as evaluated in Appendix K.

In view of these considerations, DOE has concluded that the waste transporter derailment accident scenario 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

Note: In an effort to ensure consistency among Yucca Mountain Project documents, DOE has altered the format of the references and some of the citations in the text in this Final EIS from those in the Draft EIS. The following list contains notes where applicable for references cited differently in the Draft EIS.

103168 Chanin and Young 1998 Chanin, D. and Young, M.L. 1998. *Code Manual for MACCS2. Preprocessor Codes COMIDA2, FGRDCF, IDCF2.* NUREG/CR-6613, Vol. 2. Washington, D.C.: U.S. Nuclear Regulatory Commission. TIC: 243881.

100204 CRWMS M&O 1996 CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 1996. *Preliminary MGDS Hazards Analysis.* B00000000-01717-0200-00130 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19961230.0011.

104695 CRWMS M&O 1996 CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 1996. *Source Terms for Design Basis Event Analyses.* BBA000000-01717-0200-00019 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19970203.0121. In the Draft EIS, this reference was cited as DOE 1996a in Appendix H.

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100217 CRWMS M&O 1997 CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 1997. *Canister Transfer System Design Analysis.* BCBD00000-01717-0200-00008 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980108.0054. In the Draft EIS, this reference was cited as DOE 1997a in Appendix H.

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104718	CRWMS M&O 1998	CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 1998. *Monitored Geologic Repository Operations Staffing Report.* BC0000000-01717-5705-00021 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.199812110036.
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152010	CRWMS M&O 2000	CRWMS M&O (Civilian Radioactive Waste Management System Management and Operating Contractor) 2000. *Repository Surface Design Engineering Files Report Supplement.* TDR-WHS-EV-000001 REV 00 ICN 1. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000626.0025.

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Appendix J

Section J.1.3.2.2.3
Figures J-8a and J-8b from Section J.1.4.2.1
Section J.2.4.3.1
Aircraft Crash Accident from Section J.3.3.1

J.1.3.2.2.3 Incident-Free Radiation Doses to Escorts. Transporting spent nuclear fuel to the Yucca Mountain site would require the use of physical security and other escorts for the shipments. Regulations (10 CFR 73.37) require escorts for highway and rail shipments. These regulations require two escorts (individuals) for truck shipments traveling in highly populated (urban) areas. One of the escorts must be in a vehicle that is separate from the shipment vehicle. For rail shipments in urban areas, at least two escorts must maintain visual surveillance of a shipment from a railcar that accompanies a cask car.

In areas that are not highly populated (suburban and rural), one escort must accompany truck shipments. The escort can ride in the cab of the shipment vehicle. At least one escort is required for rail shipments in suburban and rural areas. However, for rail shipments, the escort must occupy a railcar that is separate from the cask car and must maintain visual surveillance of the shipment at all times.

For legal-weight truck shipments, the analysis assumed that a second driver, who would be a member of the vehicle crew, would serve as an escort in all areas. The analysis assigned a second escort for travel in urban areas and assumed that this escort would occupy a vehicle that followed or led the transport vehicle by at least 60 meters (about 200 feet). The analysis assumed that the dose rate at a location 2 meters (6.6 feet) behind the vehicle would be 10 millirem per hour, which is the limit allowed by U.S. Department of Transportation regulations (49 CFR 173.441). Using this information, the analysis used the RISKIND computer program to calculate a value of approximately 0.15 millirem per hour for the dose rate 60 meters behind the transport vehicle; this is the estimated value for the dose rate in a following escort vehicle. The value for the dose rate in an escort vehicle that preceded a shipment would be lower. Because the dose rate in the occupied crew area of the transport vehicle would be less than 2 millirem per hour, the dose rate 2 meters in front of the vehicle would be much less than 10 millirem per hour, the value assumed for a location 2 meters behind the vehicle. The value of 2 millirem per hour in normally occupied areas of transport vehicles is the maximum allowed by U.S. Department of Transportation regulations (49 CFR 173.441).

