F.4.2.2.1.3 Cultural Resources

No direct impacts on any cultural resources would be expected from the construction and operation of a new dry storage facility. Surveys of previously disturbed areas at the Idaho National Engineering Laboratory found no eligible cultural resources. Native American treaty rights that would affect any future land use on the Idaho National Engineering Laboratory would not be impacted (DOE, 1995g). Because activities associated with spent nuclear fuel management would take place within existing facility areas currently engaged in similar activities, DOE does not expect any impacts to important Native American resources from alteration of the visual setting or noise associated with the construction or operation of any new facilities. DOE has developed plans to be in full compliance with cultural resource laws (DOE, 1995g).

F.4.2.2.1.4 Aesthetic and Scenic Resources

Construction and operation of a new dry storage facility would not adversely impact aesthetic or scenic resources. A new dry storage facility would not be visible from any onsite or offsite public access roads. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility (DOE, 1995g).

F.4.2.2.1.5 Geology

There are no unique geologic features or minerals of economic value on the Idaho National Engineering Laboratory that would be adversely impacted by site development. Construction of a new dry storage facility would result in localized impacts to surficial soils, and would necessitate the clearing and grading of 3.7 ha (9 acres). Site preparation, land shaping, and grading activities associated with construction would present a slight to moderate erosion hazard, but would be controlled and minimized by implementing best-management practices. The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

F.4.2.2.1.6 Air Quality

Nonradiological Emissions: Potential impacts from construction activities at the Idaho National Engineering Laboratory would include fugitive dust from construction activities (e.g., clearing of land, grading, road preparation) and vehicle emissions from the heavy equipment utilized during the construction phase of the project. Construction of a new dry fuel storage facility would be located near the center of the Idaho National Engineering Laboratory. The construction of this facility would require disturbance of approximately 3.7 ha (9 acres) of land. However, the overall construction impacts to the ambient air quality of the region should be minimal due to the short duration (3 months to 6 years). As outlined in Table F-42, the ambient air quality impacts associated with construction-related activities would be minimal and the Idaho National Engineering Laboratory compliance with Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population.

Table F-42 Estimated Maximum Concentrations of Criteria Pollutants at the Idaho National Engineering Laboratory Attributable to New Dry Storage Construction

Pollutant	Averaging Time	Ambient Standard[a]	Baseline Concentration	Construction Activities
Idaho National Engineering Laboratory Boundary (μg/m³)[b]				
• Particulate Matter (PM_{10})	24-hr	150	112	0.0274
	Annual	50	19	0.0014
• Carbon Monoxide	1-hr	40,000	1,200	2.42
	8-hr	10,000	340	0.97
• Sulfur Dioxide	3-hr	1,300	534	0.397
	24-hr	365	238	0.085
	Annual	80	4.2	0.004
• Nitrogen Dioxide	Annual	100	14.1	0.068
Public Roads Boundary (μg/m³)				
• Particulate Matter (PM_{10})	24-hr	150	112	0.0050
	Annual	50	19	0.0006
• Carbon Monoxide	1-hr	40,000	1,200	6.69
	8-hr	10,000	340	1.28
• Sulfur Dioxide	3-hr	1,300	534	0.727
	24-hr	365	238	0.117
• Nitrogen Dioxide	Annual	100	14.1	0.211
Craters of the Moon Boundary (μg/m³)				
• Particulate Matter (PM_{10})	24-hr	150	112	0.00037
	Annual	50	19	0.00003
• Carbon Monoxide	1-hr	40,000	1,200	0.61
	8-hr	10,000	340	0.08
• Sulfur Dioxide	3-hr	1,300	534	0.054
	24-hr	365	238	0.009
	Annual	80	4.2	0.0006
• Nitrogen Dioxide	Annual	100	14.1	0.009

[a] *Source: DOE, 1995g.*

[b] *To convert to μg/ft³, multiply by 0.0283.*

No nonradiological air emissions would be expected during operation of a new dry storage facility. Any emissions would be directly attributable to front-end wet storage activities only.

Radiological Emissions: No radiological emissions would be produced during construction of a new dry storage facility.

Based on fuel drying and storage operations conducted at the Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides, such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively higher contamination) would help provide contamination control. High-efficiency particulate air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides released during the cask opening operation depends on the following parameters: (1) the number of spent nuclear fuel clad defects, (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask, and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a gaseous state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage, small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

For this analysis, radiological emissions from the operation of a new dry storage facility for foreign research reactor spent nuclear fuel were calculated based on the methodology and assumptions described in Section F.6. The radiological consequences of air emissions from dry storage operation at the Idaho National Engineering Laboratory are discussed in Section F.4.2.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.2.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 6,500 million l (1,717 million gal) for the Idaho National Engineering Laboratory, these amounts represent approximately a 0.03 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Idaho National Engineering Laboratory.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Idaho National Engineering Laboratory. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Idaho National Engineering Laboratory could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Idaho National Engineering Laboratory would still be well within the design capacities of treatment systems at the Idaho National Engineering Laboratory. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.2.2.1.8 Ecology

Terrestrial Resources: DOE expects that construction impacts, which would include the loss of some wildlife habitat due to land clearing and facility development, would be greatest under the Regionalization and Centralization Alternatives under the Programmatic SNF&INEL Final EIS at the Idaho National Engineering Laboratory. Construction impacts from foreign research reactor spent nuclear fuel storage would not be significant because the construction activity would take place either within the boundaries of heavily developed areas or adjacent to those areas. However, construction activities could provide opportunities for the spread of exotic plant species, such as cheatgrass and Russian thistle (DOE, 1995g).

Wetlands: There would be no construction impacts to wetlands, which would be excluded from development, and impacts to threatened and endangered species would be unlikely given the location of previously-developed areas and the maximum size of the affected area of 3.7 ha (9 acres). Construction activities at the Idaho National Engineering Laboratory probably would not affect either of the endangered species found onsite (e.g., bald eagle and peregrine falcon). Both of these birds of prey are associated with riparian areas, wetlands, and larger bodies of water (e.g., reservoirs) and inhabit dry upland areas only temporarily when migrating. Disturbance to other sensitive (but not Federally-listed) species (e.g., the burrowing owl, northern goshawk, ferruginous hawk, Swainson's hawk, gyrfalcon, Townsend's western big-eared bat, and pygmy rabbit) would be possible but unlikely given the scale of the planned construction. Any impacts would be negligible and would last only as long as construction activities continue (DOE, 1995g).

Threatened and Endangered Species: Representative impacts from operations would include the disturbance and displacement of animals (such as the pronghorn antelope) caused by the movement and noise of personnel, equipment, and vehicles. Such impacts would be greatest under the Regionalization by Fuel Type and Geography, and Centralization Alternatives under the Programmatic SNF&INEL Final EIS at the Idaho National Engineering Laboratory, which would involve a generally higher level of operational activity; however, these impacts would be minor (DOE, 1995g). DOE has completed consultations with the U.S. Fish and Wildlife Service regarding threatened and endangered species for the potential construction site of foreign research reactor spent nuclear fuel storage facilities at the Idaho National Engineering Laboratory, as required by the Endangered Species Act. Letters regarding consultation under the Endangered Species Act are included in Volume 2, Appendix B of the Programmatic SNF&INEL Final EIS (DOE, 1995[illegible]

F.4.2.2.1.10 Traffic and Transportation

Construction materials, wastes, and excavated materials would be transported both onsite and offsite. These activities would result in increases in operation of personal-use vehicles by commuting construction workers, commercial truck traffic, and in traffic associated with the daily operations of the Idaho National Engineering Laboratory. Again, traffic congestion would not be a significant problem.

Traffic due to operations of a new dry storage facility would not increase site levels because the required workers would be drawn from the existing Idaho National Engineering Laboratory labor force.

F.4.2.2.1.11 Occupational and Public Health and Safety

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. Table F-43 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Idaho National Engineering Laboratory. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-43 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Idaho National Engineering Laboratory (New Dry Storage)

Facility	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Receipt/Unloading at: • New Dry Storage Facility[a]	0.00056	2.8×10^{-10}	0.0045	0.0000023
Storage at: • New Dry Storage Facility	0	0	0	0

[a] *The doses for this new dry storage facility are assumed to be equal to those for IFSF/CPP-749.*

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 2B involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into the existing dry and wet storage facilities (IFSF/CPP-749 and FAST) during Phase 1, the preparation of 161 transportation casks for shipment to a dry storage facility at the end of Phase 1, and the receipt of 193 shipments of foreign research reactor spent nuclear fuel at the new dry storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. Collective doses were calculated for both dry storage designs, the vault and the dry cask. The assumptions and methodology used to calculate the doses are described in Section F.5 of this appendix.

Table F-44 presents the doses that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Idaho National Engineering Laboratory. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-44 Handling-Related Impacts to Workers at the Idaho National Engineering Laboratory (New Dry Storage)

	Worker Population Dose (person-rem)		*Worker Population Risk (LCF)*	
	New Dry Storage Cask	*New Dry Storage Vault*	*New Dry Storage Cask*	*New Dry Storage Vault*
Phases 1 and 2[a]	424	370	0.17	0.15
Phases 1 and 2[b]	416	363	0.17	0.15

[a] *Phase 1 at IFSF/CPP-749*

[b] *Phase 1 at FAST*

F.4.2.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Idaho National Engineering Laboratory would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-45. These requirements represent a small percent of current requirements for the Idaho National Engineering Laboratory. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Idaho National Engineering Laboratory is expected to decrease because of changes in site mission and a general reduction in employment.

F.4.2.2.1.13 Waste Management

Construction of a new dry storage facility at the Idaho National Engineering Laboratory would generate 1,800 m^3 (2,400 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-46. These quantities represent a very small percent increase above current levels at the Idaho National Engineering Laboratory. Existing waste management storage and disposal activities at the Idaho National Engineering Laboratory could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Idaho National Engineering Laboratory waste management capacities would be minimal.

Table F-46 Annual Waste Generated for New Dry Storage at the Idaho National Engineering Laboratory

Waste Form	Baseline Site Generation	Dry Storage Generation	Percent Increase
High-Level (m^3/yr)	750	none	0 percent
Transuranic (m^3/yr)	712	none	0 percent
Solid Low-Level (m^3/yr)	4,795	22[a] 1[b]	0.5 percent[a] 0.02 percent[b]
Wastewater (l/yr)	540,000,000	1,590,000[a] 400,000[b]	0.29 percent[a] 0.074 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.2.2.2 Wet Storage

Analysis option 2C involves long-term wet storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory. This analysis option would require the construction of a new wet storage facility at the site (Implementation Alternative 5 of Management Alternative 1).

F.4.2.2.2.1 Land Use

A new wet storage facility could be located in one of several developed areas, including the ICPP. These areas, which have already been developed for industrial use, occupy about 4,560 ha (11,400 acres). Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land. This represents about 0.06 percent of the developed space at these areas. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at any of the areas would significantly impact land use patterns on the Idaho National Engineering Laboratory.

F.4.2.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be $449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $112.2 million. This represents approximately 18.7 percent of the estimated FY 1995 total expenditures for the Idaho National Engineering Laboratory (600 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage.

These costs represent about 3.8 percent and 0.6 percent of FY 1995 total expenditures for the Idaho National Engineering Laboratory. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Idaho National Engineering Laboratory of approximately 11,600 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be small to both the region of influence and the Idaho National Engineering Laboratory.

F.4.2.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.2.2.1.3).

F.4.2.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.2.2.1.4).

F.4.2.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.2.2.1.5).

F.4.2.2.2.6 Air Quality

For other building areas which would be sources of airborne radioactive contamination, the heating, ventilation, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite concentrations (DOE, 1989b).

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the IFSF at the Idaho National Engineering Laboratory. The annual air emission level for the IFSF was designed to result in ground-level concentrations of less than 0.003 percent of DOE Order 5480.1B limits for uncontrolled areas.

Radiological emissions from the operation of the new wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading, and storage are provided in Section F.6.6.1.

No radiological emissions would be produced during construction of a new wet storage facility.

F.4.2.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities and storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-47 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Idaho National Engineering Laboratory for wet storage. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-47 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

Facility	MEI Dose (mrem/yr)	Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Receipt/Unloading at • New Wet Storage Facility	0.00038	1.9×10^{-10}	0.0031	0.0000016
Storage at: • New Wet Storage Facility	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 2C involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into the existing facilities (IFSF/CPP-749 and FAST) during Phase 1, the preparation of 161 transportation casks for shipment to a wet storage facility at the end of Phase 1, and the receipt of 193 shipments directly from the ports into the new wet storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-48 presents the population dose that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Idaho National Engineering Laboratory. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-48 Handling-Related Impacts to Workers at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

Facility	Worker Population Dose (person-rem)	Worker Population Risk (LCF)
Phase 1: IFSF/CPP-749	257	0.10
Phase 1 and Phase 2	367	0.15
Phase 1: FAST	250	0.10
Phase 1 and Phase 2	360	0.14

F.4.2.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at the Idaho National Engineering Laboratory would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-49. These requirements represent a small percent of current requirements for the Idaho National Engineering Laboratory. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity at the Idaho National Engineering Laboratory is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-49 Annual Utility and Energy Requirements for New Wet Storage at the Idaho National Engineering Laboratory (Implementation Alternative 5 to Management Alternative 1)

Table F-50 Annual Waste Generated for New Wet Storage at the Idaho National Engineering Laboratory (Implementation Alternative 5 to Management Alternative 1)

Waste Form	Baseline Site Generation	Wet Storage Generation	Percent Increase
High-Level (m^3/yr)	750	none	0 percent
Transuranic (m^3/yr)	712	none	0 percent
Solid Low-Level (m^3/yr)	4,795	16[a] 1[b]	0.33 percent 0.02 percent
Wastewater (l/yr)	540,000,000	1,590,000[a] 400,000[b]	0.3 percent 0.07 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.2.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Idaho National Engineering Laboratory is presented here based on the methodology presented in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated. Information concerning radiation doses to individuals and the general population are the same as set forth in Section F.4.1.3.

Table F-51 presents frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-51 Frequency and Consequences of Accidents at the Idaho National Engineering Laboratory

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
Dry Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	1.3	0.67	15	28
• Dropped Fuel Cask	0.0001	0.074	0.0033	0.83	0.12
• Aircraft Crash w\Fire	1×10^{-6}	180	2.9	2,000	120
Wet Storage Accidents[b]					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0016	0.0036	0.43	0.14
• Accidental Criticality	0.0031	28	30	140	1800
• Aircraft Crash	1×10^{-6}	22	9.8	250	400

[a] *IFSF/CPP-749 or New Dry Storage Facility*

[b] *New Wet Storage and FAST facility*

Multiplying the frequency of each accident times its

Table F-52 Annual Risks of Accidents at Idaho National Engineering Laboratory

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	1.1×10^{-7}	5.5×10^{-8}	0.0012	0.0000018
• Dropped Fuel Cask	3.7×10^{-12}	1.7×10^{-13}	4.2×10^{-8}	4.8×10^{-12}
• Aircraft Crash w\Fire	9.0×10^{-11}	1.5×10^{-12}	0.0000010	4.8×10^{-11}
Wet Storage Accidents[b]				
• Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
• Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
• Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}

[a] *IFSF/CPP-749 or New Dry Storage Facility*

[b] *New Wet Storage and FAST Facility*

Table F-53 presents the frequency and consequences of the accidents analyzed for Idaho National Engineering Laboratory for new wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Idaho National Engineering Laboratory. These annual risks are multiplied by the maximum duration of implementation alternative at each site to obtain conservative estimates of risks at the Idaho National Engineering Laboratory. Table F-54 presents the risk estimates from this implementation alternative.

Table F-53 Frequency and Consequences of Accidents at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	Consequences			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
Wet Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0016	0.0036	0.43	0.14
• Accidental Criticality	0.0031	28	30	140	1800
• Aircraft Crash	1×10^{-6}	22	9.8	250	400

[a] *New Wet Storage Facility*

Table F-54 Annual Risks of Accidents at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Wet Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
• Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
• Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}

[a] *New Wet Storage Facility*

F.4.2.3.1 Secondary Impact of Radiological Accidents at the Idaho National Engineering Laboratory

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the calculations presented in Section F.4.2.3. Radiological accidents that resulted in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Idaho National Engineering Laboratory, the radiological accident with the highest MEI dose is the aircraft crash into a dry storage facility with fire (Table F-51). For this accident, the MEI dose would be 180 mrem. For the air immersion and ground surface pathways only, the dose would be 3.1 mrem, which is less than the 100 mrem limit used in this analysis. Therefore, no secondary impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies from radiological accidents involving foreign research reactor spent nuclear fuel storage are expected at the Idaho National Engineering Laboratory.

F.4.2.4 Cumulative Impacts at the Idaho National Engineering Laboratory

This section presents the cumulative impacts of the proposed action, potential impacts of other major contemplated DOE actions and current activities at the Idaho National Engineering Laboratory. The contemplated DOE actions are the proposed construction and operation of an accelerator facility for tritium production (along with associated support facilities) (DOE, 1995d), the management of DOE-owned spent nuclear fuel discussed in Appendix B of the Programmatic SNF&INEL Final EIS (DOE, 1995g), and the storage and disposition of weapons-usable fissile materials at the Idaho National Engineering Laboratory site.

Tables F-55 and F-55A summarize the cumulative impacts for land use, socioeconomics, nonradiological air quality, occupational and public health and safety, energy and water consumption, and waste generation. As shown in the tables, the contribution of foreign research reactor spent nuclear fuel management to the cumulative impacts at the Idaho National Engineering Laboratory would be minimal.

F.4.2.5 Unavoidable Adverse Environmental Impacts

The construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would result in some adverse impacts to the environment. Changes in designs and other methods of mitigation could eliminate, avoid, or reduce most of these to minimal levels. The following paragraphs identify adverse impacts that mitigation could not reduce to minimal levels or avoid altogether.

The generation of some fugitive dust during construction would be unavoidable, but would be controlled by water and dust suppressants. Similarly, construction activities would result in some minor, yet unavoidable, noise impacts from heavy equipment, generators, and vehicles.

Table F-55 Cumulative Impacts at the Idaho National Engineering Laboratory

Environmental Impact Parameter	*FRR SNF Contribution*	*Current Activities*[a]	*Other Activities*[a]	*Cumulative Impact*
Land Use (acres)	9	11,400[b]	604	12,013[b]
Socioeconomics (persons)	190[c]/30[d]	(e)	1980[c]/1080[d]	2,170[c]/1,110[d]
Air Quality (nonradiological)	See Table F-55A	See Table F-55A	See Table F-55A	See Table F-55A
Occupational and Public Health and Safety				
• MEI Dose (rem/yr)	5.6×10^{-7}	0.000056	0.0000057	0.000062
LCF (per year)	2.8×10^{-10}	2.8×10^{-8}	2.8×10^{-9}	3.1×10^{-8}
• Population Dose (person-rem/yr)	0.0045	0.34	32	32.3
LCF (per year)	2.25×10^{-6}	0.00017	0.016	0.016
• Worker Collective dose (person-rem/yr)	10[f]	30	344	384

Table F-55A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Idaho National Engineering Laboratory Boundary[a]

Pollutant	Averaging Time	Regulatory Standard ($\mu g/m^3$)	Cumulative Concentration ($\mu g/m^3$)[b]

the survey would be conducted prior to any disturbance. If cultural resources were discovered, they would be evaluated according to National Register criteria. Wherever possible, important resources would be left undisturbed. If the impacts are determined to be adverse and it is not feasible to leave the resource undisturbed, then measures would be initiated to reduce impacts. All mitigation plans would be developed in consultation with the State Historic Preservation Office and the Advisory Council on Historic Preservation and would conform to appropriate standards and guidelines established for historic preservation activities by the Secretary of the Interior (DOE, 1995g).

Some actions may affect areas of religious, cultural, or historic value to Native Americans. DOE has implemented a Working Agreement to ensure communication with the Shoshone-Bannock Tribes, especially relating to the treatment of archaeological sites during excavation, as mandated by the Archaeological Resources Protection Act; the protection of human remains, as required under the Native American Graves Protection and Repatriation Act; and the free exercise of religion as protected by the American Indian Religious Freedom Act. In keeping with DOE Native American policy, DOE Order [illegible] to be defined in the Final Cultural Resources Management Plan, DOE would

- DOE Emergency Operations Centers,
- County and State Emergency Command Centers,
- medical, health physics, and industrial hygiene specialists,
- protective clothing and equipment (respirators, breathing air supplies, etc.), and
- periodic training exercises and drills within and between the organizations involved in implementing the response plans.