To calculate the dose to escorts, the analysis assumed that escorts in separate vehicles would be required in urban areas as shipments traveled to the Yucca Mountain site. The calculations used the RISKIND computer program (DIRS 101483-Yuan et al. 1995, all); the distance of travel in urban areas provided by the HIGHWAY and INTERLINE computer codes; and the estimated speed of travel in urban areas based on data in Table J-15 to estimate the dose to escorts. For example, truck shipments could be escorted through an average of five urban areas on average for 30 minutes in each. Using these assumptions and the estimated dose rate in an escort vehicle, the estimated dose for escorts in separate vehicles is 0.38 millirem per shipment (0.38 millirem = 5 areas per shipment × 0.5 hour per area × 0.15 millirem per hour). For the 24 years of the Proposed Action, the total dose to escorts in separate vehicles would, therefore, be about 20 rem (0.38 millirem per shipment × 53,000 shipments). This dose would lead to 0.008 latent cancer fatality in the population of escorts who would be affected. If escorts were required in every population zone for legal-weight truck transport, the total occupational dose would increase by approximately 360 person-rem, or 2.5 percent.

For rail shipments, the analysis assumed that escorts would be 30 meters (98 feet) away from the end of the shipping cask on the nearest railcar. This separation distance is the sum of the:

- Length of a buffer car [about 15 meters (49 feet)] between a cask car and an escort car required by U.S. Department of Transportation regulations [see 49 CFR 174.85(b) and (d), and 49 CFR 174.700(c)],
- Normal separation between cars [a total of about 2 meters (6.6 feet) for two separations],
- Distance from the end of a cask to the end of its rail car [about 5 meters (16 feet)], and

- Assumed average distance from the escort car's near-end to its occupants [5 to 10 meters (16 to 32 feet)].

This analysis assumed that the dose rate at 2 meters (6.6 feet) from the end of the cask car would be 10 millirem per hour, the maximum allowed by U.S. Department of Transportation regulations (49 CFR 173.441). The analysis used these assumptions and the RISKIND computer program to estimate 0.71 millirem per hour as the dose rate in the occupied areas of the escort railcar. For example, an individual escort who occupied the escort car continuously for a 5-day cross-country trip would receive a maximum dose of about 85 millirem. Escorting 26 shipments in a year, this individual would receive a maximum dose of 2.2 rem. Over the 24 years of the Proposed Action, if the same individual escorted 26 shipments every year, he or she would receive a dose of about 53 rem. However, DOE would control worker exposure through administrative procedures (see DIRS 156764-DOE 1999, Article 211). Actual worker exposure would likely be 2 rem per year, or a maximum of 48 rem over 24 years. The use of the dose-to-risk conversion factors recommended by the International Commission on Radiation Protection (DIRS 101836-ICRP 1991, p. 22) projects this dose to increase the potential for the individual to contract a fatal cancer from about 23 percent (DIRS 153066-Murphy 2000, p. 5) to 25 percent. If escorts were required in every population zone, the total occupational dose could increase by as much as 1,000 person-rem, or 30 percent.

J.2.4.3.1 *Radiological Impacts of Accidents*

The analysis of risks from accidents during heavy-haul truck, rail, and legal-weight truck transport of spent nuclear fuel and high-level radioactive waste used the RADTRAN 5 computer code (DIRS 150898-Neuhauser and Kanipe 2000, all; DIRS 155430-Neuhauser, Kanipe, and Weiner 2000, all) in conjunction with an Access database and the analysis approach discussed in Section J.1.4.2. The analysis of risks due to barging used the same methodology with the exception of conditional probabilities. For barge shipments, the conditional accident probabilities and release fractions (Table J-31) for each cask response category were based on a review of other barge accident analyses.