F.4.3 Hanford Site

If the Hanford Site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period required for the Hanford Site to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, this period (Phase 1) is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2) the Hanford Site would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory, and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Hanford Site until ultimate disposition.

The amount of spent nuclear fuel that would be received and managed at the Hanford Site under Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly in Phase 2, the Hanford Site could receive TRIGA foreign research reactor spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1, Western foreign research reactor spent nuclear fuel under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative. As a Phase 2 site, the Hanford Site would receive and manage foreign research reactor spent nuclear fuel at a new dry storage facility constructed on the 200 Area Plateau or the FMEF, which is a partially completed, large, hot cell facility. The new dry storage facility is described in Section 2.6.5.1.1. Description of the FMEF is provided in Appendix F, Section F.3.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Hanford Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

3A. The spent nuclear fuel that was managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Hanford Site where it would be managed at a new dry storage facility constructed either at the 200 Area Plateau or at the FMEF. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of this analysis, the total amount of storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Hanford Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Hanford Site would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site and manage it in facilities sized accordingly. The impacts from the management of this lesser amount of spent nuclear fuel would be bounded by analysis option 3A (above).

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Hanford Site would receive only HEU from the Idaho National Engineering Laboratory and/or the Savannah River Site. The amount would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Hanford Site would be bounded by analysis option 3A (above).

nuclear fuel would be taken. The choices do not affect the management impacts at the Hanford Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Hanford Site for Phase 2 until ultimate disposition. For this implementation alternative, an analysis option 3B, which is similar to 3A, is considered as follows:

3B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Hanford Site where it would be managed at a new wet storage facility constructed at either the 200 Area Plateau or the WNP-4 Spray Pond facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of this analysis, the total amount of spent nuclear fuel to be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements). If the Hanford Site receives only TRIGA spent nuclear fuel from the Idaho National Engineering Laboratory, or only western fuel, the dry storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Hanford Site would not be considered as a site for chemical separation.

Under Management Alternative 3 (Hybrid Alternative) the Hanford Site is not considered.

F.4.3.1 Existing Facilities

Existing facilities at the Hanford Site include the FMEF and the WNP-4 Spray Cooling Pond for dry and wet storage, respectively, of foreign research reactor spent nuclear fuel. For this analysis, existing facilities at the Hanford Site were considered essentially as new because of the significant modifications that would be required to use them for foreign research reactor spent nuclear fuel storage. Handling and transfer operations at the FMEF and the WNP-4 Spray Cooling Pond would be used to support new dry and wet storage facilities, respectively. The evaluation of potential environmental impacts is presented in Section F.4.3.2 and reflects the foreign research reactor spent nuclear fuel storage options described in Section F.4.3.

F.4.3.2 New Facilities (Phase 2)

Analysis options 3A and 3B involve the use of new or major additions to existing facilities as discussed above. The environmental impacts analyzed relate to the construction and operation of these facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities, and energy; and waste management.

F.4.3.2.1 Dry Storage

Dry storage is associated with analysis option 3A, which would require the construction of a new dry storage facility near the 200 Area Plateau or at the FMEF (FMEF currently has handling and transfer, but not adequate storage capabilities). The dry storage option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 and Appendix F, Section F.3. There are no environmental impact parameters that would discriminate between the two designs. For the purpose of this analysis the impacts from the larger dry vault design are presented.

F.4.3.2.1.1 Land Use

A new dry storage facility would be located in either the 200 Area Plateau or at the FMEF in the 400 Area. These areas have been generally developed for industrial use. Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land at either area. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at either area would significantly impact land use patterns on the Hanford Site.

F.4.3.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be \$370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about \$92.5 million. This represents approximately 7.2 percent of the estimated FY 1995 total expenditures for the Hanford Site (1,288 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new dry storage facility are estimated to be \$15.6 million for receipt and handling and \$0.6 million for storage. These costs represent approximately 1.2 percent and 0.05 percent of FY 1995 total expenditures for the Hanford Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from direct and secondary construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Hanford Site of approximately 18,500 persons, the relative socioeconomic impact of this temporary [illegible] would be insignificant. Direct employment associated with receipt on the location of the 200 Area Plateau relative to sacred and culturally important areas which have been identified through ethno-historical research and interviews with elders of bands that formerly used the Hanford Site (DOE, 1995g).

Modification of FMEF for dry storage would be inside the fence of the 400 Area. No cultural resources are known to exist within that area. Because of its location, no cultural resources on the Hanford Site would be disturbed by construction.

F.4.3.2.1.4 Aesthetic and Scenic Resources

Any changes caused by construction and operation of either dry storage facility would be consistent with the existing overall visual environment of the Hanford Site. Topographic features would obstruct both candidate storage sites from the view of populated areas. Although the new dry storage facility could be seen from the farmland bluffs that overlook the Columbia River to the east, these lands are on private property that is not readily accessible to the public. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility (DOE, 1995g).

F.4.3.2.1.5 Geology

There are no unique geologic features or minerals of economic value on the Hanford Site that would be adversely impacted by site development. Construction of a new dry storage facility would result in localized impacts to surficial soils and would necessitate the clearing and grading of 3.7 ha (9 acres). Site preparation, land shaping, and grading activities associated with construction would present a slight to moderate erosion hazard, but would be controlled and minimized by implementing best-management practices. The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

F.4.3.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts associated with construction include generation of fugitive dust (particulate matter) and smoke from earth moving and clearing operations and emissions from construction equipment. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed surfaces.

Emissions of sulfur dioxide and nitrogen dioxide would result entirely from diesel exhaust. For this analysis, all vehicular emissions were conservatively assumed to occur within 1 year during 200 ten-hour work days. As shown in Table F-56, air quality impacts associated with construction-related activities would be minimal, and compliance with Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population.

Table F-56 Estimated Maximum Concentrations of Criteria Pollutants at the Hanford Site Attributable to New Dry Storage Construction

		Ambient	*Baseline*	*Construction*
Hanford Site Boundary ($\mu g/m^3$)				

For this analysis, radiological emissions from the operation of a new dry storage facility for foreign research reactor spent nuclear fuel were calculated based on the methodology and assumptions described in Appendix F, Section F.6. The radiological consequences of air emissions from the operation of a new dry storage facility at the Hanford Site are discussed in Section F.4.3.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.

F.4.3.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 15,000 million l (3,960 million gal) for the Hanford Site, these amounts represent no more than a 0.04 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Hanford Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Hanford Site. The impact on water quality during operations would also be small. Existing water treatment facilities at the Hanford Site could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Hanford Site would still be well within the design capacities of treatment systems at the Hanford Site. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.3.2.1.8 Ecology

Terrestrial Resources: Vegetation within construction areas would be destroyed during land-clearing

activities. Plant species that are dominant on the 200 Area Plateau include: big sagebrush, cheatgrass, and Sanberg's bluegrass. Total area destroyed would amount to about less than one percent of this community on the Hanford Site. Although the plant communities to be disturbed are well represented on Hanford

Very small quantities of radionuclides would be released to the atmosphere during dry storage facility operations. No organisms studied to date are reported to be more sensitive than man to radiation. Therefore, the effects of these releases on terrestrial organisms are expected to be minor (Bergsman et al., 1994).

Any impacts to the vegetation and animal communities would be mitigated by minimizing the amount of land disturbed during construction, employing soil erosion control measures during construction activities, and revegetating disturbed areas with native species. These mitigation measures would limit the amount of direct and indirect disturbance to the construction area and surrounding habitats and would speed the recovery process for disturbed lands (Bergsman et al., 1994).

Operational impacts on terrestrial biotic resources would include exposure of plants and animals to small amounts of radionuclides released during operation of the new dry storage facility. The levels of radionuclide exposure would be below those levels that produce adverse effects (Bergsman et al., 1994).

Wetlands: There are no wetlands on or near either candidate storage site (Bergsman et al., 1994).

Threatened and Endangered Species: Construction and operation of the new dry storage facility would remove 3.7 ha (9 acres) of relatively pristine big sagebrush/cheatgrass/Sanberg's bluegrass habitat. This sagebrush habitat is considered priority habitat by the State of Washington because of its relative scarcity in the State and its use as nesting/breeding habitat by loggerhead shrikes, sage sparrows, sage thrashers, burrowing owls, pygmy rabbits, and sagebrush voles (Bergsman et al., 1994).

Loggerhead shrikes, listed as a Federal candidate (Category 2) and State candidate species, forage on the proposed spent nuclear fuel site and are relatively common on the Hanford Site. This species is sagebrush-dependent, as it is known to select primarily tall big sagebrush as nest sites. Construction of the new dry storage facility would remove big sagebrush habitat which would preclude loggerhead shrikes from nesting there. Foreign research reactor spent nuclear fuel site development would also be expected to reduce the value of the site as foraging habitat for shrikes known to nest in adjacent areas (Bergsman et al., 1994).

Sage sparrows and sage thrashers, both State candidate species, occur in mature sagebrush/bunchgrass habitat at the Hanford Site. The sage sparrow was observed on the proposed site in a survey during spring 1994. These species are known to nest primarily in sagebrush. Construction of the new dry storage facility would preclude both of these species nesting there and reduce the site's suitability as foraging habitat for these species (Bergsman et al., 1994).

Dry storage facility construction is not expected to substantially decrease Hanford Site population of loggerhead shrikes, sage sparrows, or sage thrashers because similar sagebrush habitat is still relatively common on the Hanford Site. However, the cumulative effects of constructing the new dry storage facility, in addition to future developments that further reduce sagebrush habitat (causing further fragmentation of nesting habitat), could negatively affect the long-term viability of populations of these species on the Hanford Site (Bergsman et al., 1994).

Burrowing owls, a State candidate species, are relatively common on Hanford Site and nest in abandoned ground squirrel burrows on the 200 Area Plateau. Construction would remove sagebrush and disturb soil, displacing ground squirrels and thus reducing the suitability of the area for nesting by burrowing owls, and would also displace small mammals, which constitute a portion of the prey base for this species. Dry storage facility construction would not be expected to negatively impact the viability of the population of burrowing owls on the Hanford Site, as their use of ground squirrel burrows as nests is not limited to burrows in big sagebrush habitat (Bergsman et al., 1994).

Pygmy rabbits, a Federal candidate (Category 2) and State-listed threatened species, are known to utilize tall clumps of big sagebrush habitat throughout most of their range. However, this species has not recently been observed on the Hanford Site. Construction of the new dry storage facility would therefore reduce the potential for this species' occurrence by removing habitat suitable for its use (Bergsman et al., 1994).

Sagebrush voles, a State minor species, are common on Hanford Site and select burrow sites near sagebrush; however, this species is common only at higher elevations around the Hanford Site. Construction of the new dry storage facility would remove sagebrush habitat, precluding sagebrush voles from utilizing the site. However, construction would not affect the overall viability of sagebrush vole populations on Hanford Site because the majority of the population is found on the Fitzner/Eberhardt Arid Lands Ecology Preserve (Bergsman et al., 1994).

The closest known nests of ferruginous hawks, a Federal candidate (Category 2) and State threatened [illegible]

the 200 Area Plateau. The potential site comprises a portion of the foraging range of these hawks. Construction of the new dry storage facility is not expected to disrupt the nesting activities of these

F.4.3.2.1.10 Traffic and Transportation

Construction materials, wastes, and excavated materials would be transported both onsite and offsite. These activities would result in increases in operation of personal-use vehicles by commuting construction workers, commercial truck traffic, and in traffic associated with the daily operations of the Hanford Site. Again, traffic congestion would not be a significant problem.

Traffic congestion, although moderate at shift changes, would not be noticeably worse due to this level of construction effort.

F.4.3.2.1.11 Occupational and Public Health and Safety

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Hanford Site would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-57 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Hanford Site. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-57 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Hanford Site (Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• FMEF (dry storage)	0.00020	1.0×10^{-10}	0.011	0.0000055
• New Dry Storage Facility	0.00025	1.3×10^{-10}	0.015	0.0000075
Storage at:				
• FMEF (dry storage)	0	0	0	0
• New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 3A involves the receipt and unloading of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or Savannah River Site and 193 shipments directly from ports into a dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-58 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Hanford Site. The

Table F-58 Handling-Related Impacts to Workers at the Hanford Site (New Dry Storage)

	Worker Population Dose (Person-rem)	Worker Population Risk (LCF)
	FMEF/New Dry Storage	FMEF/New Dry Storage
Phase 2	266/113[a]	0.11/0.05[a]

[a] *The two numbers represent the cask/vault designs respectively.*

administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

F.4.3.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Hanford Site would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-59. These requirements represent a small percent of current requirements for the Hanford Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Hanford Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-59 Annual Utility and Energy Requirements for New Dry Storage at the Hanford Site

Commodity	Baseline Site Usage	Dry Storage Usage	Percent Increase
Electricity (MW-hr/yr)	340,000	800 - 1,000	0.3 percent
Fuel (l/yr)	83,000,000	0	0 percent
Water (l/yr)	15,000,000,000	1,590,000[a] 400,000[b]	0.01 percent[a] 0.003 percent[b]

[a] *During receipt and handling.*

[b] *During storage.*

F.4.3.2.1.13 Waste Management

Construction of a new dry storage facility at the Hanford Site would generate 1,800 m^3 (2,340 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-60. These quantities, represent a very small percent increase above current levels at the Hanford Site. Existing waste management storage and disposal activities at Hanford Site could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Hanford Site waste management capacities would be minimal.

Table F-60 Annual Waste Generated for New Dry Storage at the Hanford Site

Waste Form	Baseline Site Generation	Dry Storage Generation	Percent Increase
High-Level (m^3/yr)	240	none	0 percent
Transuranic (m^3/yr)	170	none	0 percent
Solid Low-Level (m^3/yr)	20,000	22[a] 1[b]	0.11 percent[a] 0.005 percent[b]
Wastewater (l/yr)	210,000,000	1,590,000[a] 400,000[b]	0.75 percent[a] 0.2 percent[b]

[a] *During receipt and handling.*

[b] *During storage.*

F.4.3.2.2 Wet Storage

Analysis option 3B involves long-term wet storage of foreign research reactor spent nuclear fuel at the Hanford Site. This storage option would require the construction of a new wet storage facility.

F.4.3.2.2.1 Land Use

A new wet storage facility would be located on the 200 Area Plateau or in conjunction with the WNP-4 Spray Cooling Pond. These areas have already been developed for industrial use. Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land at either area. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at either area would significantly impact land use patterns on the Hanford Site.

F.4.3.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be $449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $112.2 million. This represents approximately 8.7 percent of the estimated FY 1995 total expenditures for the Hanford Site (1,288 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage. These costs represent about 1.8 percent and 0.3 percent of FY 1995 total expenditures for the Hanford Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at Hanford Site of approximately 18,500 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be small to both the region of influence and the Hanford Site.

F.4.3.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.3.2.1.3).

The potential for impacting cultural resources would be even less for the WNP-4 Spray Pond because the structures are all essentially in place. Thus, there would be no opportunity for discovery of cultural resources during construction.

F.4.3.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.3.2.1.4).

F.4.3.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.3.2.1.5).

F.4.3.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of 2.8 ha (7 acres) of land. In comparison, 3.7 ha (9 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction (Section F.4.3.2.1.6).

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological-Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3 micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operation in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs as an indicator nuclide. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilation, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE 5480.1B (DOE, 1989b) for both onsite and offsite concentrations.

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the IFSF at the Idaho National Engineering Laboratory. The annual air emission for the IFSF was designed to result in ground-level concentrations of less than 0.003 percent of DOE 5480.1B limits for uncontrolled areas.

Radiological emissions from the operation of the new wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading and storage are provided in Section F.6.6.1.

... during construction of a new wet storage facility

F.4.3.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (0.72 million gal), respectively. With an annual average water usage of approximately 15,000 million l (3,960 million gal) for the Hanford Site, these amounts represent an increase of about 0.02 percent and less than 0.005 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Hanford Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Hanford Site. ... during operations would also be negligible. Existing water treatment facilities resulting doses are discussed in Section F.5 of this appendix. Table F-61 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Hanford Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-61 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Hanford Site (Implementation Alternative 5 of Management Alternative 1)

Facility	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Receipt/Unloading at:				
• WNP-4 Spray Pond	0.00022	1.1×10^{-10}	0.0058	0.0000029
• New Wet Storage Facility	0.00020	1.0×10^{-10}	0.012	0.000006
Storage at:				
• WNP-4 Spray Pond	5.9×10^{-10}	3.0×10^{-16}	1.6×10^{-8}	8.0×10^{-12}
• New Wet Storage Facility	8.8×10^{-10}	4.4×10^{-16}	6.9×10^{-8}	3.5×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 3B involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from Idaho National Engineering Laboratory and/or Savannah River Site and 193 shipments directly from the ports into a wet storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-62 presents the population dose that would be received by the members of the working crew and the associated risks if that working crew handled the total number of transportation casks at the Hanford Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

F.4.3.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at the Hanford Site would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-63. These requirements represent a small percent of current requirements for the Hanford Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Hanford Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-62 Handling-Related Impacts to Workers at the Hanford Site

[illegible]

Table F-65 presents frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-65 Frequency and Consequences of Accidents at the Hanford Site

	Frequency (per year)	*Consequences*			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
Dry Storage Accidents[a]					
• Spent Fuel Assembly Breach	0.16	3.0	0.57	42	50
• Dropped Fuel Cask	0.0001	0.26	0.0085	3.0	0.22
• Aircraft Crash w\Fire[b]	NA	NA	NA	NA	NA
Dry Storage Accidents at FMEF					
• Spent Fuel Assembly Breach	0.16	4.7	2.1	46	0.99
• Dropped Fuel Cask	0.0001	0.2	0.032	3.2	0.0049
• Aircraft Crash w\Fire[b]	NA	NA	NA	NA	NA

NA = Not Applicable

[a] *New Dry Storage Facility*

[b] *Aircraft Crash accidents are not applicable to Hanford Site because their frequency of occurrence is less than one every ten million years.*

Multiplying the frequency of each accident times its

Table F-67 Frequency and Consequences of Accidents at the Hanford Site (Implementation Alternative 5 of Management Alternative 1)

		Consequences			

In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Hanford Site, the radiological accident with the highest MEI dose is the fuel assembly breach at a dry storage facility located at the FMEF (Table F-65). For this accident, the MEI dose would be 3.9 mrem, which is less than the 100 mrem limit used in this analysis. Therefore, no secondary impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies from radiological accidents involving foreign research reactor spent nuclear fuel storage are expected at the Hanford Site.

F.4.3.4 Cumulative Impacts at the Hanford Site

This section presents the cumulative impacts of the proposed action, potential impacts of other major contemplated DOE actions, and current activities at the Hanford Site. A major portion of the presentation is based on information included in the Programmatic SNF&INEL Final EIS (DOE, 1995g), the Management of Spent Nuclear Fuel from the K Basins Draft EIS (DOE, 1995d) and the Safe Interim Storage of Hanford Tank Wastes Final EIS (DOE, 1995c).

Table F-69 summarizes the cumulative impacts for land use, socioeconomics, air quality, occupational and public health and safety, energy and water consumption and waste generation. The table also presents the contributions from the storage of foreign research reactor spent nuclear fuel on the cumulative impacts at the Hanford Site. For the purposes of this analysis, both the contributions from management of foreign research reactor spent nuclear fuel and the cumulative impacts were maximized by selecting the Centralization Alternative of the Programmatic SNF&INEL Final EIS at the Hanford Site.

As shown in Table F-69, the contribution from management of foreign research reactor spent nuclear fuel to the cumulative impacts at the Hanford Site would be minimal. It is concluded, therefore, that the implementation of any of the alternatives (including the Centralization Alternative) for the DOE spent nuclear fuel management program would not be expected to significantly contribute to cumulative impacts.

F.4.3.5 Unavoidable Adverse Environmental Impacts

Unavoidable impacts associated with foreign research reactor spent nuclear fuel management activities would derive principally from construction activities needed for new storage facilities. There would be displacement of some animals from the construction site and the destruction of plant life within the area scoped for construction [up to 4 ha (10 acres)]. Criteria pollutants and radionuclides, would also be released in up to permitted quantities. Traffic congestion and noise would be expected to increase by a few percent during the construction of major facilities.