The definitions of the accident severities listed in Table J-31 are based on the analyses reported in DIRS 152476-Sprung et al. (2000, pp. 7-75 to 7-76). DOE used the same accident severity category definitions as those used in the rail analysis described in Section J.1.4.2. If radioactive material was shipped by barge, both water and land contamination would be possible. Based on a review of Coast Guard accident data files, most barges stay afloat following a collision, justifying the assumption that there would be an airborne plume from a severe barge accident. Furthermore, severity categories 3 through 6 involve fires, which are possible because many barges do not sink after an accident. DIRS 104784-Ostmeyer (1986, all) analyzed the potential importance of water pathway contamination for a spent nuclear fuel transportation accident risk using a "worst-case" water contamination scenario. The analysis showed that the impacts of the water contamination scenario would be about one-fiftieth of the impacts of a comparable accident on land. Therefore, the analysis assumed that deposition would occur over land, not water. DOE used population distributions developed from 1990 Census data to calculate route-specific collective doses. Table J-32 lists the total accident risk for mostly rail case heavy-haul truck scenario, the mostly rail case barge scenario, and the mostly truck scenario.

Impact Speed

Impact speed exceeds 120 mph	1[a] Seal Failure: Impact (Part) 6.0×10^{-7}[b] (Ru) 6.0×10^{-7} (Cs) 2.4×10^{-8} (Kr) 8.0×10^{-1} (Crud) 2.0×10^{-3} Prob 1.53×10^{-8}[c]	11 Seal Failure: Impact (Part) 6.1×10^{-7} (Ru) 6.1×10^{-7} (Cs) 2.4×10^{-8} (Kr) 8.2×10^{-1} (Crud) 2.0×10^{-3} Prob 1.44×10^{-10}	12 Seal Failure: Impact (Part) 6.7×10^{-7} (Ru) 6.7×10^{-7} (Cs) 2.7×10^{-8} (Kr) 8.9×10^{-1} (Crud) 2.2×10^{-3} Prob 1.02×10^{-12}	13 Seal Failure: Impact (Part) 6.8×10^{-7} (Ru) 6.8×10^{-7} (Cs) 5.9×10^{-6} (Kr) 9.1×10^{-1} (Crud) 2.5×10^{-3} Prob 0	17 Shear/Puncture; Seal Failure by Fire (Part) 6.8×10^{-7} (Ru) 6.4×10^{-6} (Cs) 5.9×10^{-6} (Kr) 9.1×10^{-1} (Crud) 3.3×10^{-3} Prob 0
Impact speed from 90 to 120 mph		8 Seal Failure by Fire (Part) 6.1×10^{-7} (Ru) 6.1×10^{-7} (Cs) 2.4×10^{-8} (Kr) 8.2×10^{-1} (Crud) 2.0×10^{-3} Prob 1.13×10^{-8}	9 Seal Failure by Fire (Part) 6.7×10^{-7} (Ru) 6.7×10^{-7} (Cs) 2.7×10^{-8} (Kr) 8.9×10^{-1} (Crud) 2.2×10^{-3} Prob 8.03×10^{-11}	10 Seal Failure by Fire (Part) 6.8×10^{-7} (Ru) 6.8×10^{-7} (Cs) 5.9×10^{-6} (Kr) 9.1×10^{-1} (Crud) 2.5×10^{-3} Prob 0	16 Shear/Puncture; Seal Failure by Fire (Part) 6.8×10^{-7} (Ru) 6.4×10^{-6} (Cs) 5.9×10^{-6} (Kr) 9.1×10^{-1} (Crud) 3.3×10^{-3} Prob 0
Impact speed from 60 to 90 mph		5 Seal Failure by Fire (Part) 3.2×10^{-7} (Ru) 3.2×10^{-7} (Cs) 1.3×10^{-8} (Kr) 4.3×10^{-1} (Crud) 1.8×10^{-3} Prob 4.65×10^{-7}	6 Seal Failure by Fire (Part) 3.7×10^{-7} (Ru) 3.7×10^{-7} (Cs) 1.5×10^{-8} (Kr) 4.9×10^{-1} (Crud) 2.1×10^{-3} Prob 3.31×10^{-9}	7 Seal Failure by Fire (Part) 2.1×10^{-6} (Ru) 2.1×10^{-6} (Cs) 2.7×10^{-5} (Kr) 8.5×10^{-1} (Crud) 3.1×10^{-3} Prob 0	15 Shear/Puncture; Seal Failure by Fire (Part) 9.0×10^{-6} (Ru) 5.0×10^{-5} (Cs) 5.5×10^{-5} (Kr) 8.5×10^{-1} (Crud) 5.9×10^{-3} Prob 0
Impact speed from 30 to 60 mph		2 Seal Failure by Fire (Part) 1.0×10^{-7} (Ru) 1.0×10^{-7} (Cs) 4.1×10^{-9} (Kr) 1.4×10^{-1} (Crud) 1.4×10^{-3} Prob 5.88×10^{-5}	3 Seal Failure by Fire (Part) 1.3×10^{-7} (Ru) 1.3×10^{-7} (Cs) 5.4×10^{-9} (Kr) 1.8×10^{-1} (Crud) 1.8×10^{-3} Prob 1.81×10^{-6}	4 Seal Failure by Fire (Part) 3.8×10^{-6} (Ru) 3.8×10^{-6} (Cs) 3.6×10^{-5} (Kr) 8.4×10^{-1} (Crud) 3.2×10^{-3} Prob 7.49×10^{-8}	14 Shear/Puncture; Seal Failure by Fire (Part) 1.8×10^{-5} (Ru) 8.4×10^{-5} (Cs) 9.6×10^{-5} (Kr) 8.4×10^{-1} (Crud) 6.4×10^{-3} Prob 7.49×10^{-11}
No Impact	19 No Releases Prob 0.99993			18 Seal Failure by Fire (Part) 6.7×10^{-8} (Ru) 6.7×10^{-8} (Cs) 1.7×10^{-5} (Kr) 8.4×10^{-1} (Crud) 2.5×10^{-3} Prob 5.86×10^{-6}	
	No Fire	*End temperature: ambient to 350°C (662°F)*	*End temperature: 350°C to 750°C (662°F to 1,382°F)*	*End temperature: 750°C to 1,000°C (1,382°F to 1,832°F)*	*End temperature: 750°C to 1,000°C (1,382°F to 1,832°F)*