F.4.3.6 Irreversible and Irretrievable Commitments of Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel would involve materials that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. The construction and operation of facilities for foreign research reactor spent nuclear fuel facilities at the Hanford Site would consume irretrievable amounts of electrical energy, fuel, concrete, sand, and gravel. Other resources used in construction would probably not be recoverable. These would include

Table F-69 Cumulative Impacts at the Hanford Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Land Use (acres)	9	84,343[b]	84,352
Socioeconomics (persons)	190[c]/30[d]	3300/1220[e]	3,490[c]/1250[d]
Air Quality (nonradiological)	See Table F-56	NA	(f)
Occupational and Public Health and Safety			
• MEI Dose (rem/yr)	2.5×10^{-7}	0.0000036	0.0000036
LCF (per year)	1.3×10^{-10}	1.5×10^{-9}	1.5×10^{-9}
• Population dose (person-rem/yr)	0.015	0.22	0.235
LCF (per year)	0.0000075	0.00011	0.00011
• Worker Collective dose (person-rem/yr)	8.9[g]	116.5	125.4
LCF (per year)	0.0035	0.0466	0.05
Energy and Water Consumption			
• Electricity (MW-hr/yr)	1,000	495,600	496,600

pollution prevention technologies. Program components include waste minimization, source reduction and recycling, and procurement practices that preferentially procure products made from recycled materials. The pollution prevention program at the Hanford Site is being

[redacted]

Traffic and Transportation: At sites with increasing traffic concerns, DOE would encourage use of high-occupancy vehicles (such as vans or buses), implementing carpooling and ride-sharing programs, and staggering work hours to reduce peak traffic.

Occupational and Public Health and Safety: Although no radiological impacts on workers or the public were evident from the evaluation of incident-free foreign research reactor spent nuclear fuel activities at Hanford, further improvement in controls to protect both workers and the general public is a continuing activity. The "as low as reasonably achievable" principle would be used for controlling radiation exposure and exposure to hazardous/toxic substances. The Hanford Site would continue to refine its current emergency planning, emergency preparedness, and emergency response programs in place to protect both workers and the public (DOE, 1995g).

Site Utilities and Support Services: No mitigation measures beyond those identified for ground disturbance activities associated with bringing power and water to the foreign research reactor spent nuclear fuel site would appear necessary. In those cases, use of standard dust suppression techniques and revegetation of disturbed areas would mitigate ground disturbance impacts.

Accidents: The Hanford Site maintains an emergency response center and has emergency action plans and equipment to respond to accidents and other emergencies. These plans include training of workers, local emergency response agencies (such as fire departments) and the public communication systems and protocols, readiness drills, and mutual aid agreements. The plans would be updated to include consideration of new foreign research reactor spent nuclear fuel facilities and activities. Design of new facilities to current seismic and other facility protection standards would reduce the potential for accidents, and implementation of emergency response plans would substantially mitigate the potential for impacts in the event of an accident.

F.4.4 Oak Ridge Reservation

If the Oak Ridge Reservation site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period required for the Oak Ridge Reservation to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, this period (Phase 1) is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2), the Oak Ridge Reservation would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Oak Ridge Reservation until ultimate disposition.

The amount of spent nuclear fuel that would be received and managed at the Oak Ridge Reservation under Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, in Phase 2, the Oak Ridge Reservation could receive the aluminum-based foreign research reactor spent nuclear fuel managed at the Savannah River site during Phase 1, Eastern foreign research reactor spent nuclear fuel under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative.

As a Phase 2 site, the Oak Ridge Reservation would receive and manage foreign research reactor spent nuclear fuel at a new dry storage facility to be constructed on the West Bear Creek Valley Site. The location is preferred among the four locations considered in a siting study performed for spent nuclear fuel management (MMES, 1994). Description of the new dry storage facility is provided in Section 2.6.5.1.1.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation is based on the above considerations. The analysis options selected do not represent all possible combinations but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

4A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Oak Ridge Reservation where it would be managed at a new dry storage facility until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of this analysis, the total amount of spent nuclear fuel that would be managed in the new dry storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States discussed in Section 2.2.2 introduce additional analysis options that could be considered for the Oak Ridge Reservation as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Oak Ridge Reservation would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and manage it in facilities sized for this amount of spent fuel. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 4A above.

the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in option 4A above.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States as the foreign research reactor operators would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of fuel, in this case, cannot be quantified, however, the upper limit, as considered under analysis option 4A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management impacts at the Oak Ridge Reservation.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Oak Ridge Reservation for Phase 2 until ultimate disposition. For this implementation alternative an analysis option 4B, which is similar to 4A, is considered as follows:

4B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Oak Ridge Reservation where it would be managed at a new wet storage facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel to be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Oak Ridge Reservation would not be considered as a site for chemical separation.

Under Management Alternative 3 (Hybrid Alternative) the Oak Ridge Reservation is not considered.

F.4.4.1 Existing Facilities

There are no existing facilities for storing foreign research reactor spent nuclear fuel at Oak Ridge Reservation. Consequently, all potential environmental consequences from foreign research reactor spent nuclear fuel storage are related to new facility construction and operation.

F.4.4.2 New Facilities (Phase 2)

Analysis options 4A and 4B involve the use of new facilities as discussed above. The environmental impacts analyzed relate to the construction and operation of these facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities, and energy; and waste management.

F.4.4.2.1 Dry Storage

Analysis option 4A involves long-term storage of foreign research reactor spent nuclear fuel at Oak Ridge Reservation. This analysis option would require the construction of a new dry storage facility. The analysis option encompasses both the dry storage vault design and the dry cask design as described in Section 2.6.5 and earlier in this appendix. There are no environmental impact parameters that would discriminate between the two designs. For the purpose of this analysis, the impacts from the larger dry vault design are presented.

F.4.4.2.1.1 Land Use

A new dry storage facility would be located in a 36-ha (90-acre) area in the eastern portion of West Bear Creek Valley. The majority of the land in this area can be characterized as vacant, unused, and ready for development. Use of West Bear Creek Valley for foreign research reactor spent nuclear fuel storage would be consistent with existing land use plans, which designate this area for general use (MMES, 1994). Construction activities, including laydown areas, would disturb 16 ha (40 acres) of land. This represents about 44 percent of the space designated for foreign research reactor spent nuclear fuel storage; however, this represents only about 0.1 percent of the entire Oak Ridge Reservation. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at any of the areas would significantly impact land use patterns on the Oak Ridge Reservation.

F.4.4.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be $370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents approximately 7.8 percent of the estimated FY 1995 total expenditures for the Oak Ridge Reservation (1,174 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new dry storage facility are estimated to be $15.6 million for receipt and handling and $0.6 million for storage. These costs represent approximately 1.3 percent and 0.05 percent of FY 1995 total expenditures for the Oak Ridge Reservation. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from direct and secondary construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Oak Ridge Reservation of approximately 17,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with receipt and storage operations is estimated to be 30 persons. Upon completion of these activities, direct employment is expected to decrease to eight persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and Oak Ridge Reservation.

F.4.4.2.1.3 Cultural Resources

There are no known historical, archaeological, paleontological, or Native American traditional sites in or around the potential storage site. No impacts to cultural resources are expected from ground disturbance, noise, or air emissions during construction or operation of the facility. Consultation with the Tennessee State Historic Preservation Office prior to project implementation is required by Section 106 of the National Historic Preservation Act of 1966. The State Historic Preservation Office may recommend further studies of the potential storage site to verify that no archaeological areas would be disturbed by construction activities (DOE, 1995g).

F.4.4.2.1.4 Aesthetic and Scenic Resources

Construction and operation of a new dry storage facility for foreign research reactor spent nuclear fuel

would have similar impact on aesthetic and scenic resources at the Oak Ridge Reservation as the construction of spent nuclear fuel facilities under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g). The spent nuclear fuel facilities associated with the Centralization Alternative would consist of a series of industrial buildings set within a 36-ha (90-acre) site. The maximum height of the buildings on the site would not exceed 12.8 m (42 ft) above ground level, or two to three stories. Since the buildings would be set into the south face of Pine Ridge, between Pine

F.4.4.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts associated with construction include generation of fugitive dust (particulate matter) and smoke from earth moving and clearing operations and emissions from construction equipment. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed surfaces.

Construction of this facility would require the clearing of approximately 16 ha (40 acres) of land. However, the overall construction impacts to the ambient air quality of the region should be minimal due to the short duration (3 months to 6 years) of the project. Emissions of sulfur dioxide, nitrogen dioxide, and carbon monoxide are assumed to result entirely from diesel exhaust during the construction process. Respirable particulate matter (e.g., PM_{10}) is assumed to be 64 percent of the total suspended particulates estimated for the construction effort. Additionally, wetting controls are assumed to reduce this amount by 50 percent, which is a very conservative estimate.

Table F-70 presents the air quality impacts associated with the construction of a new dry storage facility at the Oak Ridge Reservation. Additionally, this table shows that the ambient impacts would be minimal and compliance with existing Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population. The estimated impacts from construction activities were generated using the Environmental Protection Agency regulatory-approved Industrial Source Complex Short-Term Model, Version 2.0, in conjunction with onsite meteorological data from 1991.

Table F-70 Estimated Maximum Concentrations of Criteria Pollutants at the Oak Ridge Reservation Attributable to New Dry Storage Construction

Pollutant	Averaging Time	Ambient Standard[a]	Baseline Concentration[b]	Construction Activities
Oak Ridge Reservation Boundary ($\mu g/m^3$)[c]				
• Particulate Matter (PM_{10})[c]	24-hr	150	84.9	0.5450
	Annual	50	0.43	0.0144
• Carbon Monoxide	1-hr	40,000	2,748.0	26.756
	8-hr	10,000	2,290.8	3.345
• Sulfur Dioxide	3-hr	1,300	170.3	2.356
	24-hr	365	55.2	0.345
	Annual	80	1.1	0.006
• Nitrogen Oxide	Annual	100	2.1	0.098

[a] *64 percent of total suspended particulates is considered to be respirable particulate matter (e.g., PM_{10}) for the construction activities. The standard refers to the actual PM_{10} standard.*

[b] *Source: DOE, 1995g*

[c] *To convert to $\mu g/ft^3$, multiply by 0.0283*

Nonradiological emissions are not expected during operation of a new dry storage facility.

Radiological Emissions: No radiological emissions from construction of a new dry storage facility for foreign research reactor spent nuclear fuel are expected. Based on fuel drying and storage operations conducted at Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively higher contamination) would help provide contamination control. High-efficiency particulate air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides released during the cask opening operation depends on the following parameters: (1) the number of spent nuclear fuel clad defects; (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask, and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a gaseous state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

For this analysis, radiological emissions from the operation of a new dry storage facility were calculated based on the methodology and assumptions described in Appendix F, Section F.6. The radiological consequences of air emissions from the operation of the dry storage facilities at the Oak Ridge Reservation are discussed in Section F.4.4.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.4.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 3,060 million l (808 million gal) for the Oak Ridge Reservation, these amounts represent no more than a 0.07 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Oak Ridge Reservation.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Oak Ridge Reservation. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Oak Ridge Reservation could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Oak Ridge

Reservation would still be well within the design capacities of treatment systems at the Oak Ridge Reservation. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.4.2.1.8 Ecology

Terrestrial Resources: Radiation doses received by terrestrial biota from foreign research reactor spent nuclear fuel activities would be expected to be similar to those received by man. Although guidelines have not been established for acceptance limits for radiation exposure to species other than man, it is generally agreed that the limits established for humans are also conservative for other species. Evidence indicates that no other living organisms have been identified that are likely to be significantly more radiosensitive than man. Thus, so long as exposure limits protective of man were not exceeded, no significant radiological impact on populations of biota would be expected as a result of foreign research reactor spent nuclear fuel activities at the West Bear Creek Site (DOE, 1995g).

Under the Centralization Alternative, construction of the potential spent nuclear fuel management facility would result in the disturbance of approximately 36 ha (90 acres) [16 ha (40 acres) if foreign research reactor spent nuclear fuel is considered in isolation], or less than 1 percent of the Oak Ridge Reservation. It is assumed that the area to be disturbed includes construction laydown areas, grading, and new buildings, and that the access road or other rights-of-ways have not been included in the total area to be disturbed. Vegetation within the area of the potential site for the spent nuclear fuel management facility would be destroyed during land clearing activities, but may be mitigated by revegetating with native species where possible. Vegetation cover in this area is predominantly oak-hickory forest or pine-hardwood forest. Both forest types are common on the Oak Ridge Reservation and within the region (DOE, 1995g).

Construction of a new dry storage facility would have some adverse effects on animal populations. Less mobile animals, such as amphibians, reptiles, and small mammals, within the project area would be destroyed during land-clearing activities. Larger mammals and birds in construction and adjacent areas would be disturbed by construction activities and would move to nearby suitable habitat. The long-term survival of these animals would depend on whether the area to which they moved was at or below its carrying capacity. Areas that would be revegetated upon completion of construction would be of minimal value to most wildlife, but might be repopulated by more tolerant species (DOE, 1995g).

The Migratory Bird Treaty Act is primarily concerned with the destruction of migratory birds, as well as their eggs and nests. It could be necessary to survey construction sites for the nests of migratory birds prior to construction and/or avoid clearing operations during the breeding season (DOE, 1995g).

Activities associated with operation, such as noise, increased human presence and traffic, and night lighting could affect wildlife living immediately adjacent to the storage site. While these disturbances could cause some sensitive species to move from the area, most animals should be able to adjust (DOE, 1995g).

Wetlands: Construction of a new dry storage facility would likely displace the forested wetlands adjacent to tributaries of Grassy Creek flowing through the potential site. This unavoidable displacement of wetlands would be accomplished in accordance with U.S. Army Corps of Engineers and Tennessee Water Quality Control Administration requirements. The potential also exists to disturb wetlands further downstream through erosion and sedimentation. Such impacts would be controlled through implementation of a soil erosion and sediment control plan. Construction-related discharges to Grassy

Creek would be relatively low and have negligible impacts to wetlands associated with the creek. No impacts to wetlands are anticipated during facility operations (DOE, 1995g).

Construction of a new dry storage facility would require the rechanneling of tributaries to Grassy Creek that cross the potential site, thus causing the loss of this aquatic habitat. In addition, soil erosion due to construction could cause water quality changes (primarily sediment loading) to Grassy Creek and its tributaries. These impacts could be minimized by implementation of soil erosion and sediment control measures. No operational impacts to aquatic resources are anticipated. It is assumed that the potential project would have a water retention pond within the security fence that might provide minimal habitat for amphibians in the area.

Threatened and Endangered Species: No Federally-listed species are expected to be affected. Site surveys would be required to verify the presence of State-listed or other special status species. Land clearing activities could destroy protected plant species, such as purple fringeless orchid and pink lady's-slippers, that may occur within the site. State-listed species including the Cooper's, sharp-shinned, and red-shouldered hawks, the barn owl, and the black vulture, which potentially occur in the area, could be impacted by project activities. Approximately 16 ha (40 acres) of potential nesting and foraging habitat would be lost as a result of construction activities. Because this type of habitat is abundant in the area, the loss would not be expected to affect the viability of populations of these species. However, appropriate steps would be taken to prevent nest disturbance. DOE would consult with the Tennessee Department of Environment and Conservation as appropriate to avoid or mitigate imminent impacts to State-listed species (DOE, 1995g). DOE would also consult with the U.S. Fish & Wildlife Service regarding threatened and endangered species for the proposed construction sites of foreign research reactor spent nuclear fuel storage facilities at the Oak Ridge Reservation. Impacts to threatened and endangered species are not anticipated.

F.4.4.2.1.9 Noise

Noises generated on the Oak Ridge Reservation do not propagate offsite at levels that impact the general population. Thus, the Oak Ridge Reservation noise impacts for both the Centralization and Regionalization by Fuel Type and Geography Alternatives would be those resulting from transportation of personnel and materials to and from the site that affect nearby communities, and those resulting from onsite sources that may affect some wildlife near these sources (DOE, 1995g).

The transportation noises are a function of the size of the work force (e.g., an increased work force would result in increased employee traffic and corresponding increases in deliveries by construction crews). Such noise and activity associated with construction would be expected to have short-term effects on most wildlife. Under the Centralization Alternative, the projected Oak Ridge Reservation work force would increase by about nine percent in the years 2000 to 2002 during peak construction, and decrease thereafter. There would be a corresponding increase in private vehicle and truck trips to the site. The day-night average sound level at 15 m (50 ft) from the roads that provide access to the Oak Ridge Reservation would be expected to increase by less than 1 decibel. No change is expected in the community reaction to noise along these routes. No mitigation of traffic noise impacts is proposed (DOE, 1995g).

F.4.4.2.1.10 Traffic and Transportation

Construction and operation of a new dry storage facility would involve a small increase in the number of employees commuting to the Oak Ridge Reservation and transportation of foreign research reactor spent nuclear fuel and hazardous chemicals within the site.

The maximum reasonably foreseeable scenario for construction and operation traffic occurs under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS. This would occur in 2001, when there would be about 4,200 full time employees and about 409,500 people in the region of influence. Construction and operation employees would contribute little to the future traffic because they represent such a small percentage of the region of influence population growth (DOE, 1995g). This conclusion would also be valid for a new dry storage facility for foreign research reactor spent nuclear fuel.

F.4.4.2.1.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Oak Ridge Reservation would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-71 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Oak Ridge Reservation. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-71 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Oak Ridge Reservation (New Dry Storage)

Facility	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person rem/yr)	Population Risk (LCF/yr)
Receipt/Unloading at:				
• New Dry Storage Facility	0.089	4.5×10^{-8}	0.085	0.000043
Storage at:				
• New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 4A involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and 193 shipments directly from ports into a dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-72 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Oak Ridge Reservation.

The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the

Table F-74. These quantities represent a very small percent increase above current levels at the Oak Ridge Reservation. Existing waste management storage and disposal activities at Oak Ridge Reservation could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Oak Ridge Reservation waste management capacities would be minimal.

Table F-74 Annual Waste Generated for New Dry Storage at the Oak Ridge Reservation

Waste Form	Baseline Site Generation	Dry Storage Generation	Percent Increase
High-Level (m^3/yr)	0	0	0 percent
Transuranic (m^3/yr)	16	0	0 percent
Solid Low-Level (m^3/yr)	6,902	22[a] 1[b]	0.32 percent[a] 0.01 percent[b]
Wastewater (l/yr)	754,000,000	1,590,000[a] 400,000[b]	0.21 percent[a] 0.05 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.4.2.2 Wet Storage

Analysis option 4B involves long-term wet storage of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation. This storage option would require the construction of a new wet storage facility.

F.4.4.2.2.1 Land Use

A new wet storage facility would be located in a 36-ha (90-acres) area in the eastern portion of West Bear Creek Valley. The majority of the land in this area can be characterized as vacant, unused, and ready for development. Use of West Bear Creek Valley for foreign research reactor spent nuclear fuel storage would be consistent with existing land use plans, which designate this area for general use. Construction activities, including laydown areas, would disturb 16 ha (40 acres) of land. This represents about 44 percent of the space designated for foreign research reactor spent nuclear fuel storage; however, this represents only about 0.1 percent of the entire Oak Ridge Reservation. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at any of the areas would significantly impact land use patterns on Oak Ridge Reservation.

F.4.4.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be $449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $112.2 million. This represents approximately 8.2 percent of the estimated FY 1995 total expenditures for the Oak Ridge Reservation (1,174 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage. These costs represent about 2 percent and 0.3 percent of FY 1995 total expenditures for the Oak Ridge Reservation. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Oak Ridge Reservation of approximately 17,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Oak Ridge Reservation.

F.4.4.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.4.2.1.3).

F.4.4.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.4.2.1.4).

F.4.4.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.4.2.1.5).

F.4.4.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 3 ha (7 acres) of land. In comparison, approximately 4 ha (10 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction (Section F.4.4.2.1.6).

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double banked high efficiency particulate air filters prior to release to the environment through an exhaust air system. The high efficiency particulate air filters would have a minimum efficiency of 99.97 percent for 0.3 micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilating, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite concentrations (DOE, 1989b). Air emissions from the wet storage facility are expected to be similar to the air emissions from the CPP-603 at the Idaho National Engineering Laboratory. The annual air emission for the CPP-603 was designed to result in ground-level concentrations of less than 0.003 percent of DOE 5480.1B limits for uncontrolled areas. Radiological emissions from the operation of the wet storage facility were calculated based on the methodology and assumptions used in Section F.6.

F.4.4.2.2.7 Water Resources

The annual water usage during construction and operations of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (720,000 gal), respectively. With an annual average water usage of approximately 3,060 million l (808 million gal) for the Oak Ridge Reservation, these amounts represent an increase of about 0.06 percent and 0.09 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Oak Ridge Reservation.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Oak Ridge Reservation. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Oak Ridge Reservation could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Oak Ridge Reservation would still be well within the design capacities of treatment systems at the Oak Ridge Reservation. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.4.2.2.8 Ecology

Impacts to ecology would be the same as for new dry storage (Section F.4.4.2.1.8).