Cask Temperature in Fire

a. The numbers at the top of each cell refer to an accident scenario (called a case) in DIRS 152476-Sprung et al. (2000, p. 7-74).
b. (Part) is the release fraction for particulates; (Ru) is the release fraction for ruthenium; (Cs) is the release fraction for volatiles; (Kr) is the release fraction for gas; (Crud) is the release fraction for crud. The numbers next to them are the fraction that would be released in the accident.
c. The conditional probability that, if there was an accident, the particular cell would describe the accident scenario.

Figure J-8a. Impact speed and temperature matrix for pressurized-water reactor spent nuclear fuel in a steel-depleted uranium-steel truck cask.

Impact Speed

Impact speed exceeds 120 mph	3[a] Seal Failure by Impact (Part) 1.9×10^{-5}[b] (Ru) 1.9×10^{-5} (Cs) 1.8×10^{-5} (Kr) 8.0×10^{-1} (Crud) 6.4×10^{-2} Prob 4.49×10^{-9}[c]	13 Seal Failure by Impact (Part) 2.0×10^{-5} (Ru) 2.0×10^{-5} (Cs) 1.8×10^{-5} (Kr) 8.2×10^{-1} (Crud) 6.5×10^{-2} Prob 3.70×10^{-11}	14 Seal Failure by Impact (Part) 2.1×10^{-5} (Ru) 2.1×10^{-5} (Cs) 2.0×10^{-5} (Kr) 8.9×10^{-1} (Crud) 7.1×10^{-2} Prob 1.03×10^{-12}	15 Seal Failure by Impact (Part) 2.2×10^{-5} (Ru) 2.2×10^{-5} (Cs) 2.2×10^{-5} (Kr) 9.1×10^{-1} (Crud) 7.4×10^{-2} Prob 1.37×10^{-13}	19 Shear/Puncture; Seal Failure by Fire (Part) 2.2×10^{-5} (Ru) 2.3×10^{-5} (Cs) 2.2×10^{-5} (Kr) 9.1×10^{-1} (Crud) 7.4×10^{-2} Prob 1.37×10^{-16}
Impact speed from 90 to 120 mph	2 Seal Failure by Impact (Part) 1.3×10^{-5} (Ru) 1.3×10^{-5} (Cs) 8.6×10^{-6} (Kr) 8.0×10^{-1} (Crud) 4.4×10^{-2} Prob 5.68×10^{-7}	10 Seal Failure by Impact (Part) 1.3×10^{-5} (Ru) 1.3×10^{-5} (Cs) 8.8×10^{-6} (Kr) 8.2×10^{-1} (Crud) 4.5×10^{-2} Prob 4.68×10^{-9}	11 Seal Failure by Impact (Part) 1.5×10^{-5} (Ru) 1.5×10^{-5} (Cs) 9.6×10^{-6} (Kr) 8.9×10^{-1} (Crud) 4.9×10^{-2} Prob 1.31×10^{-10}	12 Seal Failure by Impact (Part) 1.5×10^{-5} (Ru) 1.5×10^{-5} (Cs) 1.4×10^{-5} (Kr) 9.1×10^{-1} (Crud) 5.1×10^{-2} Prob 1.74×10^{-11}	18 Shear/Puncture; Seal Failure by Fire (Part) 1.5×10^{-5} (Ru) 1.8×10^{-5} (Cs) 1.4×10^{-5} (Kr) 9.1×10^{-1} (Crud) 5.1×10^{-2} Prob 1.74×10^{-14}