F.4.4.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.4.2.1.9).

F.4.4.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.4.2.1.10).

F.4.4.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Oak Ridge Reservation would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from routine airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. Table F-75 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Oak Ridge Reservation. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-75 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

Table F-77. These requirements represent a small percent of current requirements for the Oak Ridge Reservation. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Oak Ridge Reservation is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-77 Annual Utility and Energy Requirements for Wet Storage at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

Commodity	Baseline Site Usage	Wet Storage Usage	Percent Increase
Electricity (MW-hr/yr)	335,800	800 - 1,000	0.15 percent
Fuel (l/yr)	3,600[a]	0	0 percent
Water (l/yr)	3,060,000,000	2,700,000[b] 1,500,000[c]	0.09 percent 0.05 percent

[a] *Decatherms/yr of natural gas*

[b] *During receipt and handling*

[c] *During storage*

F.4.4.2.2.13 Waste Management

Construction of a new wet storage facility at the Oak Ridge Reservation would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-78. These quantities represent a very small percentage increase above current levels at the Oak Ridge Reservation. Existing waste management storage and disposal activities at the Oak Ridge Reservation could accommodate the waste generated by a new wet storage facility. Therefore, the impact of this waste on existing the Oak Ridge Reservation waste management capacities would be minimal.

Table F-78 Annual Waste Generated for Wet Storage at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

Waste Form	[illegible]	[illegible]	[illegible]

locations and the same accidents at any of the sites evaluated. Information concerning radiation doses to individuals and the general population are the same as set forth in Section F.4.1.3.

Table F-79 presents the frequencies and the consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-79 Frequency and Consequences of Accidents at the Oak Ridge Reservation

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
Dry Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	22	42	55	140
• Dropped Spent Nuclear Fuel Cask	0.0001	1.4	0.18	15	0.61
• Aircraft Crash w\Fire	0.000001	2300	180	2900	610

[a] *New Dry Storage Facility*

Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Oak Ridge Reservation. These annual risks are multiplied by the maximum duration of this implementation alternative at each site to obtain conservative estimates of risks for the Oak Ridge Reservation. These risk estimates are presented in Table F-80.

Table F-80 Annual Risks of Accidents at the Oak Ridge Reservation

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Dry Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	0.0000018	0.0000034	0.0044	0.0000088
• Dropped Spent Nuclear Fuel Cask	7.0×10^{-11}	9.0×10^{-12}	7.5×10^{-7}	2.4×10^{-11}
• Aircraft Crash w/Fire	1.2×10^{-9}	9.0×10^{-11}	0.0000015	2.4×10^{-10}

[a] *New Dry Storage Facility*

Table F-81 presents the frequency and consequences of the accidents analyzed for each site for wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Oak Ridge Reservation. These annual risks are multiplied by the maximum duration of this implementation alternative at each site to obtain conservative estimates of risks at the Oak Ridge Reservation. Table F-82 presents the risk estimates from this implementation alternative.

F.4.4.3.1 Secondary Impact of Radiological Accidents at the Oak Ridge Reservation

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the calculations presented in Section F.4.4.3. Radiological

Table F-81 Frequency and Consequences of Accidents at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

	Frequency	Consequences			
				Population	

include activities associated with the waste management at the site, storage and disposition of weapons-usable fissile materials, and stockpile stewardship and management program.

Tables F-83 and F-83A summarize the cumulative impacts for land use, socioeconomics, air quality, occupational and public health and safety, energy and water consumption, and waste generation at the site. Table F-83 also presents the contribution from the storage of foreign research reactor spent nuclear fuel on the cumulative impacts at the Oak Ridge Reservation. For the purposes of this analysis, both the contributions from management of foreign research reactor spent nuclear fuel and the cumulative impacts were maximized by selecting the Centralization Alternative of the Programmatic SNF&INEL Final EIS at the Oak Ridge Reservation.

As shown in Table F-83, the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts (under the Centralization Alternative) at the Oak Ridge Reservation would be minimal. The Programmatic SNF&INEL Final EIS concludes that the implementation of any of the alternatives (including the Centralization Alternative) for the DOE spent nuclear fuel management program would not be expected to significantly contribute to cumulative impacts (DOE, 1995g). This conclusion is also valid for the implementation of any of the alternatives considered in this EIS for storage of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation.

F.4.4.5 Unavoidable Adverse Environmental Impacts

Construction of the potential foreign research reactor spent nuclear fuel storage facilities would require the disturbance of approximately 16 ha (40 acres) of mostly forested undeveloped land. Although this represents less than one percent of the undeveloped land on the Oak Ridge Reservation, it would eliminate potential foraging and nesting habitat and would destroy plant species in the area. It would also require the dedication of a reasonably level land parcel that could otherwise accommodate other construction projects.

F.4.4.6 Irreversible and Irretrievable Commitments of Resources

Construction and operation of new foreign research reactor spent nuclear fuel storage facilities would require commitments of electrical energy, fuel, concrete, steel, sand, gravel and miscellaneous chemicals. Most of the water that would be withdrawn from the Clinch River to operate the foreign research reactor spent nuclear fuel facilities would be returned to surface water in the Clinch River watershed, although some evaporative losses would be unavoidable. The land dedicated to the foreign research reactor spent nuclear fuel facilities could become available for other urban uses following closure and decommissioning. However, the soils on the site would have to be amended to support land uses such as agriculture, forestry, or wildlife management.

F.4.4.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Oak Ridge Reservation could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the Oak Ridge Reservation would be minimal in most environmental media.

Pollution Prevention: The DOE Oak Ridge Field Office established a Waste Minimization and Pollution Prevention Awareness Plan to reduce the quantity and toxicity of hazardous, mixed, and radioactive wastes generated at the Oak Ridge Reservation. The plan is designed to reduce the possible pollutant releases to the environment and thus increase the protection of employees and the public. All contractors and users that exceed the U.S. Environmental Protection Agency criteria for small-quantity generators are

Table F-83 Cumulative Impacts at the Oak Ridge Reservation

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Land Use (acres)	40	14,335[b]	14,375
Socioeconomics (persons)	190[b]/30[c]	3,917[b]/930[c]	4,107[b]/960[c]
Air Quality (nonradiological)	See Table F-83A	See Table F-83A	See Table F-83A
Occupational and Public Health and Safety			
• MEI Dose (rem/yr)	0.00009	0.0155	0.0156

Table F-83A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Oak Ridge Reservation[a]

Pollutant	Averaging Time	Regulatory Standard (μg/m³)	Cumulative Concentration[b] (μg/m³)

F.4.5 Nevada Test Site

If the Nevada Test Site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period required for the Nevada Test Site to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, this period (Phase 1) is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2), the Nevada Test Site would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory, and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Nevada Test Site until ultimate disposition.

Although the Nevada Test Site has no existing facilities to receive foreign research reactor spent nuclear fuel at the beginning of the policy period, it has facilities that could be modified to receive foreign research reactor spent nuclear fuel within 5 years. These facilities are large hot cells located in the Nevada Research and Development Area on Jackass Flats. Presently these facilities (e.g., E-MAD) have little usage, but some are in acceptable condition. To use the E-MAD facility, a small pool would have to be constructed to be used for transferring the spent nuclear fuel from the transportation casks to containers designed for dry storage. A description of the E-MAD facility is included in Appendix F (Section F.3). The E-MAD facility could be ready within 5 years of the start of the proposed policy period.

The amount of spent nuclear fuel that would be received and managed at the Nevada Test Site under Management Alternative 1, is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, during Phase 2, the Nevada Test Site could receive the TRIGA spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1, Western foreign research reactor spent nuclear fuel under the Regionalization by Geography Alternative, or all foreign research reactor

spent nuclear fuel under the Centralization Alternative.

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Nevada Test Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Nevada Test Site would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and manage it in facilities sized for the reduced amount of spent nuclear fuel. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 5A above.

- Under Implementation Subalternative 1b (Section 2.3.1), the Nevada Test Site would receive from the Idaho National Engineering Laboratory and/or the Savannah River Site only HEU. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the storage of this amount of fuel would be bounded by analysis option 5A (above).

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Nevada Test Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 5A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years and, therefore, the amount of spent nuclear fuel available for acceptance would also be decreased. In such a case, the Nevada Test Site would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Nevada Test Site would be bounded by analysis option 5A above.

- Under Implementation Subalternative 2b (Section 2.2.2.2) the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis option 5A.

- Under Implementation Subalternative 3, (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. The various arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States as the foreign research reactor operators would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent fuel, in this case, cannot be quantified; however, the upper limit, as considered under analysis option 5A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management impacts at the Nevada Test Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Nevada Test Site for Phase 2 until ultimate disposition. For this implementation alternative an analysis option 5B, which is similar to 5A, is considered as follows:

5B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Nevada Test Site where it would be managed at a new wet storage facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel that would be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements). If the Nevada Test Site receives TRIGA spent nuclear fuel from the Idaho National Engineering Laboratory or only western spent fuel, the wet storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

- Under Implementation Alternative 6 (Section 2.3.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Nevada Test Site would not be considered as a site for chemical separation.

Under Management Alternative 3 (Hybrid Alternative) the Nevada Test Site is not considered.

F.4.5.1 Existing Facilities

Existing facilities considered for foreign research reactor spent nuclear fuel storage at the Nevada Test Site include the E-MAD facility in Area 25. For this analysis, the E-MAD facility was considered essentially as new because of the significant modifications needed to use it for foreign research reactor spent nuclear fuel storage. These modifications could be completed sometime between 1996 and 2006. The potential environmental impacts associated with the modification would be bounded by the impacts associated with the construction of a dry storage facility presented in Section F.4.5.2. Impacts from the operation of the E-MAD facility are presented below.

F.4.5.1.1 Socioeconomics

Potential socioeconomic impacts associated with storage option 5A would be attributable to staffing requirements at the E-MAD facility. The staffing requirements for dry storage would be about 120 full time employees. Considering that the total work force at the Nevada Test Site is approximately 4,000 (DOE, 1995g), the addition of 120 full time employees for foreign research reactor spent nuclear fuel storage is not expected to have any measurable socioeconomic impact in the region of influence.

F.4.5.1.2 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Nevada Test Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. For the purpose of these calculations, the refurbished E-MAD facility is treated as a generic dry storage facility. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6. Table F-84 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Nevada Test Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-84 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• E-MAD (dry storage)	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
Storage at:				
• E-MAD (dry storage)	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 5A involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and 193 shipments directly from ports into a dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-85 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Nevada Test Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-85 Handling-Related Impacts to Workers at the Nevada Test Site

	Worker Population Dose (person-rem)	*Worker Population Risk (LCF)*
	E-MAD	*E-MAD*
Phase 2	113	0.05

F.4.5.1.3 Material, Utility, and Energy Requirements

The material, utility, and energy requirements for the E-MAD facility are typical of those for dry storage. Table F-86 presents the estimated material, utility and energy consumption for dry storage.

Table F-86 Annual Utility and Energy Requirements for Dry Storage at the Nevada Test Site

Commodity	Baseline Site Usage	Dry Storage Usage	Percent Increase
Electricity (MW-hr/yr)	176,440	800 - 1,000	0.6 percent
Fuel (l/yr)	[a]	0	0 percent
Water (l/yr)	1,138,000,000	1,590,000[b] 400,000[c]	0.14 percent 0.04 percent

[a] *The majority of the energy used at the Nevada Test Site is provided by electricity. Current usage is not available*

[b] *During receipt and handling*

[c] *During storage*

These requirements represent a small percent of current requirements for the Nevada Test Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Nevada Test Site is expected to decrease because of changes in site mission and a general reduction in employment.

F.4.5.1.4 Waste Management

The contribution of waste associated with the operation of the E-MAD facility is typical of that for new dry storage (Section F.4.5.2.1.13).

F.4.5.1.5 Air Quality

The contribution of air emissions associated with the operation of the E-MAD facility is typical to that for new dry storage (Section F.4.5.2.1.5).

F.4.5.1.6 Water Resources

The effect of the operation of the E-MAD facility on the water usage is typical to that for new dry storage (Section F.4.5.2.1.7).

F.4.5.2 New Facilities (Phase 2)

Analysis options 5A and 5B involve the use of new facilities as discussed above. The environmental impacts analyzed relate to the construction and operation of these facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities, and energy; and waste management.

F.4.5.2.1 Dry Storage

Analysis option 5A involves long-term dry storage of foreign research reactor spent nuclear fuel at the Nevada Test Site. This analysis option would require the construction of a new dry storage facility. The analysis option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 and earlier in this appendix. There are no environmental impact parameters that would discriminate between the two designs.

F.4.5.2.1.1 Land Use

A new dry storage facility would be located in Area 5 in the southeastern portion of the Nevada Test Site. The land in this area can be characterized as sparsely vegetated desert, ready for development. Use of Area 5 for foreign research reactor spent nuclear fuel storage would be consistent with existing land use plans, which designate this area for general use. Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at any of the areas would significantly impact land use patterns on the Nevada Test Site.

F.4.5.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be $370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents approximately 66 percent of the estimated FY 1995 total expenditures for the Nevada Test Site (141 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new dry storage facility are estimated to be $15.6 million for receipt and handling and $0.6 million for storage. These costs represent about 11 percent and 0.5 percent of FY 1995 total expenditures for the Nevada Test Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new dry storage facility is estimated to be facilities for spent nuclear fuel management under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g).

The proposed spent nuclear fuel facilities under Centralization, when fully constructed and under operation, would consist of a series of industrial buildings set within a 36-ha (90-acre) site. The site would not be visible from areas outside the Nevada Test Site. The new dry storage facility for foreign research reactor spent nuclear fuel would be constructed and operated under similar conditions. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility.

F.4.5.2.1.5 Geology

The new dry storage facility for foreign research reactor spent nuclear fuel would be situated on tertiary volcanic or sedimentary rocks near volcanic or intrusive centers where small to medium-size precious metal deposits could be developed. However, because the Nevada Test Site is closed to mining operations, any precious metal deposits that might exist in or around the potential storage site would not be impacted (DOE, 1995g). Further, no mass movement or subsidence and sediment runoff from land disturbances would be expected (DOE, 1995g). The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

F.4.5.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts at the Nevada Test Site associated with the dry storage facility include the generation of fugitive dust from construction activities (e.g., clearing of land, grading, and road preparation) and vehicle emissions from the heavy equipment utilized during the construction phase of the project. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed, barren surfaces.

The construction of this facility would require the clearing of 3.7 ha (7 acres) of land. However, the overall construction impacts to the ambient air quality of the region should be minimal due to the short duration (3 months to 6 years) of the project. Emissions of sulfur dioxide, nitrogen dioxide, and carbon monoxide are assumed to result entirely from diesel exhaust during the construction process. Respirable particulate matter (e.g., PM_{10}) is assumed to be 64 percent of the total suspended particulates estimated for the construction effort. Additionally, wetting controls are assumed to reduce this amount by 50 percent, which is a very conservative estimate.

Table F-87 presents the air quality impacts associated with the construction of the dry storage facility at the Nevada Test Site. Additionally, this table shows that the ambient impacts would be minimal and compliance with existing Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population. The estimated impacts from construction activities were generated using

Table F-87 Estimated Maximum Concentrations of Criteria Pollutants at the Nevada Test Site Attributable to New Dry Storage Construction

Pollutant	Averaging Time	Ambient Standard[b]	Baseline Concentration[a]	Construction Activities
Boundary ($\mu g/m3$):[b, c]				
• Particulate Matter (PM_{10})[a]	24-hour	150	84.90	0.0020
	Annual	50	0.43	0.1107
• Carbon Monoxide	1-hour	40,000	2,748.0	26.756
	8-hour	10,000	2,290.8	3.345
• Sulfur Dioxide	3-hour	1,300	170.3	2.356
	24-hour	365	55.2	0.345
	Annual	80	1.1	0.006
• Nitrogen Oxides	Annual	100	d	0.098

[a] *Source: (DOE, 1995g)*

[b] *64 percent of total suspended particulates is considered to be respirable particulate matter (e.g., PM_{10}) for the construction activities. The standard refers to the actual PM_{10} standard.*

[c] *To convert to $\mu g/ft^3$, multiply by 0.0283*

[d] *No sources indicated*

the U.S. Environmental Protection Agency's regulatory-approved Industrial Source Complex Short-Term Model, Version 2.0 in conjunction with onsite meteorological data from 1991.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or a gaseous [illegible] nuclear fuel], or less than 1 percent of site location. No wetlands are expected to be disturbed because none exist in or around the proposed storage site (DOE, 1995g).

Threatened and Endangered Species: The project area is located within the range of the desert tortoise, a Federally listed threatened species. Recent pre-activity surveys for other nearby projects have not

No change is expected in the community reaction to noise along this route. No mitigation of traffic noise impacts is proposed (DOE, 1995g).

F.4.5.2.1.10 Traffic and Transportation

Construction and operation of a new dry storage facility would involve a small increase in the number of employees commuting to the Nevada Test Site and transportation of foreign research reactor spent nuclear fuel and hazardous chemicals within the site.

The maximum reasonably foreseeable scenario for construction and operations traffic occurs under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS. This would occur in 2001, when there would be about 3,400 full-time employees, and about 1,200,000 people in the region of influence. None of the future baseline levels of service would change due to spent nuclear fuel-related impacts (DOE, 1995g). These conclusions are equally valid for a new dry storage facility for...

F.4.5.2.1.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Nevada Test Site would be attributed to emissions of radioactive material that would be...

the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-89 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Nevada Test Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-89 Handling-Related Impacts to Workers at the Nevada Test Site

	Worker Population Dose (Person-rem)		Worker Population Risk (LCF)	
	New Dry Storage Cask	New Dry Storage Vault	New Dry Storage Cask	New Dry Storage Vault
Phase 2	266	113	0.11	0.05

F.4.5.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Nevada Test Site would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water.

The annual utility and energy requirements during operations are shown in Table F-90. These requirements represent a small percent of current requirements for the Nevada Test Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Nevada Test Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-90 Annual Utility and Energy Requirements for New Dry Storage at the Nevada Test Site

Commodity	Baseline Site Usage	Dry Storage Usage	Percent Increase
Electricity (MW-hr/yr)	176,440	800 - 1,000	0.6 percent
Fuel (l/yr)	[a]	0	0 percent
Water (l/yr)	1,138,000,000	1,590,000[b] 400,000[c]	0.14 percent[b] 0.04 percent[b]

[a] *The majority of energy used at the Nevada Test Site is provided by electricity.*

[b] *During receipt and handling*

[c] *During storage*

F.4.5.2.1.13 Waste Management

Construction of a new dry storage facility at the Nevada Test Site would generate 1,800 m^3 (2,400 yd^3) of [illegible]

debris. The annual quantities of waste generated during operations are shown in Table F-91. These quantities represent a very small percent increase above current levels at the Nevada Test Site. Existing waste management storage and disposal activities at the Nevada Test Site could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Nevada Test Site waste management capacities would be minimal.

Table F-91 Annual Waste Generated for New Dry Storage at the Nevada Test Site

Waste Form	Baseline Site Generation	Dry Storage Generation	Percent Increase
High-Level (m^3/yr)	0	none	0 percent
Transuranic (m^3/yr)	0	none	0 percent
Solid Low-Level (m^3/yr)	10,845	22[a] 1[b]	0.20 percent[a] 0.01 percent[b]
Wastewater (l/yr)	11,000,000	1,590,000[a] 400,000[b]	14.4 percent[a] 3.6 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.5.2.2 Wet Storage

Analysis option 5B involves long-term wet storage of foreign research reactor spent nuclear fuel at the Nevada Test Site. This storage option would require the construction of a new wet storage facility.

F.4.5.2.2.1 Land Use

A new wet storage facility would be located in Area 5 in the southeastern portion of the Nevada Test Site. The land in this area can be characterized as sparsely vegetated desert, ready for development. Use of Area 5 for foreign research reactor spent nuclear fuel storage would be consistent with existing land use plans, which designate this area for general use. Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Nevada Test Site of approximately 4,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Nevada Test Site.

F.4.5.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.5.2.1.3).

F.4.3.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.5.2.1.4).

F.4.5.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.5.2.1.5).

F.4.5.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 3 ha (7 acres) of land. In comparison, approximately 4 ha (10 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction, as presented in Section F.4.5.2.1.6.

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3 micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilating, and air conditioning system would be designed to maintain airflow from areas of low potential

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the CPP-603 at Idaho National Engineering Laboratory. The annual air emission for the CPP-603 was designed to result in ground-level concentrations of less than 0.003 percent of DOE 5480.1B limits for uncontrolled areas. Radiological emissions from the operation of the wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading, and storage are provided in Section F.6.6.1.