Impact speed from 60 to 90 mph	1 Seal Failure by Impact (Part) 2.5×10^{-7} (Ru) 2.5×10^{-7} (Cs) 1.2×10^{-8} (Kr) 4.1×10^{-1} (Crud) 1.4×10^{-3} Prob 8.20×10^{-6}	7 Seal Failure by Impact (Part) 2.6×10^{-7} (Ru) 2.6×10^{-7} (Cs) 1.3×10^{-8} (Kr) 4.3×10^{-1} (Crud) 1.5×10^{-3} Prob 6.76×10^{-8}	8 Seal Failure by Impact (Part) 2.9×10^{-7} (Ru) 2.9×10^{-7} (Cs) 1.5×10^{-8} (Kr) 4.9×10^{-1} (Crud) 1.7×10^{-3} Prob 1.88×10^{-9}	9 Seal Failure by Impact (Part) 6.8×10^{-6} (Ru) 6.8×10^{-6} (Cs) 2.7×10^{-5} (Kr) 8.5×10^{-1} (Crud) 4.5×10^{-3} Prob 2.51×10^{-10}	17 Shear/Puncture; Seal Failure by Fire (Part) 8.9×10^{-5} (Ru) 5.0×10^{-5} (Cs) 5.5×10^{-5} (Kr) 8.5×10^{-5} (Crud) 5.4×10^{-5} Prob 2.51×10^{-5}
Impact speed from 30 to 60 mph		4 Seal Failure by Fire (Part) 1.0×10^{-7} (Ru) 1.0×10^{-7} (Cs) 4.1×10^{-9} (Kr) 1.4×10^{-1} (Crud) 1.4×10^{-3} Prob 2.96×10^{-5}	5 Seal Failure by Fire (Part) 1.3×10^{-7} (Ru) 1.3×10^{-7} (Cs) 5.4×10^{-9} (Kr) 1.8×10^{-1} (Crud) 1.8×10^{-3} Prob 8.24×10^{-7}	6 Seal Failure by Fire (Part) 1.4×10^{-5} (Ru) 1.4×10^{-5} (Cs) 3.6×10^{-5} (Kr) 8.4×10^{-1} (Crud) 5.4×10^{-3} Prob 1.10×10^{-7}	16 Shear/Puncture; Seal Failure by Fire (Part) 1.8×10^{-5} (Ru) 8.4×10^{-5} (Cs) 9.6×10^{-5} (Kr) 8.4×10^{-1} (Crud) 6.4×10^{-3} Prob 4.15×10^{-11}
No Impact	21 No Release Prob 0.99996			20 Seal Failure by Fire (Part) 2.5×10^{-7} (Ru) 2.5×10^{-7} (Cs) 1.7×10^{-5} (Kr) 8.4×10^{-1} (Crud) 9.4×10^{-3} Prob 4.91×10^{-5}	
	No Fire	*End temperature: ambient to 350°C (662°F)*	*End temperature: 350°C to 750°C (662°F to 1,382°F)*	*End temperature: 750°C to 1,000°C (1,382°F to 1,832°F)*	*End temperature: 750°C to 1,000°C (1,382°F to 1,832°F)*