F.4.5.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (1.2 million gal) and 2.7 million l (720,000 gal), respectively. With an annual average water usage of approximately 1,138 million l (301 million gal) for the Nevada Test Site, these amounts represent an increase of about 0.17 percent and 0.23 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Nevada Test Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Nevada Test Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Nevada Test Site could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Nevada Test Site would still be well within the design capacities of treatment systems at the Nevada Test Site. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.5.2.2.8 Ecology

Impacts to the ecology would be the same as for new dry storage (Section F.4.5.2.1.8).

F.4.5.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.5.2.1.9).

F.4.5.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.5.2.1.10).

F.4.5.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Nevada Test Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an

80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-92 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Nevada Test Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-92 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at: • New Wet Storage Facility	0.00052	2.6×10^{-10}	0.00052	2.6×10^{-7}
Storage at: • New Wet Storage Facility	4.0×10^{-9}	2.0×10^{-15}	4.7×10^{-9}	2.4×10^{-12}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 5B involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site and 193 shipments directly from ports into a new wet storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-93 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Nevada Test Site. The

worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI requirements represent a small percent of current requirements for the Nevada Test Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Nevada Test Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-94 Annual Utility and Energy Requirements for Wet Storage at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

Commodity	Baseline Site Usage	Wet Storage Usage	Percent Increase
Electricity (MW-hr/yr)	176,440	800 - 1,000	0.84 percent
Fuel (l/yr)	[a]	0	0 percent
Water (l/yr)	1,139,000,000	2,700,000[b] 1,500,000[c]	0.23 percent 0.13 percent

[a] *The majority of energy used at the Nevada Test Site is provided by electricity.*

[b] *During receipt and handling*

[c] *During storage*

F.4.5.2.2.13 Waste Management

Construction of a new wet storage facility at the Nevada Test Site would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-95. These quantities represent a very small percentage increase above current levels at the Nevada Test Site. Existing waste management storage and disposal activities at the Nevada Test Site could accommodate the waste generated by a new wet storage facility. Therefore, the impact of this waste on the existing the Nevada Test Site waste management capacities would be minimal.

Table F-95 Annual Waste Generated for Wet Storage at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

Waste Form	Baseline Site Generation	Wet Storage Generation	Percent Increase
High-Level (m^3/yr)	0	none	0 percent
Transuranic (m^3/yr)	0	none	0 percent
Solid Low-Level (m^3/yr)	10,845	16[a] 1[b]	0.15 percent 0.01 percent
Wastewater (l/yr)	11,000,000	1,590,000[a] 400,000[b]	14.5 percent 3.6 percent

[a] *During receipt and handling*

Table F-96 presents the frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-96 Frequency and Consequences of Accidents at the Nevada Test Site

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem	Worker (mrem)
Dry Storage Accidents[a]					

Table F-98 Frequency and Consequences of Accidents at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	*Consequences*			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
• Spent Nuclear Fuel Assembly Breach	0.16	0.054	0.0016	0.33	0.10
• Accidental Criticality	0.0031	88	15	54	1,300
• Aircraft Crash	1×10^{-6}	29	4.2	61	290

Table F-99 Annual Risks of Accidents at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

	Consequences			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
• Spent Nuclear Fuel Assembly Breach	4.2×10^{-9}	1.3×10^{-10}	0.000026	6.4×10^{-9}
• Accidental Criticality	1.4×10^{-7}	2.3×10^{-8}	0.000084	0.000016
• Aircraft Crash	1.5×10^{-11}	2.1×10^{-12}	3.1×10^{-8}	1.2×10^{-10}

accidents that resulted in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Nevada Test Site, the radiological accident with the highest MEI dose is the aircraft crash into a dry storage facility with fire (Table F-96). For this accident, the MEI dose would be 180 mrem. For the air immersion and ground surface pathways only, the dose would be 1.0 mrem, which is less than the 100 mrem limit used in this analysis. Therefore, no secondary impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies from radiological accidents involving foreign research reactor spent nuclear fuel storage would be expected at the Nevada Test Site.

F.4.5.4 Cumulative Impacts at the Nevada Test Site

The section presents the cumulative impacts of the proposed action, potential impacts of other contemplated DOE actions, and current activities at the site. A major portion of the presentation is based on information included in the Programmatic SNF&INEL Final EIS (DOE, 1995g) and the Tritium Supply

Table F-100 Cumulative Impacts at the Nevada Test Site

Table F-100A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Nevada Test Site[a]

Pollutant	Averaging Time	Regulatory Standard ($\mu g/m^3$)	Cumulative Concentration[b] ($\mu g/m^3$)
Carbon Monoxide	1-hour	40,000	2,815 (7%)
	8-hour	10,000	2,306 (23%)
Nitrogen Oxides	Annual	100	4.2 (4.2%)
Sulfur Dioxide	3-hour	1300	173.6 (13.3%)
	24-hour	365	55.5 (15.2%)
	Annual	80	1.1 (1.3%)
Particulate Matter (PM_{10})	24-hr	150	85 (56.6%)
	Annual	50	0.54 (1.1%)

[a] *Concentrations represent activities from: foreign research reactor spent nuclear fuel management, DOE-owned spent nuclear fuel management, construction and operation of an Expended Core Facility, and construction and operation of a tritium production and recycling facilities*

[b] *Number in parentheses indicate the percentage of the Regulatory Standard*

F.4.5.5 Unavoidable Adverse Environmental Impacts

Construction of the potential new foreign research reactor spent nuclear fuel storage facilities would require the disturbance of approximately 4 ha (10 acres) of undeveloped land. Although this represents less than one percent of the undeveloped land on the Nevada Test Site, it would eliminate potential terrestrial wildlife habitat, including habitat potentially suitable for the Federally-listed desert tortoise. It would also require the dedication of a small land parcel potentially suitable for other construction projects, but similar land parcels are abundant on the Nevada Test Site.

F.4.5.6 Irreversible and Irretrievable Commitments of Resources

Construction and operation of new foreign research reactor spent nuclear fuel facilities would require commitments of electrical energy, fuel, concrete, steel, sand, gravel and miscellaneous chemicals. Groundwater to operate the foreign research reactor spent nuclear fuel facilities would be withdrawn from an aquifer that is presently experiencing localized overdraught. Further studies would be necessary to quantify any irreversible effects on future groundwater availability attributable to spent nuclear fuel withdrawals from that aquifer. The land dedicated to the foreign research reactor spent nuclear fuel facilities would become available for other rural uses following closure and decommissioning.

F.4.5.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Nevada Test Site could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the site would be minimal in most environmental media.

Pollution Prevention: The DOE Nevada Field Office published a Waste Minimization and Pollution Prevention Awareness Plan in June 1991 to reduce the quantity and toxicity of hazardous, mixed, and radioactive wastes generated at DOE Nevada Field Office facilities. The plan is designed to reduce the possible pollutant releases to the environment and thus increase the protection of employees and the public. All DOE Nevada Field Office contractors and the Nevada Test Site users that exceed the Environmental Protection Agency criteria for small-quantity generators are establishing their own waste [illegible] awareness programs that are implemented by the DOE Nevada

F.5 Occupational Radiation Impacts from Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel

Occupational exposure to gamma radiation would depend largely on the operational history of the spent nuclear fuel elements to be stored in the facility and the length of time that these elements have been allowed to decay from the time that they were taken out of the reactor until they were placed in the cask for shipment to the storage facility. Normally, the decay time for fuel elements is established so that the gamma heating in the transportation cask is within specification and the radiation field on the outside cask surface is 200 mrem per hour or less. Special shipments can be made, however, with higher cask surface radiation fields, provided other requirements are met. Radiation exposures to personnel during receiving operations and surveys would depend on the level of radiation that is measurable on the exterior (surface) of the transportation cask. These initial operations are anticipated to provide the majority of personnel exposure since the remaining operations would be remote and could take advantage of the shielding built into the facility.

Realistic annual occupational radiation exposure estimates for facility operation can be performed once the following have been established:

- determination of accurate decay-time averaged values for the spent nuclear fuel,
- development of shielding characteristics for transportation casks for the spent nuclear fuel to be shipped to the facility,
- definition of personnel requirements for each of the individual operations to be accomplished within the facility, and
- completion of a time-motion study for the spent nuclear fuel element movement through the preliminary design of the facility.

The analyses in this appendix are based on a best estimate of the above conditions. The potential impacts are given in doses per cask shipment, so that the results can be simply multiplied by the total number of shipments for each potential storage arrangement.

Wet Storage: Occupational radiation exposure from the receipt, handling and storage of foreign research reactor spent nuclear fuel at a wet storage facility is treated in a generic way for all potential management sites, since the activities are essentially identical regardless of where the facility is located. It is based on actual handling experience of spent nuclear fuel at the Idaho National Engineering Laboratory and the Savannah River Site.

The workers involved with each cask were assumed to include the shipping agent, shift foremen, health physics technicians, and equipment operators. The equipment operators include onsite workers who remove each cask from its shipping container and transport it to the receiving bay, and those who perform most of the actual labor involved thereafter, such as transferring the spent nuclear fuel to storage and decontaminating the empty cask prior to returning it to the owners. Thus, while the assessment does not distinguish between them, the operators are a diverse group of workers whose distinct duties make it unlikely that the same operators could receive all of the calculated individual doses discussed below. As a result, it was assumed that the two foremen and two operators involved in handling the spent nuclear fuel casks outside the receiving facility would be different than the two foremen and two operators working inside the facility (the health physics technicians were assumed to be the same). This provides a conservative estimate of 12 workers.

In order to estimate the occupational radiation doses from the handling of foreign research reactor spent nuclear fuel transportation casks at the spent nuclear fuel management sites, it was necessary to develop a curve of dose rate versus distance for these casks. Historical data based on 44 research reactor spent nuclear fuel transportation cask receipts at either the Savannah River Site or the Idaho National Engineering Laboratory were obtained and evaluated. This historical data showed an average measured dose rate of approximately 2.3 mrem per hour at 1 m (3.3 ft) from the surface of the transportation cask. One cask, however, was measured to be 20 mrem per hour at 1 m (3.3 ft). To encompass this historical data, including the highest measured dose rate cask, an analysis was performed that assumed a dose rate of 23 mrem per hour at 1 m (3.3 ft) from the cask surface. It should be noted that, in the unlikely event that a higher dose rate transportation cask was received at the management site, radiological control procedures for as low as reasonably achievable limits would be utilized to ensure that the worker doses would be minimized. Dose rate reduction is usually accomplished by a combination of restrictions on time, distance from the source, and the provision of additional radiation shielding.

The plot of bounding transportation cask dose rate versus distance in Figure F-50 was developed using the ZYLIND computer code and appropriate conservative methodology. ZYLIND (RSIC, 1990) is a shielding computer code that uses the point kernel method to calculate photon dose rates from a cylindrical source and shield geometry. ZYLIND was developed in Germany in 1989 and then released to the Oak Ridge National Laboratory Radiation Shielding Information Center. ZYLIND has been extensively validated by comparison to measured dose rates from several hundred cylindrical containers with radioactive materials. ZYLIND calculated dose rates that were conservative and within 10 to 20 percent of the measured dose rates. ZYLIND allows the photon energy source to be divided into up to 20 energy groups from 0 to 10 million electron volts (Mev), allows up to eight materials regions, and includes mass attenuation and dose buildup information for a wide range of shielding materials.

The methodology used in calculating bounding transportation cask dose rates had four underlying assumptions. First, it was assumed that the dose rate at 1 m (3.3 ft) from the cask surface is 23 mrem per hour. Second, neutron dose rates from foreign research reactor spent nuclear fuel were assumed to be negligible and the only dose was assumed to be due to gamma (photon) radiation. A third assumption was that the foreign research reactor spent nuclear fuel source term inside a cask could be conservatively simulated by a single 1.0 Mev gamma energy group. Traditional NRC source terms (DiNunno et al, 1962) for spent nuclear fuel fission products assume an average gamma energy of 0.7 Mev. By using 1.0 Mev, the average gamma energy is expected to be conservatively bounded. Finally, it was assumed that the use of a point kernel cylindrical source-shield computer code (i.e., ZYLIND) would conservatively calculate the dose rates from a transportation cask.

The principal inputs for the calculation were the ZYLIND computer code manual (Radiation Shielding Information Center ZYLIND) and the U.S. Department of Transportation Certificates of Competent Authority for seven transportation cask designs that are likely to be used for the shipment of foreign research reactor spent nuclear fuel. These seven designs are: TN7, GNS-11, LHRL-120, NAC/LWT, PEGASE (IU-04), BMI-1, and GE-2000. These transportation casks are described in Appendix B, Section B.2. The U.S. Department of Transportation Certificates of Competent Authority provided geometry data on the cask inside cavity dimensions and the thickness and material composition of shielding adjacent to the cavity for each design.

With the cask geometry information, a set of ZYLIND calculations was performed for each design. An initial 1.0 Mev gamma source was estimated and ZYLIND was executed to calculate the dose rate at 1 m (3.3 ft) from the cask surface. This source was iterated upon until the 1 m (3.3 ft) dose rate equaled 23 mrem per hour. After this source was determined, the same source and cask geometry were rerun to calculate the dose rate at distances of from 0-50 m (0-164 ft) from the cask surface. This process was repeated for each of the transportation cask designs. The resulting dose rates at distance for each cask design were compared and the highest dose rate response at all distances was synthesized from this data to produce Figure F-50.

Table F-101 shows the actual dose rates encountered during receipt and handling for essentially all of the foreign research reactor spent nuclear fuel casks, which are expected to be one to two orders of magnitude lower than the limit, based on actual experience with foreign research reactor spent nuclear fuel in the past.

Table F-102 presents the wet storage collective dose for unloading one transportation cask using time, distance, and personnel data from the Idaho National Engineering Laboratory and the dose rate curve in Figure F-50. The total worker dose per transportation cask was calculated to be 0.31 person-rem. The actual distances for each worker are based on conservative estimates of actual work experience that would reflect an as low as reasonably achievable Radiation Protection Program as required by DOE regulation (10 CFR 835).

Generic Dry Cask Storage: The receipt, handling, and storage occupational radiation doses (deep dose equivalents) for dry storage are also treated in a generic way, since the operation of the general facility designed for dry storage would be the same at any management site. The assessment is based on Pressurized Water Reactor spent nuclear fuel from the reactor's spent nuclear fuel storage pool to an NRC-licensed dry storage facility at the Calvert Cliffs nuclear power plant in Maryland. The system employed is the horizontal module system (NUHOMS), which was selected for this assessment for two reasons: (1) it is a current, regulatory-approved design that is readily available for foreign research reactor spent nuclear fuel dry storage, and (2) the worker dose rates calculated for the system are among the highest of the current systems now in use for storage of commercial spent nuclear fuel. As a result, the system analysis provides a reasonably conservative estimate for storage of foreign research reactor spent nuclear fuel in NUHOMS and a reasonable upperbound assessment for all other foreign research reactor spent nuclear fuel generic dry storage. The Calvert Cliffs Safety Analysis Report does not identify each category of worker associated with receipt, handling, transfer, and storage of spent nuclear fuel. As a result, for this assessment, the doses (deep dose equivalent) were assumed to be the same for all workers (titled "operators" in the assessment). However, it would appear from the work activities that it cannot be the same operators who support each of the activities. As a result, job titles comparable to those considered for wet storage have been defined in order to determine the average worker dose per cask. Therefore, it is assumed that the following categories of workers are involved: foremen (2), health physics technicians (2), equipment operators for the storage pool activities (2), and different equipment operators for the one-site transport and transfer of the spent nuclear fuel from the transfer cask to the dry storage cask (3), welders (2), helium leak test technician (1), and dye penetrant test technician (1). Each of the distances listed is the average distance for all of the workers involved in each one of the 25 specific activities associated with receipt, handling, transfer, and dry storage. Thus, for example, the first activity [loading fuel into the container (dry shielded canister)], would involve four workers (one foreman, one health physics technician, and two operators) in the Spent Fuel Pool area. The results indicate that the collective dose to the working crew of 13 would be 1.5 person-rem per NUHOMS cask transfer. A transfer cask load is approximately equal to the foreign research reactor spent nuclear fuel inventory of eight transportation casks.

IFSF (Dry Vault) Specific Dry Storage: Based on data provided by the Idaho National Engineering Laboratory, Table F-103 was generated to present the occupational dose for unloading one transportation cask into the IFSF. The collective dose to unload one transportation cask into the IFSF was calculated to be 0.32 person-rem. This dose is considered representative of a generic dry vault storage facility.

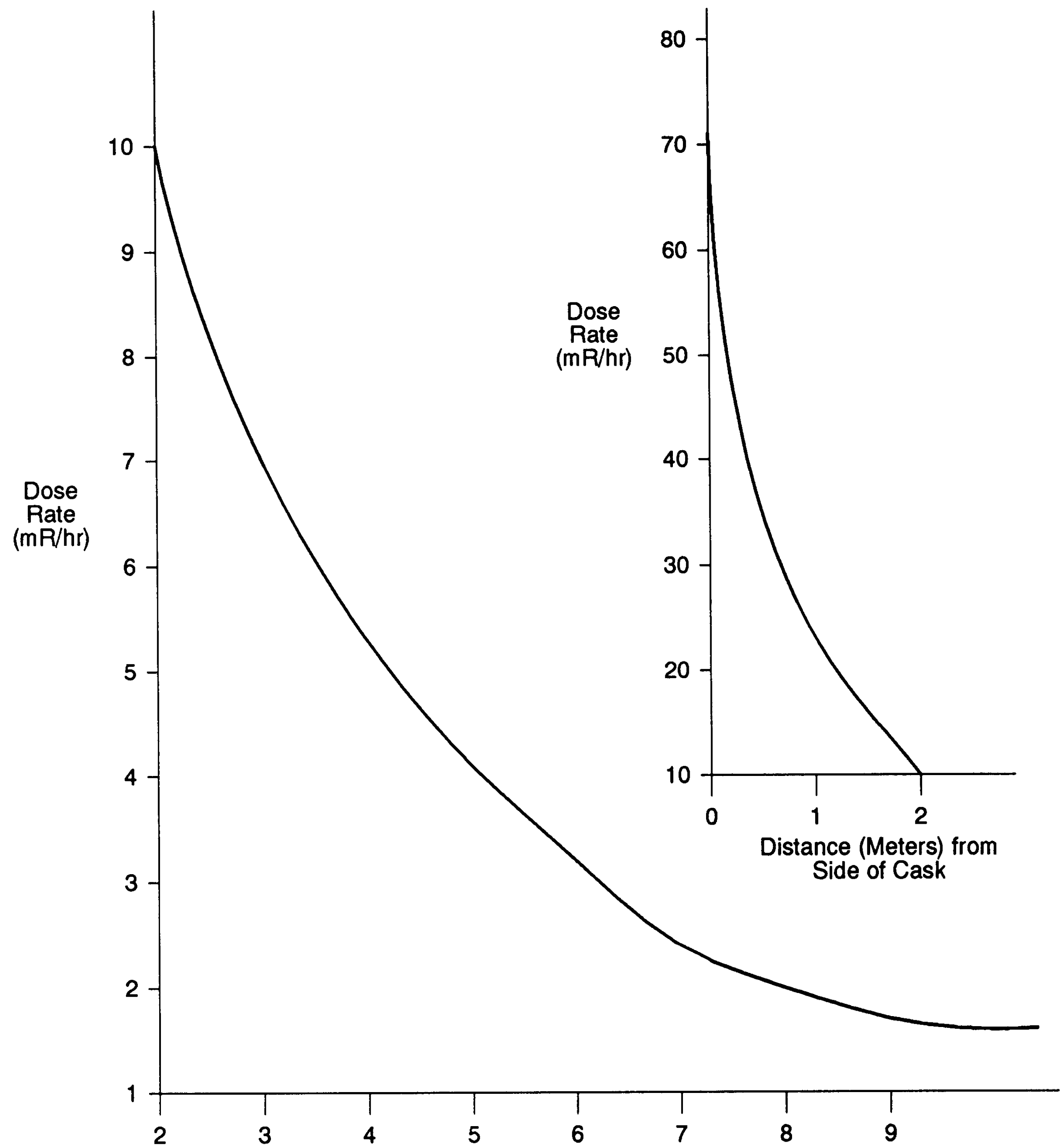

Figure F-50 Bounding Transportation Cask
(Dose Rate at a Distance Normalized to 10 millirem/hour at 2 meters [6.6 feet])

Table F-101 Actual Foreign Research Reactor Spent Nuclear Fuel Transportation Cask Dose Rate Measurements