Cask Temperature in Fire

a. The numbers at the top of each cell refer to an accident scenario (called a case) in DIRS 152576-Sprung et al. (2000, p. 7-76).
b. (Part) is the release fraction for particulates; (Ru) is the release fraction for ruthenium; (Cs) is the release fraction for volatiles; (Kr) is the release fraction for gas; (Crud) is the release fraction for crud. The numbers next to them are the fraction that would be released in the accident.
c. The conditional probability that, if there is an accident, the particular cell will describe the accident scenario.

Figure J-8b. Impact speed and temperature matrix for pressurized-water reactor spent nuclear fuel in a steel-lead-steel rail cask.

Table J-31. Release fractions and conditional probabilities for spent nuclear fuel transported by barge.

Severity category	Case	Conditional probability	Release fractions (pressurized-water reactor/boiling-water reactor)				
			Krypton	Cesium	Ruthenium	Particulates	Crud
1	21	0.994427	0.0	0.0	0.0	0.0	0.0
2	1, 4, 5, 7, 8	5.00×10^{-3}	$1.96 \times 10^{-1}/2.35 \times 10^{-2}$	$5.87 \times 10^{-9}/7.04 \times 10^{-10}$	$1.34 \times 10^{-7}/1.47 \times 10^{-8}$	$1.34 \times 10^{-7}/1.47 \times 10^{-8}$	$1.37 \times 10^{-3}/5.59 \times 10^{-4}$
3	20	5.00×10^{-6}	$8.39 \times 10^{-1}/8.39 \times 10^{-1}$	$1.68 \times 10^{-5}/1.68 \times 10^{-5}$	$2.52 \times 10^{-7}/2.52 \times 10^{-7}$	$2.52 \times 10^{-7}/2.52 \times 10^{-7}$	$9.44 \times 10^{-3}/9.44 \times 10^{-2}$
4	2, 3, 10	5.00×10^{-4}	$8.00 \times 10^{-1}/8.00 \times 10^{-1}$	$8.71 \times 10^{-6}/8.71 \times 10^{-6}$	$1.32 \times 10^{-5}/1.32 \times 10^{-5}$	$1.32 \times 10^{-5}/1.32 \times 10^{-5}$	$4.42 \times 10^{-3}/4.42 \times 10^{-2}$
5	6	0.0	$8.35 \times 10^{-1}/8.37 \times 10^{-1}$	$3.60 \times 10^{-5}/4.12 \times 10^{-5}$	$1.37 \times 10^{-5}/1.82 \times 10^{-5}$	$1.37 \times 10^{-5}/1.82 \times 10^{-5}$	$5.36 \times 10^{-3}/5.43 \times 10^{-3}$
6	9,11,12,13,14,15,16, 17,18,19	1.30×10^{-6}	$8.47 \times 10^{-1}/8.45 \times 10^{-1}$	$5.71 \times 10^{-5}/7.30 \times 10^{-5}$	$4.63 \times 10^{-5}/5.94 \times 10^{-5}$	$1.43 \times 10^{-5}/1.96 \times 10^{-5}$	$1.59 \times 10^{-2}/1.60 \times 10^{-2}$

Table J-32. Comparison of accident risks for the mostly rail heavy-haul truck and barge shipping scenarios.[a]

Category	Mostly rail (heavy-haul option–24 sites)	Mostly rail (barge option–17 of 24 heavy-haul sites)	Mostly truck
Population dose (person-rem)	0.89	1.5	0.5
Estimated LCFs[b]	0.00045	0.001	0.0002
Traffic fatalities[c]	2.7	2.7	4.5

a. Impacts are totals over 24 years.
b. LCF = latent cancer fatality.
c. Traffic fatality impacts for mostly rail scenarios are averages of range of estimated traffic fatality impacts (2.3 to 3.1) for national transportation for the Proposed Action.

Excerpt from Section J.3.3.1

2. *Aircraft Crash Accident.* Two of the three intermodal transfer station locations are near airports that handle large volumes of air traffic. The Apex/Dry Lake location is about 16 kilometers (10 miles) northeast of the Nellis Air Force Base runways. Between 60,000 and 67,000 takeoffs and landings occur at Nellis Air Force Base each year (DIRS 148083-Luedke 1997, all). The Sloan/Jean intermodal transfer area begins about 16 kilometers southwest of McCarran International Airport in Las Vegas. In 1996, McCarran had an average of 1,300 daily aircraft operations (DIRS 104725-Best 1998, all). Because of the large number of aircraft operations at these airports, the probability of an aircraft crash on the proposed intermodal transfer station could be within the credible range. To assess the consequences of an aircraft crash, an analysis evaluated the ability of large aircraft parts to penetrate the shipping casks. The parts with the highest chance of penetration are the jet engines and jet engine shafts (DIRS 101810-DOE 1996, p. 58). The analysis used a recommended formula (DIRS 101810-DOE 1996, p. 69) for predicting the penetration of steel targets, as follows:

$$T^{1.5} = 0.5 \times M \times V^2 \div 17{,}400 \times K_s \times D^{1.5}$$

where:

- T = predicted thickness to just perforate a steel plate (inches)
- M = projectile mass (weight/gravitational acceleration)
- V = projectile impact velocity (feet per second)
- K_s = constant depending on the grade of steel (usually about 1.0)
- D = projectile diameter (inches)

The primary jet aircraft operating at Nellis Air Force Base are the F-15 and F-16 high-performance fighters, which represent more than 70 percent of Base aircraft operations (DIRS 103472-USAF 1999, pp. 1-34 and 1-35). Because both of these aircraft use the same engine (DIRS 156757-Morissette 2001, p. 1), DOE selected that engine as the military aircraft engine for the penetration analysis. For the commercial aircraft penetration analysis, DOE selected the B-767, a large widely used commercial jet. Table J-52 lists the engine characteristics for these aircraft.

Table J-52. Aircraft engine projectile characteristics.[a]

Aircraft	Engine weight (kilograms)[b]	Engine diameter (centimeters)[c]	Engine shaft weight (kilograms)[b]	Engine shaft diameter (centimeters)[c]
F-15, 16	1,900	91	25	7.6
B-767	4,500	240	110	15

a. Source: DIRS 156757-Morissette (2001, all).
b. To convert kilograms to pounds, multiply by 2.2046.
c. To convert centimeters to inches, multiply by 0.3937.

The velocity selected for the penetration analysis was 500 feet per second (550 kilometers or 340 miles per hour). This velocity is based on the discussion in DIRS 101810-DOE (1996, p. C-7 that indicates that impact velocities would typically be less than 500 feet per second. Because the selected intermodal transfer station would be near airports, anticipated aircraft velocities would be less because operations would involve takeoffs and landings using lower speeds. Thus, the selection of 500 feet per second for the impact velocity is conservative.

The results in Table J-53 indicate that none of the aircraft projectiles considered would penetrate the shipping casks, which would have steel walls about 18 centimeters (7 inches) thick (DIRS 101837-JAI 1996, all).

Table J-53. Results of aircraft projectile penetration analysis.[a]

Projectile	Penetration thickness (centimeters)[a]
F-15, 16 engine	6.7
F-15, 16 engine shaft	4.5
B-757 engine	4.7
B-757 engine shaft	6.3

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

This evaluation found no credible accidents with the potential for radioactive release at an intermodal transfer station. In a separate analysis performed following the events of September 11, 2001, Bechtel SAIC Company, LLC analysts reached a similar conclusion that the impact of large and small missiles produced during an aircraft crash would not perforate or crack a cask (DIRS 157210-BSC 2001, p. iii). However, the analysis did not preclude the potential for the impact and resultant fire to cause seal failure. The consequences of such an event would be less than 0.65 latent cancer fatality if the crash occurred in an urban area (DIRS 157210-BSC 2001, all).