Date	*Cask Model*	*Fuel*	*Side Cask Measured Dose Rate (mrem/hr) at 1 m (3.3 ft)*
Savannah River Site-Provided Data			
10/2/94	IU04-PEGASE		0.4
10/2/94	PEGASE		2.08
10/2/94	TN-7		8.4
10/2/94	GNS-11		1.2
12/18/87	PEGASE	DR3	1.5
1/26/89	PEGASE	DR3	0.6
12/31/87	PEGASE	ORPHEE	1.4
10/30/86	PEGASE	ORPHEE	0.5
9/1/87	PEGASE	SILOE	1.0
11/5/87	PEGASE	SILOE	0.3
12/30/87	PEGASE	SILOE	0.3
2/7/89	TN-7/2	RHF	15.0
8/16/88	TN-7/2	RHF	20.0
9/25/81	SWED.R2-B/23	AAR	0.9
1/20/89	TN-1	HFR	6.0
7/20/88	TN-1	HFR	0.5
6/8/88	GNS-11	FRJ-2	8.0
8/30/88	GNS-11	FRJ-2	0.8
8/30/88	GNS-11	FRJ-2	10.0
2/27/86	GOSLAR NO.1	FRG	2.0
11/12/86	GOSLAR NO.1	FRG	1.0
2/26/86	GOSLAR NO.2	FRG	0.2
3/10/80	GOSLAR NO.1	ASTRA	0.2
3/14/80	GOSLAR NO.2	ASTRA	0.1
1/29/86	BMI-1	RINC	0.5
2/6/86	BMI-1	RINC	0.4
6/5/84	BMI-1	U.VA.	0.5
6/11/84	BMI-1	U.VA.	0.5
9/11/84	BMI-1	U.MICH.	0.1
10/13/87	BMI-1	U.MICH.	0.1
7/14/81	BMI-1	U.MICH.	1.0
10/6/87	BMI-1	U.MICH.	0.1
Idaho National Engineering Laboratory-Provided Data			
	BMI-1	CORNELL-TRIGA	3.5
	BMI-1	BERKLEY-TRIGA	0.5
	BMI-1	MICHIGAN-TRIGA	1.0
	BMI-1	BERKLEY-TRIGA	0.5
	GE-700	BNL-HFBR	1.0
	GE-700	BNL-HFBR	1.0
	GE-700	BNL-HFBR	1.0
	GE-700	U. OF MISSOURI	0.1
	BMI-1	CORNELL-TRIGA	3.5
	BMI-1	MICHIGAN-TRIGA	0.1
	BMI-1	HANFORD-TRIGA	1.0
	BMI-1	HANFORD-TRIGA	1.0

Table F-102 Worker Dose Assessment for Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel in Wet Storage

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
Transport Receipt						
Shipping Agent	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00		1.1E-03
Health Physics Tech	1.0	23.0	5	1.9E+00	1	1.9E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			15	3.6E+00		3.6E-03
Guards	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00		1.1E-03
Remove Container Cover						
Foreman	10.0	1.5	20	5.0E+01	1	5.0E-04
	5.0	4.2	10	7.0E+01	1	7.0E-04
Subtotal			30	1.2E+00		1.2E-03
Operators	0.3	50.0	15	1.3E+01	3	3.8E-02
	0.3	50.0	10	8.3E+00	1	8.3E-03
	1.0	23.0	1	3.8E-01	1	3.8E-04
	2.0	10.0	10	1.7E+00	2	3.3E-03
Subtotal			36	2.3E+01		5.0E-02
Survey Cask						
Health Physics Tech	0.3	50.0	45	3.8E+01	1	3.8E-02
Subtotal			45	3.8E+01		3.8E-02
Removal of Impact Limiters						
Foreman	5.0	4.2	50	3.5E+00	1	3.5E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	5.0	4.2	45	3.2E+00	1	3.2E-03
	0.3	50.0	15	1.3E+01	1	1.3E-02
Subtotal			60	1.6E+01		1.6E-02
Operators	0.3	50.0	10	8.3E+00	3	2.5E-02
	0.5	36.0	60	3.6E+01	2	7.2E-02
Subtotal			70	4.4E+01		9.7E-02
Move Cask						
Equipment Operators	0.3	50.0	5	4.2E+00	1	4.2E-03
Subtotal			5	4.2E+00		4.2E-03

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
Testing & Verification of Integrity						
Health Physics Tech	5.00	4.2	30	2.1E+00	1	2.1E-03
	2.00	10.0	28	4.7E+00	1	4.7E-03
	0.3	50.0	2	1.7E+00	1	1.7E-03
Subtotal			60	8.4E+00		8.4E-03
Operators	0.30	50.0	30	2.5E+01	2	5.0E-02
	4.00	5.3	30	2.7E+00	2	5.3E-03
Subtotal			60	2.8E+01		5.5E-02
Movement of Cask to Unloading Pool and Immersion						
Operators	0.30	50.0	2	1.7E+00	3	5.0E-03
	NA	0.1	58	9.7E-02	2	1.9E-04
Subtotal			60	1.8E+00		5.2E-03
Cask Unloading/Inspection/Storage						
Foreman	NA	0.1	240	4.0E-01	1	4.0E-04
Subtotal			240	4.0E-01		4.0E-04
Safeguards	NA	0.1	240	4.0E-01	1	4.0E-04
Subtotal			240	4.0E-01		4.0E-04
Operators	NA	0.1	240	4.0E-01	5	2.0E-03
Subtotal			240	4.0E-01		2.0E-03
Removal of Cask from Unloading Pool						
Operators	NA	0.1	60	1.0E-01	2	2.0E-04
Subtotal			60	1.0E-01		2.0E-04
Replacement of Cask of Transport and all subsequent operators assumed to be no exposure greater than background						
	NA	0.0	90	0	5	0
Subtotal				0		0
Total				**4.4E+01 (Max.)**		**3.1E-01**

Table F-103 Worker Dose Assessment for Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel in a Dry Storage Facility (Irradiated Fuel Storage Facility) or Generic Vault

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
Transport Receipt						
Shipping Agent	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00		1.1E-03
Health Physics Tech	1.0	23.0	5	1.9E+00	1	1.9E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			15	3.6E+00		3.6E-03
Guards	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00	1	1.1E-03
Remove Container Cover						
Foreman	10.0	1.5	20	5.0E-01	1	5.0E-04
	6.0	1.2	10	7.0E-01	1	7.0E-04
Subtotal			30	1.2E+00		1.2E-03
Operators	0.3	50.0	15	1.3E+01	3	3.8E-02
	0.3	50.0	10	8.3E+00	1	8.3E-03

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
	1.0	23.0	1	3.8E-01	1	3.8E-04
	2.0	10.0	10	1.7E+00	2	3.3E-03
Subtotal			36	2.3E+01		5.0E-02
Survey Cask						
Health Physics Tech	0.3	50.0	45	3.8E+01	1	3.8E-02
Subtotal			45	3.8E+01		3.8E-02
Removal of Impact Limiters						
Foreman	5.0	4.2	50	3.5E+00	1	3.5E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	5.0	4.2	45	3.2E+00	1	3.2E-03
	0.3	50.0	15	1.3E+01	1	1.3E-02
Subtotal			60	1.6E+01		1.6E-02
Operators	0.3	50.0	10	8.3E+00	3	2.5E-02
	0.5	36.0	60	3.6E+01	2	7.2E-02
Subtotal			70	4.4E+01		9.7E-02
Removal of Cask from Transport to Transfer Cart						
Foreman	8.0	2.1	45	1.6E+00	1	1.6E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
	1.0	23.0	5	1.9E+00	1	1.9E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	8.00	2.1	30	1.1E+00	1	1.1E-03
	2.00	10.0	20	3.3E+00	1	3.3E-03
	1.00	23.0	10	3.8E+00	1	3.8E-03
Subtotal			80	8.2E+00		8.2E-03
Equipment Operators	4.00	5.3	59	5.2E+00	2	1.0E02
	1.00	23.0	1	3.8E-01	1	3.8E-04
Subtotal			60	5.6E+00		1.1E-02
Testing & Verification of Integrity, Lid Bolt Removal						
Foreman	4.00	5.3	60	5.3E+00	1	5.3E-03
Subtotal			60	5.3E+00		5.3E-03
Health Physics Tech	5.00	4.2	30	2.1E+00	1	2.1E-03
	2.00	10.0	28	4.7E+00	1	4.7E-03
	0.30	50.0	2	1.7E+00	1	1.7E-03
Subtotal			60	8.4E+00		8.4E-03
Operators	0.30	50.0	30	2.5E+01	2	5.0E-02
	4.00	5.3	30	2.7E+00	2	5.3E-03
Subtotal			60	2.8E+01		5.5E-02
Movement of Cask into Handling Cove						
Foreman	4.00	5.3	60	5.3E+00	1	5.3E-03
Subtotal			60	5.3E+00		5.3E-03
Operators	4.0	5.3	10	8.8E-01	2	1.8E-03
	NA	0.1	50	8.3E-02	2	1.7E-04
Subtotal			60	9.7E-01		1.9E-03
Cask Unloading/Inspection/Storage						
Foreman	NA	0.1	480	8.0E-01	1	8.0E-04
Subtotal			480	8.0E-01		8.0E-04
QA Inspector	NA	0.1	480	8.0E-01	1	8.0E-04
Subtotal			480	8.0E-01		8.0E-04

Exposed Workers	*A.* *Exposure Distance (m)*	*B.* *Dose Rate (mrem/hr)*	*C.* *Exposure Time (minutes/cask)*	*D.* *Dose/Cask-Person (mrem)*	*E.* *Number of Exposed Workers*	*F.* *Collective Dose (Person-rem)*
Operators	NA	0.1	480	8.0E-01	2	1.6E-03
Subtotal			480	8.0E-01		1.6E-03
Removal of Cask from Handling Cove						
Foreman	NA	0.1	60	1.0E-01	1	1.0E-04
Subtotal			60	1.0E-01		1.0E-04
Operators	NA	0.1	60	1.0E-01	2	2.0E-04
Subtotal			60	1.0E-01		2.0E-04
Replacement of Cask of Transport and All Subsequent Operators Assumed to be No Exposure Greater than Background	NA	0.0	90	0	5	0
Subtotal			90	0		0
Total				**4.4E+01 (Max.)**		**3.2E-01**

DSC = Dry Shielded Canister

Transfer Between Storage Facilities: The collective doses were calculated for loading fuel into a pod, a dry vault (i.e., the IFSF), and dry cask (i.e., Calvert Cliffs NUHOMS) or during transfer between these facilities. It was assumed that larger commercial spent nuclear fuel transportation casks are used for intersite and intrasite movement of foreign research reactor spent nuclear fuel within the United States. Their capacity is approximately four times that of the foreign research reactor spent nuclear fuel transportation casks from overseas. It was also assumed that the transfer cask for the dry cask design has a capacity which is approximately eight times that of the overseas foreign research reactor spent nuclear fuel transportation casks.

F.6 Evaluation Methodologies and Assumptions for Incident-Free Operations and Hypothetical Accidents at Management Sites

Appendix F.6 describes only the methodologies and assumptions used for estimating radiation exposure (doses) to individuals and the general public from releases of radioactivity during incident-free operations and hypothetical accidents at potential management sites. The descriptions of similar evaluations for ground and marine transportation and port accidents are documented in Appendix E and Appendix D.

F.6.1 Analysis Methods for Evaluation of Radiation Exposure

F.6.1.1 General

The evaluation of incident-free operations and hypothetical accidental radioactive material releases at the proposed storage sites was performed to assess the impact of possible radiation exposure to individuals and the general population. The analysis assumes that the same operations are being carried out at different potential storage locations. The impact of the same radioactive material releases was evaluated at all potential sites. This approach provides a consistent method for comparing the effects of the proposed alternative actions.

F.6.1.2 Exposure Impacts to Be Estimated

The impact of radiation exposure (dose) to the following individuals and the general population is calculated for incident-free operation of the spent nuclear fuel storage facility and for accident conditions:

- ***Worker:*** An individual located 100 m (330 ft) from the radioactive material release point.[2] The dose to the worker is calculated for the 50th-percentile meteorology only (DOE, 1992a).

- ***MEI:*** A theoretical individual living at the storage site boundary and receiving the maximum exposure.

- ***NPAI:*** At some storage sites, highways used by the public may cross the Federal reservation where foreign research reactor spent nuclear fuel operations could be conducted. Consequently, these analyses included evaluation of the exposure to a [illegible]

  Based on experience from emergency exercises, emergency response teams would be a [illegible] ble to evacuate such an individual within 2 hours, so this was the exposure time used in the calculations.

- ***General population*** within an 80 km (50 mi) radius of the facility.

The doses to the NPAI, MEI and general population are calculated for the 50th- and 95th-percentile

Exposed Workers	*A. Exposure Distance (m)*	*B. Dose Rate (mrem/hr)*	*C. Exposure Time (minutes/cask)*	*D. Dose/Cask-Person (mrem)*	*E. Number of Exposed Workers*	*F. Collective Dose (Person-rem)*
Operators	NA	0.1	480	8.0E-01	2	1.6E-03
Subtotal			480	8.0E-01		1.6E-03
Removal of Cask from Handling Cove						
Foreman	NA	0.1	60	1.0E-01	1	1.0E-04
Subtotal			60	1.0E-01		1.0E-04
Operators	NA	0.1	60	1.0E-01	2	2.0E-04
Subtotal			60	1.0E-01		2.0E-04
Replacement of Cask of Transport and All Subsequent Operators Assumed to be No Exposure Greater than Background	NA	0.0	90	0	5	0
Subtotal			90	0		0
Total				**4.4E+01 (Max.)**		**3.2E-01**

DSC = Dry Shielded Canister

Transfer Between Storage Facilities: The collective doses were calculated for loading fuel into a pod, a dry vault (i.e., the IFSF), and dry cask (i.e., Calvert Cliffs NUHOMS) or during transfer between these facilities. It was assumed that larger commercial spent nuclear fuel transportation casks are used for intersite and intrasite movement of foreign research reactor spent nuclear fuel within the United States. Their capacity is approximately four times that of the foreign research reactor spent nuclear fuel transportation casks from overseas. It was also assumed that the transfer cask for the dry cask design has a capacity which is approximately eight times that of the overseas foreign research reactor spent nuclear fuel transportation casks.

F.6 Evaluation Methodologies and Assumptions for Incident-Free Operations and Hypothetical Accidents at Management Sites

Appendix F.6 describes only the methodologies and assumptions used for estimating radiation exposure (doses) to individuals and the general public from releases of radioactivity during incident-free operations and hypothetical accidents at potential management sites. The descriptions of similar evaluations for ground and marine transportation and port accidents are documented in Appendix E and Appendix D.

F.6.1 Analysis Methods for Evaluation of Radiation Exposure

F.6.1.1 General

An evaluation of incident-free operations and hypothetical accidental radioactive material releases at the proposed storage sites was performed to assess the impact of possible radiation exposure to individuals and the general population. The analysis assumes that the same operations are being carried out at different potential storage locations. The impact of the same radioactive material releases was evaluated at all potential sites. This approach provides a consistent method for comparing the effects of the proposed alternative actions.

intestine, small intestine, and stomach. Weighting factors are used for various body organs to calculate weighted or committed effective dose equivalent (EDE) from radiation inside the body due to inhalation or ingestion. The committed EDE value is the summation of the committed dose equivalent to the specific organ weighted by the relative risk to that organ compared to an equivalent whole-body exposure.

The program also estimates deep-dose equivalent for the external exposure pathways (immersion in the radioactive material and exposure to ground contamination) and a 50-year committed EDE for the internal exposure pathways. The sum of the deep-dose equivalent for external pathways and the committed EDE for internal pathways is called the cumulative dose or "total EDE" in this EIS and is also estimated by the GENII program.

The exposure from ingestion of contaminated terrestrial food and animal products is calculated on a yearly basis. However, it is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose exceeds the protective action guidelines for use in the event of radiological accidents (EPA, 1991). No reduction of exposure due to protective actions was accounted for in this analysis, however. This results in a conservative approach that may overestimate health effects within an exposed population, but allows for consistent comparisons between alternatives.

F.6.1.3 Evaluation of Health Effects

Health effects are calculated from the exposure results. The risk factors used for calculations of health effects are taken from International Commission on Radiological Protection Publication 60 (ICRP, 1991) (see Table F-104). From this list only the factors associated with the fatal cancers were used in the analysis. Other factors are given as additional information for completeness.

Table F-104 Risk Estimators for Health Effects from Ionizing Radiation

of percentage of time that the wind blows in specific directions (i.e., south, south-southwest, southwest, etc.) for the given midpoint (or average) wind speed class and atmospheric stability. Accident consequence calculations were performed using 50th- and 95th-percentile meteorological conditions. The 50th-percentile condition represents the median meteorological condition, and is defined as that for which more severe conditions occur 50 percent of the time. The 95th-percentile condition represents relatively low probability meteorological conditions which produce higher calculated exposures, and is defined as that condition which is not exceeded more than 5 percent of the time. GENII determines 50th- and 95th-percentile meteorological conditions using site-specific joint frequency distribution weather data.

F.6.1.6 Computer Programs

The following computer programs were used to evaluate the radiation exposure to the specified individuals and the general population.

GENII: The GENII code (Napier et al., 1988) was used to model both acute and chronic releases to the atmosphere. This code was developed by the Pacific Northwest Laboratory to incorporate the internal dosimetry models recommended in International Commission on Radiological Protection Publication 26 (ICRP, 1977) and Publication 30 (ICRP, 1979-1982) into environmental pathway analysis models in use at the Pacific Northwest Laboratory. This code has been used by the Pacific Northwest Laboratory and other laboratories in site-wide dosimetry calculations. It has been extensively validated and quality assured.

ORIGEN2: ORIGEN2 (Croff, 1980) is a computer code system for calculating the buildup and decay of radioactive materials (fission products, actinides, and activation products). The code input was modeled to describe the HEU and low enriched uranium (LEU) research reactor nuclear fuel system and used neutronic cross-section data that are distinct to these fuels. ORIGEN2 has been used extensively by the Argonne National Laboratory in the RERTR program in estimating nuclide inventories of irradiated fuels. The code and the specific neutronic cross-section parameters for HEU and LEU fuels were acquired from the Argonne National Laboratory. ORIGEN2 is widely used and accepted throughout the nuclear industry.

F.6.2 Screening/Selection of Accidents for Detailed Examination

Accidents considered for inclusion in the detailed analyses are similar to those analyzed in the Programmatic SNF&INEL Final EIS for the spent nuclear fuel storage facility operations (DOE, 1995g). The analyzed accident scenarios in the Programmatic SNF&INEL Final EIS for each potential storage site were reviewed to identify the bounding accidents to be considered in this EIS. The review included accidents initiated by natural phenomena (earthquakes, tornadoes, hurricanes, etc.) and accidents initiated from human or equipment failure (fires, explosions, aircraft crashes, transportation accidents, and terrorism).

A review of accidents indicates that only severe accident conditions could result in a release of radioactive material to the environment or an increase in radiation levels. Some types of accidents, such as procedure violations, spills of small volumes of water containing radioactive particles, and most other types of common human error may occur more frequently than the more severe accidents analyzed. However, these accidents do not involve enough radioactive material or radiation to result in a significant release to the environment or a meaningful increase in radiation levels. Stated another way, the very low consequences associated with these events produce smaller risks than those for the accidents analyzed, even when combined with a higher probability of occurrence. Consequently, they have not been included in the results presented in this EIS.

Accidents initiated at nearby facilities, either by other activities unrelated to spent nuclear fuel handling or storage or during construction of a wet or dry storage facility, would not produce effects more severe than the sequence of events being analyzed. This is because foreign research reactor spent nuclear fuel undergoing examination or in the process of being stored would not need special conditions or uninterrupted operator attention to prevent overheating or to maintain containment or shielding. Therefore, evacuation in response to an accident at some other facility would not compromise integrity of the spent nuclear fuel.

The potential for common-cause accidents at a storage facility has been considered. It is possible for natural phenomena, like an earthquake, to produce more than one accident at a site causing a situation that results in a release of radioactive material into the atmosphere or an increase in radiation levels due to loss of shielding. However, the probability of two or more accidents having maximum consequences occurring concurrently is less than the probability of the individual events. For example, if an earthquake affected the wet storage facility, a crane might fail causing damage to stored spent nuclear fuel, and the water pool might drain. The impacts for this could be conservatively estimated by summing the consequences. Similarly, consequences from spent nuclear fuel facilities within a DOE site could be combined to conservatively estimate site-wide impacts. But again, the probability of a common-cause event resulting in this number of consequences is lower than the probability of individual accidents because, due to separation distances, the severity of impact will vary between facilities. The existing security measures in effect at the management sites would essentially preclude any sabotage or terrorist activity. Further, any acts of terrorism are expected to result in consequences which are bounded by the results of accidents analyzed. Thus, no specific analyses of the results of terrorist acts were conducted.

Based on the above, the review identified the following bounding accident scenarios:

- criticality caused by human error during operation, equipment failure, or earthquake;
- mechanical damage to foreign research reactor spent nuclear fuel during examination and preparation (cropping off the aluminum and nonfuel end of a fuel); and
- accident involving an impact by either an internal or external initiator with and without an ensuing fire.

F.6.3 Accident Scenarios Considered

A total of six bounding accident scenarios for the handling and storage of foreign research reactor spent nuclear fuel were identified for detailed analysis. Each of these accident scenarios was evaluated at each storage location using identical source terms. As described below, three of the bounding accident scenarios apply to wet storage and three apply to dry storage.

F.6.3.1 Wet Storage Bounding Accident Scenarios

Three hypothetical accident scenarios were evaluated for foreign research reactor spent nuclear fuel stored in water pools: (a) fuel element breach (i.e., cutting into the fuel region) or mechanical damage due to operator error, (b) an accidental criticality, and (c) an aircraft crash into the water pool facility. In addition to these three scenarios, a dropped fuel cask was also considered to be a foreseeable accident. However, as will be seen in Section F.6.4.4.4, the consequences of this accident are bounded by the cutting into a fuel region scenario. Therefore, a dropped fuel cask was not evaluated in detail.

F.6.3.2 Dry Storage Bounding Accident Scenarios

Three hypothetical accidents were evaluated for foreign research reactor spent nuclear fuel handled in dry storage: (a) fuel element breach (i.e., cutting into the fuel region) or mechanical damage during examination work and handling, (b) dropping of a fuel cask, and (c) an aircraft crash with ensuing fire in the dry storage facility. No credible mechanism was identified for an accident criticality in dry storage.

F.6.4 Bounding Accident Evaluation

F.6.4.1 Basic Assumptions

The analysis of airborne releases from hypothetical accidents is performed using the GENII Version 1.485 computer program. Unless otherwise stated, the following conditions were used when performing calculations. In most cases, these are the default conditions in the GENII program.

Meteorological Data:

- Fiftieth- and 95th-percentile meteorological conditions for each storage site were defined using site-specific joint frequency distribution weather data.
- The release is assumed to occur at ground level (0 m).
- Mixing layer height is 1,000 m (3,280 ft). Airborne materials freely diffuse in the atmosphere near ground level in what is known as the mixing depth. A stable layer exists above the mixing depth which restricts vertical diffusion above 1,000 m.
- Wet deposition is zero (it is assumed that no rain occurs to accelerate deposition and reduce the size of area affected by the release).
- Dry deposition of the cloud is modeled. During movement of the radioactive plume, a fraction of the radioactive material in the plume is deposited on the ground due to gravitational forces. The deposited material no longer contributes to the air immersion dose from the plume, but now contributes as exposure from ground surface radiation and ingestion.
- The quantity of deposited radioactive material is proportional to the material particle size and deposition velocities (in m/sec) used in the GENII code as follows:

solids = 0.001	halogens = 0.01	noble gases = 0.0
cesium = 0.001	ruthenium = 0.001	

- If radioactive releases occur through a stack, then additional plume dispersion can be accounted for by considering the beneficial effects of jet plume rise. In this analysis, jet plume rise is ignored.
- When released gases have a heat content, the plume can disperse more quickly. In this calculation, buoyant plume effects are ignored.

Inhalation Data:

- Breathing rate is 330 cm^3/sec (20.1 in^3/sec) for the worker and the NPAI; 270 cm^3/sec (16.5 in^3/sec) for people at the site boundary and beyond (the MEI and the general population).
- Particle size is 1.0 micro-meter (micron).
- The internal exposure period is 50 years for the individual organs and tissues evaluated.
- Exposure during passage of the entire plume is assessed for the MEI and the general public. Exposures to the worker and NPAI are discussed below.
- Inhalation exposure factors are based on International Commission on Radiological Protection Publication 30 (ICRP, 1979-1982).

Mitigating Factors:

For the MEI and members of the general public residing at the site boundary and beyond, no allowances are made for any preventive or mitigative actions that would limit their exposure. These individuals are assumed to be exposed to the contaminated plume during the entire period of its passage, as it travels downwind from the accident site. Similarly, no action is taken to prevent these people from continuing their normal daily routine, including ingestion of the potentially contaminated terrestrial food and animal products. It is assumed, however, that the public would spend approximately 30 percent (about 8 hours) of the day within their homes or other buildings. Therefore, the exposure of the general public to radiation from contaminated ground surface is reduced appropriately. Calculations were done on a yearly basis to determine the effective annual dosage from inhalation, external exposure, and ingestion, and an associated dose commitment extending over a 50-year period from initiation of intake (NRC, 1977a).

Onsite workers would be trained to take quick, decisive action during an accident. These individuals would be trained to quickly evacuate the affected area and move to well-defined "relocation" areas on the facility. Therefore, it is assumed that workers would be exposed to only 5 minutes of the radioactive plume as they move to relocation centers. Once the plume has moved offsite and downwind, the workers would be instructed to walk to vehicles waiting to evacuate them from the site. It is assumed that an additional 15 minutes would be required to evacuate the workers from the contaminated area and, therefore, the workers would receive a total of 20 minutes of exposure to radioactive material deposited on the ground. No ingestion of contaminated foods is assumed for these individuals.

Individuals that may be traversing the site in a vehicle (i.e., NPAI) would be evacuated from the affected area within 2 hours. This is based on the availability of security personnel at all locations to oversee the removal of collocated workers and travelers in a safe and efficient manner. Therefore collocated workers and travelers would be exposed to the entire contaminated plume as it travels downwind for a period not to exceed 2 hours. Similarly, the radiation from the deposited radioactive materials would be limited to a 2-hour period. No ingestion of contaminated foods is assumed for these individuals.

Table F-105 provides the individual exposure times used in the accident analyses presented later in this appendix.

Table F-105 Estimated Individual Exposure Times

Exposure Type	*Worker (100 m)*	*NPAI*	*MEI/General Public*
To Plume	5 min	100% of release time up to 120 min	100% of release time
To Fallout on Ground Surface	20 min	120 min	0.70 yr
To Food	NA	NA	1 yr

F.6.4.2 Source Term

The source term is the amount of respirable radioactive material, in terms of Ci (curies), that are released to the air. The airborne source term is typically estimated by the following five-component linear equation:

Source Term = MAR x DR x ARF x RF x LPF

where:

MAR = Material-at-Risk (g or Ci),

DR = Damage Ratio,

ARF = Airborne Release Fraction (or Airborne Release Rate for continuous release),

RF = Respirable Fraction, and

LPF = Leak Path Factor.

***MAR*:** The MAR is the amount of radionuclides (in g or Ci of activity for each radionuclide) available to be acted upon by a given physical stress (i.e., an accident). The MAR is specific to a given process in the facility of interest. It is not necessarily the total quantity of material present, but is that amount of material in the scenario of interest postulated to be available for release.

***DR*:** This is the fraction of material exposed to the effects of the energy/force/stress generated by the postulated event. For the bounding accident scenarios discussed in this document, the value of DR is assumed to be one (i.e., all exposed material is released), unless otherwise specified.

***ARF*:** This is the fraction of the material that becomes airborne due to the accident. Generic ARF values from DOE sources (Elder et al., 1986; DOE, 1994d) are used in this document unless other values more appropriate to a particular accident scenario are used for ARF. The values for ARF are summarized in Table F-106.

***RF*:** This is the fraction of the material, with particle sizes of 10 micro-meters (microns) or less (DOE, 1994d) that could be retained in the respiratory system following inhalation. The term RF is applied only for the inhalation pathway.

***LPF*:** The LPF accounts for the action of removal mechanisms, such as containment systems, filtration, deposition, etc., to reduce the amount of airborne radioactivity that is ultimately released to occupied spaces of the facility or to the environment. An LPF of 1.0 (i.e., no reduction) is assigned in accident scenarios involving a major failure of confinement barriers.

Table F-106 Release Fractions[a] for Various Release Mechanisms

Material	Release Mechanisms		
	Fuel breach[a]	Fire	Criticality Accident[b]
Gas			1.0
Noble Gas	1.0	1.0	
Krypton	0.3	1.0	
Other Noble Gas	0.1	1.0	
Halogens	0.1	1.0	0.25[d]
Iodine-129	0.25		
Solids			
Volatile	0.01[c]	2.5×10^{-4} [d,e]	
Nonvolatile	0.01	2.5×10^{-6} [d,e]	

Source: DOE, 1995g

[a] *As recommended in Elder et al., 1986.*

[b] *Regulatory Guide values (NRC, 1977b, 1979b, and 1988b).*

[c] *Actually semi-volatile (cesium, rhodium, antimony, selenium, technetium, and tellurium); review on a case-by-case basis.*

[d] *Includes release fraction, respirable fraction and plate-out.*

[e] *Data from DOE, 1995g.*

F.6.4.3 Description of Radiological Accident Scenarios and Generic Parameters

As discussed previously, the accident screening and selection process led to selection of six bounding accident scenarios involving radioactive materials. Appropriate assumptions also have been discussed regarding meteorological parameters, dispersion parameters, dose estimates, and emergency response and protective actions. Each of the accident scenarios is described in the following text according to the major headings listed below:

- Description of Accident,
- Development of Radioactive Source Term, and
- Dose Calculations and Results.

The contents of these sections and a summary of the generic parameters used follow.

Description of Accident provides a basis for accident selection and discusses possible initiating events. A qualitative assessment of scenario likelihood is provided.

Development of Radioactive Source Term describes the assumptions that apply to the development of the resulting source term. Specifically, it discusses the various multipliers (defined earlier in this section) that convert the MAR to the source term.

These multipliers have the following values:

- DR is 1.0, unless otherwise specified.
- ARF is taken from Table F-106, or clearly stated if different.
- LPF is 1.0 for a major failure of confinement barriers.

Dose Calculations and Results relates the computer modeling to the specific accident scenario, and documents the results. Specifically, these subsections accomplish the following:

- describe assumptions and unique input parameters (other than the source term) used in the computer model,
- document the computer model output in terms of exposure to radionuclides for individuals and for the general population within a 80 km (50 mi) radius, and
- assess the potential for health effects.

Unless otherwise specified, the meteorological/dispersion parameters and estimated exposure times summarized are used in the dosimetry calculations for specific accident scenarios. Under some circumstances, facility worker exposures could be either greater or less than these nominal values.

F.6.4.4 Accident Scenario Descriptions and Source Terms

F.6.4.4.1 Fuel Element Breach

Description of Conditions: Fuel element mechanical damage due to handling during examination, such as accidentally cutting into the fuel region, was assessed. This hypothetical accident results from inadvertent cutting across the fuel region when cropping off the aluminum and nonfuel ends of a fuel unit. All noble gas isotopes are postulated to be released to the facility building and escape to the environment. The majority of the volatile and solid nuclides are likely to be retained in the fuel or the facility exhaust filters. The resulting airborne release to the environment was evaluated.

Likelihood: The frequency of this scenario is estimated to be 0.16/yr (DOE, 1995g). This frequency estimate is based on historical operation data (one event in 6 years) for a spent nuclear fuel storage facility. This estimate is conservative for the case of foreign research reactor spent nuclear fuel storage because the majority of the spent nuclear fuel elements are expected to be cropped prior to their emplacement in a transportation cask at a foreign research reactor. Nevertheless, this estimate is retained for the evaluation of the potential risk associated with the handling and preparation of foreign research reactor spent nuclear fuel for storage in both a dry and a wet storage facility.

Source Term: Conditions used in developing the source term are as follows:

- Only one spent nuclear fuel element is damaged. This is because only one spent nuclear fuel element is being handled at a time.

If the spent nuclear fuel cutting accident occurs in a dry cell (dry storage), the following assumptions apply:

- All (100 percent) of the noble gases available for release are released to the atmosphere. Here, it was assumed that all noble gases in an irradiated fuel element would be released. This is conservative, since foreign research reactor fuels are dispersion fuels in which the gaseous fission products are essentially trapped within the fuel matrix. This is different than for commercial reactor fuel, where gaseous fission products collect in the gap between the fuel and its sealed metal fuel rod and are readily released if the rod is damaged.

- Twenty-five percent of the halogens in the spent nuclear fuel are released to the environment. This is also conservative for the reason stated above.
- One percent of the particulate fission products is released to the dry cell from the spent nuclear fuel element, and 99.9 percent removed prior to release to the environment by the normally installed high-efficiency particulate air filters. The use of 99.9 percent efficiency is conservative, since normal efficiency of installed high-efficiency particulate air filters is greater than 99.99 percent.
- Cesium (Cs) and Ruthenium (Ru) behave like particulate fission products.
- The release to the environment occurs at a constant rate over a 15-minute period.

If the spent nuclear fuel cutting accident occurs under water (wet storage) the following assumptions apply:

- All (100 percent) of the noble gases available for release are released to the environment.
- Twenty-five percent of the halogens available for release will be released to the pool, and only 10 percent of this amount will be released to the air. This additional reduction is due to the fact that halogen gases dissolve in the water as they escape (leak out) from the failed fuel. Based on solubility alone, it is expected that all iodines are dissolved in the water pool before they get to the pool surface. In spite of this fact, for the purposes of the analyses, it was assumed that 2.5 percent of halogens available for release will be released to the atmosphere.
- There is no particulate fission product release to the environment. All particulates are retained in the pool water.
- Since only gaseous fission products are released to the air inside the facility, installed high-efficiency particulate air filters would not provide additional reduction in the amount of material released to the environment.
- The release to the environment occurs at a constant rate over a 15-minute period.

F.6.4.4.2 Accidental Criticality

Description of Conditions: In this hypothetical accident scenario, an accidental uncontrolled chain reaction producing 1×10^{19} fissions is postulated. The 10^{19} fission criticality is a very conservative assumption for the spent nuclear fuel pool. This assumption is only applicable to liquid processes (such as uranium reprocessing) as stated in Regulatory Guides 3.33 and 3.34 (NRC, 1979a and 1979b). This criticality is assumed to consist of an initial burst of 10^{18} fissions in 0.5 seconds, followed at 10 minute intervals for the next 8 hours by a burst of 2×10^{17} fissions, for a total of 10^{19} fissions. The total yield for a moderated solid system, as applicable to the spent nuclear fuel in a wet pool, is estimated to be on the order of 10^{18} fissions. This is because the initial criticality will disrupt the critical geometry and no further criticality burst will occur.

The criticality occurs in the water pool and the spent nuclear fuel remains covered in the water. The fission products released include those specified in Regulatory Guide 3.34 (NRC, 1979b) from the criticality over an 8-hour period, plus fission products existing in the fuel as a result of its original use in the foreign research reactor. Removal of fission products by the pool water is considered in the analysis.

Criticality is not considered in the dry storage because the licensing design basis for spent nuclear fuel dry storage design facilities precludes the consideration of any criticality accident by design. The design must demonstrate, through rigorous structural and criticality analyses, that the likelihood of a criticality is incredible or unforeseeable. No effective moderator, such as water, exists in a dry storage design; and, even if flooded, it remains subcritical.

***Likelihood*:** The frequency of this scenario is estimated at 3.1 x 10^{-3} per year (DOE, 1995g). The estimation of this frequency was conservatively based on a statistical evaluation considering that no accidental criticality event with spent nuclear fuel storage has occurred (DuPont, 1983b). This frequency is estimated by considering both the various process-related upset conditions and the natural phenomena hazard (i.e., earthquake and tornadoes) initiated criticality events. The magnitude of fission yield for such a criticality accident was estimated to range from about 5 x 1017 to 1 x 10^{19} fissions. The historical criticality accidents at different DOE facilities dealing with spent nuclear fuels indicate a much smaller fission yield than that evaluated here. The frequency of an accidental criticality of the magnitude evaluated here is estimated to be between one and two orders of magnitude less than the estimated frequency.

***Source Term*:** Conditions used in developing the source term are as follows:

- The fractions of the fission products from damaged spent nuclear fuel elements released to the building are 100 percent of the noble gases, 25 percent of the halogens, 0.1 percent of the Ru, and 0.05 percent of the Cs and remaining solids (NRC, 1977b, 1979b, and 1988b).
- Fission products from 10 spent nuclear fuel elements damaged in the criticality accident are also released in addition to the gaseous fission products created by the criticality event.
- A high-efficiency particulate air filter removes 99.9 percent of the solid fission products that were released to the air inside the facility before they enter the environment.
- The release to the environment occurs at a constant rate over a 15-minute period. This is conservative as compared to the 8-hour release allowed in Regulatory Guide 3.34 (NRC, 1979b).

F.6.4.4.3 Aircraft Crash

Dry Storage:

***Description of Conditions*:** A hypothetical aircraft accident scenario was developed for the dry storage option. This accident is analyzed only at storage sites that have a likelihood of accident occurrence greater than 10^{-7} per year. The consequences of this accident are expected to bound all other dry storage accident scenarios involving an impact that results in fire. The aircraft crash accident is postulated to cause damage to a single transfer container in the dry unloading cell in a modular vault storage facility. Engineering experience indicates that most of the aircraft structure is stopped by the dry storage building structure. Only a heavy dense jet engine rotor shaft is expected to be capable of penetrating the building and damaging the container. Due to the severity of the impact, it was assumed that the cask is breached and the fuel elements in the cask are damaged. The release of fission products occurs due to the impact and resultant fire (i.e., from aviation fuel).

The accident scenario for a dry cask storage facility is similar to that of a modular vault facility. The aircraft crash analysis is the only accident scenario applicable to a dry cask storage. In this scenario, it is expected that the concrete structure which houses the storage canisters is sufficiently rugged that it can survive an aircraft accident with no significant damage to the spent nuclear fuel.

Likelihood: The frequency of this scenario is site dependent. DOE, as part of the Programmatic SNF&INEL Final EIS, has performed calculations of aircraft crash hit frequencies at potential storage sites (i.e., Savannah River Site, Idaho National Engineering Laboratory, Oak Ridge Reservation, Hanford Site, and Nevada Test Site) for naval fuel (DOE, 1995g). The reported crash frequencies are: 2 x 10^{-6} per year for the Savannah River Site, 1 x 10^{-6} per year for the Oak Ridge Reservation, 4 x 10^{-7} per year for Nevada Test Site, 7 x 10^{-8} per year for the Idaho National Engineering Laboratory, and 4 x 10^{-8} per year for the Hanford Site. These frequency estimates were based on the number of commercial air carriers and military aircraft passing within a 10-mile radius of the proposed storage location at these sites. The calculations for the Idaho National Engineering Laboratory also included potential hazards from a nearby airport. These calculations were performed very conservatively, by considering that all the overflights within the 10-mile radius will pass directly over the storage location at each site.

A new assessment of aircraft impact probabilities for the Idaho National Engineering Laboratory chemical processing plant indicates a frequency of aircraft crash into a dry storage facility the size of the IFSF of about 2.6 x 10^{-10} per year from overflights and 3.5 x 10^{-7} per year from airport-related flights near the plant (WINCO, 1994). (The IFSF effective area is five times that considered in the evaluation for the naval fuel storage area, which represents the critical areas containing spent nuclear fuel. Therefore both results are consistent, from the overall crash frequency point of view at the Idaho National Engineering Laboratory.)

In order to provide an understanding of the rationale used in this EIS for this scenario, an overview of the aircraft crash analysis approach is presented. In general, the aircraft crash hit frequency is calculated based on four factors: number of flight operations (takeoff, landing, overflight), aircraft crash rate, facility effective area, and an assumption of crash area distribution. Several models are currently used to estimate the hit frequency. The results of these models are driven by the assumptions regarding the target area and crash area distribution. For example, assuming that overflights (high or low altitude) pass over the facility inherently assumes that the crash area distribution is a straight line. This overestimates the frequency by at least a factor of 10 (approximate width of an airway). In calculating effective area, the analysis considers that an aircraft can hit a facility either directly (falling on the building, footprint area), by skidding into the building (skid area), or in an angular impact (shadow area). Depending on the assumptions of skid length and the angular approach of a crash terminating aircraft, the sum of the latter two areas may contribute between 80 to 95 percent of the total effective area. It is important to note that aircraft that fall vertically with the greatest impact contribute between 1 and 10 percent to the overall crash rate. Therefore, for the majority of cases, the aircraft will hit the ground before it hits the facility.