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CONVERSIONS

METRIC TO ENGLISH			ENGLISH TO METRIC		
Multiply	**by**	**To get**	**Multiply**	**by**	**To get**
Area					
Square meters	10.764	Square feet	Square feet	0.092903	Square meters
Square kilometers	247.1	Acres	Acres	0.0040469	Square kilometers
Square kilometers	0.3861	Square miles	Square miles	2.59	Square kilometers
Concentration					
Kilograms/sq. meter	0.16667	Tons/acre	Tons/acre	0.5999	Kilograms/sq. meter
Milligrams/liter	1[a]	Parts/million	Parts/million	1[a]	Milligrams/liter
Micrograms/liter	1[a]	Parts/billion	Parts/billion	1[a]	Micrograms/liter
Micrograms/cu. meter	1[a]	Parts/trillion	Parts/trillion	1[a]	Micrograms/cu. meter
Density					
Grams/cu. cm	62.428	Pounds/cu. ft.	Pounds/cu. ft.	0.016018	Grams/cu. cm
Grams/cu. meter	0.0000624	Pounds/cu. ft.	Pounds/cu. ft.	16,025.6	Grams/cu. meter
Length					
Centimeters	0.3937	Inches	Inches	2.54	Centimeters
Meters	3.2808	Feet	Feet	0.3048	Meters
Kilometers	0.62137	Miles	Miles	1.6093	Kilometers
Temperature					
Absolute					
Degrees C + 17.78	1.8	Degrees F	Degrees F – 32	0.55556	Degrees C
Relative					
Degrees C	1.8	Degrees F	Degrees F	0.55556	Degrees C
Velocity/Rate					
Cu. meters/second	2118.9	Cu. feet/minute	Cu. feet/minute	0.00047195	Cu. meters/second
Grams/second	7.9366	Pounds/hour	Pounds/hour	0.126	Grams/second
Meters/second	2.237	Miles/hour	Miles/hour	0.44704	Meters/second
Volume					
Liters	0.26418	Gallons	Gallons	3.78533	Liters
Liters	0.035316	Cubic feet	Cubic feet	28.316	Liters
Liters	0.001308	Cubic yards	Cubic yards	764.54	Liters
Cubic meters	264.17	Gallons	Gallons	0.0037854	Cubic meters
Cubic meters	35.314	Cubic feet	Cubic feet	0.028317	Cubic meters
Cubic meters	1.3079	Cubic yards	Cubic yards	0.76456	Cubic meters
Cubic meters	0.0008107	Acre-feet	Acre-feet	1233.49	Cubic meters
Weight/Mass					
Grams	0.035274	Ounces	Ounces	28.35	Grams
Kilograms	2.2046	Pounds	Pounds	0.45359	Kilograms
Kilograms	0.0011023	Tons (short)	Tons (short)	907.18	Kilograms
Metric tons	1.1023	Tons (short)	Tons (short)	0.90718	Metric tons
ENGLISH TO ENGLISH					
Acre-feet	325,850.7	Gallons	Gallons	0.000003046	Acre-feet
Acres	43,560	Square feet	Square feet	0.000022957	Acres
Square miles	640	Acres	Acres	0.0015625	Square miles

a. This conversion is only valid for concentrations of contaminants (or other materials) in water.

METRIC PREFIXES

Prefix	Symbol	Multiplication factor		
exa-	E	1,000,000,000,000,000,000	=	10^{18}
peta-	P	1,000,000,000,000,000	=	10^{15}
tera-	T	1,000,000,000,000	=	10^{12}
giga-	G	1,000,000,000	=	10^{9}
mega-	M	1,000,000	=	10^{6}
kilo-	k	1,000	=	10^{3}
deca-	D	10	=	10^{1}
deci-	d	0.1	=	10^{-1}
centi-	c	0.01	=	10^{-2}
milli-	m	0.001	=	10^{-3}
micro-	µ	0.000 001	=	10^{-6}
nano-	n	0.000 000 001	=	10^{-9}
pico-	p	0.000 000 000 001	=	10^{-12}