Based on the above summary, it is considered that frequencies reported in the Programmatic SNF&INEL Final EIS are conservative by at least a factor of 10 for all sites except the Idaho National Engineering Laboratory. Nonetheless, for the purposes of analyses and consistency, this EIS will consider frequencies similar to those used in the Programmatic SNF&INEL Final EIS. The potential aircraft crash frequency at the Oak Ridge Reservation, the Nevada Test Site, the Idaho National Engineering Laboratory, and the Savannah River Site is conservatively set at 10^{-6} per year. This scenario will not be applicable to the Hanford Site, where the estimated frequency is less than 10^{-7} per year.

Source Term: Conditions used in developing the source term are as follows:

- Only one transfer cask containing 20 spent nuclear fuel elements would be damaged by the impact and the resultant fire. This is based on the fact that, if an aircraft hits the building, only the transfer cask is susceptible to damage by the crash. The stored casks are protected by a three-foot concrete shield, and therefore would not be affected by the crash. Based on a conservative estimate of the duration of the transfer operation, the transfer cask could be damaged by the accident only one percent of the time.

- Of the available fission products, 100 percent of the noble gases, 100 percent of the halogens, 2.5 percent of the cesium, and 0.025 percent of the remaining solids are released to the environment. The overall, respirable fractions of fission products released to the environment are consistent with that given in Table F-106 for a fire scenario.
- The release to the environment occurs at a constant rate over a 15-minute period.
- No filtration by high-efficiency particulate air filters is assumed.

For dry cask storage, it was assumed that the ruggedness of the overall dry cask structure is similar to that of a transportation cask. Based on this assumption, the accident source terms were assumed to be similar to that of aluminum-based spent nuclear fuel source terms for the highest severity accident (cask damage and fire) utilized in the RADTRAN accident analysis (DOE, 1995g). The overall source terms for this scenario include: 63 percent of noble gases, 6×10^{-3} percent of halogens, 1×10^{-3} percent of cesium, 2.4×10^{-4} percent of ruthenium, and 1×10^{-5} percent of other solid fission products available in a dry cask.

Wet Storage:

Description of Conditions**:** Impact into water pools by aircraft with resulting damage to the spent nuclear fuel elements stored inside the pool was evaluated. The hypothetical accident might damage the fuel either by the aircraft directly striking it or by the aircraft causing sufficient damage to the building to cause part of the building to collapse and strike the fuel. Fission products are released from the spent nuclear fuel units into the water pool, however, the pool water is not released to the environment. An aircraft crash into a water pool would not produce enough force to cause the pool to leak because the walls of the water pool are constructed of thick reinforced concrete with earth surrounding them, making them very strong. In addition, based on the discussion provided above, it was judged unlikely that an aircraft would impact the water pool at an angle steep enough to expose the floor of the pool or the walls of the pool below the water level to direct impact.

Likelihood**:** The same frequency as discussed above will be used for an aircraft crash into a wet storage facility.

Source Term**:** Conditions used in developing the source term are as follows:

- It was estimated that about 140 spent nuclear fuel elements would damaged. This estimate was based on the consideration of the size of spent nuclear fuel allowing fuel stacking and an assumption that only one percent of the upper stacked fuel will be damaged.
- Of the available fission products, 100 percent of the noble gases and 25 percent of the halogens are released to the pool water. Due to the presence of pool water, a reduction of the halogen release by a factor of 10 occurs prior to release to the environment.
- The pool water is not expected to be lost and the solid fission products from ruptured and damaged fuel elements remain in the water. However, for the purposes of this analysis, it was conservatively assumed that 0.01 percent of the solid fission products (including Cs and Ru) released from the damaged fuel elements to the pool would be displaced upon

impact. Only one percent of released solid fission products would become airborne and released to the environment. This assumption considers that, upon impact, a percentage of the spent nuclear fuel fails, the solid fission products enter the pool, and only finely crushed particulates are splashed out of the pool in the same timeframe that the aircraft hits the water.

- The release to the environment occurs at a constant rate over a 15-minute period.
- Spent nuclear fuel elements remained covered in the water pool.
- The building confinement is assumed to have failed; no filtration by high-efficiency particulate air filters is assumed.

F.6.4.4.4 Fuel Cask Drop

Dry Storage:

Description of Conditions: Mechanical damage due to handling during examination, such as dropping of the spent nuclear fuel cask during transfer, was assessed. The fuel casks are certified to result in no failure for a specific drop height, (free drop from 9 m [30 ft] height onto an unyielding surface), and under no circumstances will the cask be moved above such height during operations within a storage facility. Nevertheless, it was assumed that, upon cask drop, the seals of the cask would fail, releasing the gaseous fission products from the damaged fuel inside the cask to the facility building and the environment. All of the nonvolatile and solid nuclides are assumed to be retained in the fuel or the facility high-efficiency particulate air filters. The resulting airborne release to the environment was evaluated.

Likelihood: The frequency of this scenario is estimated at 10^{-4} per year (DOE, 1995g). This estimate is considered to be an upper bound for this scenario.

Source Term: Conditions used in developing the source term are as follows:

- Only one fuel cask is involved. This is because only one fuel cask is being handled at a time. For the purposes of this analysis, it was assumed that an equivalent of one spent nuclear fuel element inside the cask is damaged, and its gaseous fission products are released inside the cask. This assumption is conservative, since the fuel is secured inside the cask and the cask is not expected to be damaged.
- All (100 percent) of the gaseous fission products and 25 percent of halogens from the damaged fuel element are released to the atmosphere.
- None of the particulate fission products are released to the environment.
- Cs and Ru behave like particulate fission products.
- The release to the environment occurs at a constant rate over a 15-minute period.

Wet Storage:

The source term for a fuel cask drop is similar to that for the fuel element breach scenario in a wet storage facility. The gaseous fission products released inside the cask are vented under water (or in the pool). Since the estimated frequency of this scenario is less than that of the fuel element breach, no specific analysis for this scenario was performed.

F.6.5 Incident-Free Operation Source Terms

This section details the assumptions and the evaluation process used to determine the risk of radiological emissions generated during different activities in incident-free operation of a storage facility. The incident-free operation emissions consist of two parts: transient (i.e., emissions from gaseous release during receipt and unloading of the transportation casks), and steady state (i.e., emissions from spent nuclear fuel in storage). Since only mechanically sound spent nuclear fuel elements are shipped, no radioactive releases are expected during transit. To ensure this, the spent nuclear fuel elements are checked prior to shipment to identify and separate any damaged fuel elements. The damaged fuel elements are then encapsulated and prepared for shipment. In spite of the fact that no spent nuclear fuel elements have ever failed during transit, it was assumed that one percent of the spent nuclear fuel elements will arrive failed and release gaseous fission products (noble gases and halogens) into the cask. Depending on the type of storage facility, the receipt and unloading of the transportation casks could occur in a dry cell or a wet pool. Unloading operations in a dry cell causes all gaseous fission products to be released to the building and eventually to the environment. If the unloading process occurs in a wet pool, a majority of the halogen gases will be absorbed in the water; only 10 percent of halogens will be released to the environment. The building high-efficiency particulate air filters will not be effective for halogens and noble gases. During the unloading process, all spent nuclear fuel elements are checked to ensure that they are mechanically sound. If a damaged fuel element is found, it is encapsulated in a can before it is placed in wet or dry storage. The potential annual radiological releases from failed fuel elements during the unloading process were estimated based on the gaseous inventories of bounding fuels (see Appendix B, Section B.1.4) ad the associated number of fuels expected over the acceptance period. The receipt and unloading process of foreign research reactor spent nuclear fuel from abroad is expected to last 13 years (see Section 2.2.1). It was assumed that failed fuel would release 100 percent of its noble gases and 25 percent of its halogens. This assumption is consistent with that used in the accident analysis.

The steady state emissions from a new wet storage facility are assumed to be similar to those released from the RBOF facility at the Savannah River Site. Although the emissions at the RBOF facility may not be a good representation of the foreign research reactor spent nuclear fuel, RBOF has the most foreign research reactor spent nuclear fuel elements stored in its pool; and as such, was considered to provide the best approximation of the expected release. Based on the emission data from RBOF, the steady-state emissions from a wet storage facility are assumed to be about 2 x 10^{-7} curies of Cesium-137 per year (DOE, 1995g). This is a conservative assumption. For existing wet storage facilities, the radiation exposure to the MEI and the general public were estimated based on the combined radionuclide atmospheric emissions originating from current conditions of the facilities and that expected from foreign research reactor spent nuclear fuel. At Savannah River Site, the average annual atmospheric emissions from the existing fuels at L-reactor disassembly basin are estimated to be 254 curies of tritium and 6.49 x 10^{-5} curies of Cesium-137 (Shedrow, 1994b) over Phase 1 of the policy period. The assumption is that the foreign research reactor spent nuclear fuel would be stored temporarily (about 10 years) in the wet pool until a more permanent dry storage facility is built. The annual atmospheric radiological emissions from RBOF and BNFP wet pools are similar to those that are currently released from RBOF and which were used for a new facility. The annual atmospheric radiological emissions from the Idaho National Laboratory's FAST wet storage facility were assumed to be similar to that of a new wet storage facility. This facility has been designed and built according to current codes and regulations.

The steady-state emissions from a dry storage facility are considered to be zero. This is because the fuel will be checked to ensure that it is mechanically sound (i.e., no damage) before it is placed into dry storage, and the dry storage canisters that house the fuel are sealed.

F.6.6 Dose Calculations and Results

F.6.6.1 Source Terms

Tables F-107 and F-108 provide the incident-free operation and accident source terms. The source terms for the annual emissions from the unloading process were calculated based on the assumption that a constant annual rate of fuel mix with bounding radionuclide inventories (as defined in Appendix B, Section B.1) is received over the acceptance period. The fission products in a BR-2 type spent nuclear fuel element were used as the MAR in the accident analysis source term calculations (see Appendix B for more details). Four fuel categories were defined in Appendix B: BR-2, NRU, RHF, and TRIGA. BR-2 fuel type constitutes the majority of the foreign research reactor spent nuclear fuels. In addition, since spent nuclear fuels come in different sizes and lengths, use of the BR-2 spent nuclear fuel in the accident analysis means involvement of a larger number of spent nuclear fuels in each accident. For example, in the source term calculations for an aircraft crash accident involving a transfer cask, it was assumed that the cask would contain 20 BR-2 spent nuclear fuel elements. If the cask contained NRU elements, there would be five elements in the cask; and if it contained RHF elements, there would be only four elements per cask. For the generic wet storage case, the bounding spent nuclear fuel is considered to have been cooled at least 300 days prior to shipment. In the case of dry storage, the fuel has been cooled for at least 3 years.

Table F-107 Annual Emission Releases From Storage Facilities

Isotope	Releases During Unloading in		Steady State	
	Dry Cell (Ci)	Wet Pool (Ci)	Dry Storage (Ci)	Wet Storage (Ci)
Tritium	39.6	39.6	---	---
Krypton-85	1.14×10^{3}	1.14×10^{3}	---	---
Iodine-129	4.87×10^{-4}	4.87×10^{-5}	---	---
Iodine-131	9.12×10^{-4}	9.12×10^{-5}	---	---
Xenon-131	2.01×10^{-5}	2.01×10^{-5}	---	---
Cesium-137	---	---	0.0	2.20×10^{-7}

Table F-108 Accident Source Terms (Curies)

	Dry Storage[a]			Wet Storage		
	Fuel Element	Dropped	Aircraft Crash	Accident	Fuel Element	

	Dry Storage[a]			Wet Storage		
	Fuel Element	Dropped	Aircraft Crash	Accident	Fuel Element	

The incident-free operation source terms for chemical separation at the Savannah River Site and the Idaho National Engineering Laboratory were taken from the Interim Management of Nuclear Materials Final EIS (DOE, 1995b) and the Programmatic SNF&INEL Final EIS (DOE, 1995g), respectively. Accident source terms for the chemical separation process were not developed for foreign research reactor spent nuclear fuel. It was considered that the consequences of chemical separation operations-related accidental scenarios are similar to those identified and analyzed in the above documents.

F.6.6.2 Site-Specific Parameters

Several site-specific parameters were required as input to the computer models. The site-specific parameters deal with meteorology, individual and general population food consumption rates, food production locations, and distances and directions of individuals and populations with respect to release locations. The food consumption rates apply only to the MEI and the population dose calculations as indicated in Table F-105. Site-specific food consumption rates consistent with those used in the Programmatic SNF&INEL Final EIS (DOE, 1995g) were utilized. Different contaminated food consumption rates were used at each site because the rate at each site is calculated based on the food production rate within an 80 km (50 mi) radius and the amount of supplemental food (uncontaminated food) that is imported from outside of the 80 km (50 mi) radius. If food production around the site is not sufficient for the population consumption rate, then uncontaminated food is imported. Otherwise, the consumed food is assumed to be contaminated.

F.6.6.3 Results

Tables F-109 through F-116 provide summaries of the consequences, in terms of mrem and/or person-rem, of postulated accident doses to the MEI, NPAI, worker and the public. Except for the worker, where the dose is calculated using the 50th-percentile meteorology, dose calculations were performed for both the 50th- and the 95th-percentile meteorologies using the assumptions and input values discussed above. The accident scenarios and source terms, as described earlier in this appendix, were generically applied to new dry and wet storage facilities. For the existing facilities at each management site, the assumptions and the related source terms were adjusted to conform to the conditions of each facility. Two types of results were provided for the offsite residents (MEI and population). Because protective action guidelines (EPA, 1991) specify mitigative actions to prevent consumption of contaminated food, the dose to offsite residents is reported for all pathways (i.e., external, inhalation, and ingestion) and without the ingestion pathway (i.e., external and inhalation). It should be noted that, as stated earlier, no reduction of exposure to the plume or to contaminated ground surface as a result of early evacuation of offsite populations due to protective action guidelines was accounted for in this analysis.

The analyses were performed for a generic wet and a generic dry storage facility at the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site, as well as for site-specific locations (BNFP, L-Reactor Basin area, and RBOF at the Savannah River Site, and FMEF and WNP-4 Spray Pond at the Hanford Site). The consequences of accident scenarios for the IFSF (dry), CPP-749 (dry) and FAST (wet) storage areas at the Idaho National Engineering Laboratory are considered to be equal to those of a generic dry and a generic wet storage facility, respectively. The consequences of accident scenarios for E-MAD at Nevada Test Site are considered to be similar to that of a generic dry storage facility at Nevada Test Site. For the RBOF and the L-Reactor disassembly basin, the criticality accident source terms were adjusted to conform with the conditions assumed in the Basis for Interim Operation reports for these facilities (WSRC, 1995b and 1995c), where no credit was taken for high efficiency particulate air filters after a criticality accident.

Table F-109 Summary of the Accident Analysis Dose Assessments at the Savannah River Site Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
Dry Storage Accidents - H-Area									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	0.24 0.038 1.9×10^{-8}	0.068 0.011 5.5×10^{-9}	9.2 1.5 0.00075	0.055 0.0088 4.4×10^{-9}	0.0043 0.00069 3.5×10^{-10}	28 4.5 1.8×10^{-6}	0.62 0.099 0.000050
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF[b]	0.018 1.8×10^{-6} 9.0×10^{-13}	0.00034 3.4×10^{-8} 1.7×10^{-14}	0.55 0.000055 2.8×10^{-8}	0.0039 3.9×10^{-7} 2.0×10^{-13}	0.000024 2.4×10^{-9} 1.2×10^{-15}	0.28 0.000028 1.1×10^{-11}	0.011 1.1×10^{-6} 5.5×10^{-10}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event Dose/yr LCF[b]	40 0.000040 2.0×10^{-11}	0.29 2.9×10^{-7} 1.5×10^{-13}	1300 0.0013 6.5×10^{-7}	8.9 8.9×10^{-6} 4.5×10^{-12}	0.019 1.9×10^{-8} 9.5×10^{-15}	120 0.00012 4.8×10^{-11}	87 0.000087 4.4×10^{-8}
New Wet Storage Accidents - H-Area									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	0.0070 0.0011 5.5×10^{-10}	0.00039 0.000062 3.1×10^{-11}	0.23 0.037 0.000019	0.0016 0.00026 1.3×10^{-10}	0.000027 4.3×10^{-6} 2.2×10^{-12}	0.14 0.0022 8.8×10^{-10}	0.016 0.00026 1.3×10^{-7}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF[b]	17 0.053 2.7×10^{-8}	9.5 0.030 1.5×10^{-8}	370 1.2 0.00060	4.0 0.012 6.0×10^{-9}	0.69 0.0021 1.1×10^{-9}	1600 5.0 2.0×10^{-6}	15 0.047 0.000024
Aircraft Crash	1×10^{-6}	Dose/event Dose/yr LCF[b]	4.1 4.1×10^{-6} 2.1×10^{-12}	0.98 9.8×10^{-7} 4.9×10^{-13}	150 0.00015 7.5×10^{-8}	0.92 9.2 x 10-7 4.6×10^{-13}	0.061 6.1×10^{-8} 3.1×10^{-14}	400 0.00040 1.6 x 10-10	10 0.000010 5.0×10^{-9}

Table F-109A Summary of the Accident Analysis Dose Assessments at the Savannah River Site Generic Storage Facilities - External and Inhalation Pathways

Table F-110 Summary of the Accident Analysis Dose Assessments at the Idaho National Engineering Laboratory Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	1.3	0.67	15	0.093	0.062	28	0.83
		Dose/yr	0.21	0.11	2.4	0.015	0.0099	4.5	0.13
		LCF[b]	1.1×10^{-7}	5.5×10^{-8}	0.0012	7.5×10^{-9}	5.0×10^{-9}	1.8×10^{-6}	0.000065
Dropped Fuel Cask	0.0001	Dose/event	0.074	0.0033	0.83	0.0052	0.00032	0.12	0.047
		Dose/yr	7.4×10^{-6}	3.3×10^{-7}	0.000083	5.2×10^{-7}	3.2×10^{-8}	0.000012	4.7×10^{-6}
		LCF	3.7×10^{-12}	1.7×10^{-13}	4.2×10^{-8}	2.6×10^{-13}	1.6×10^{-14}	4.8×10^{-12}	2.4×10^{-9}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	180	2.9	2000	13	0.27	120	110
		Dose/yr	0.00018	2.9×10^{-6}	0.0020	0.000013	2.7×10^{-7}	0.00012	0.00011
		LCF	9.0×10^{-11}	1.5×10^{-12}	1.0×10^{-6}	6.5×10^{-12}	1.4×10^{-13}	4.8×10^{-11}	5.5×10^{-8}
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	0.0016	0.0036	0.43	0.0028	0.00036	0.14	0.025
		Dose/yr	0.00026	0.00058	0.069	0.00045	0.000058	0.022	0.0040
		LCF	1.3×10^{-10}	2.9×10^{-10}	0.000035	2.3×10^{-10}	2.9×10^{-11}	8.8×10^{-9}	2.0×10^{-6}
Accidental Criticality	0.0031	Dose/event	28	30	140	3.4	12	1800	12
		Dose/yr	0.087	0.093	0.43	0.011	0.037	5.6	0.037
		LCF	4.4×10^{-8}	4.7×10^{-8}	0.00022	5.5×10^{-9}	1.9×10^{-8}	2.2×10^{-6}	0.000019
Aircraft Crash	1×10^{-6}	Dose/event	22	9.8	250	1.6	0.88	400	14
		Dose/yr	0.000022	9.8×10^{-6}	0.00025	1.6×10^{-6}	8.8×10^{-7}	0.00040	0.00014
		LCF	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	8.0×10^{-13}	4.4×10^{-13}	1.6×10^{-10}	7.0×10^{-8}

Table F-110A Summary of the Accident Analysis Dose Assessments at the Idaho National Engineering Laboratory Generic Storage Facilities - External and Inhalation Pathways

	95th-Percentile Meteorology	*50th-Percentile Meteorology*

Table F-111 Summary of the Accident Analysis Dose Assessments at the Hanford Site Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	3.0 0.48 2.4×10^{-7}	0.57 0.091 4.6×10^{-8}	42 6.7 0.0034	0.15 0.024 1.2×10^{-8}	0.061 0.0098 4.9×10^{-9}	50 8.0 3.2×10^{-6}	2.0 0.32 0.00016
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.26 0.000026 1.3×10^{-11}	0.0085 8.5×10^{-7} 4.3×10^{-13}	3.0 0.00030 1.5×10^{-7}	0.011 1.1×10^{-6} 5.5×10^{-13}	0.00031 3.1×10^{-8} 1.6×10^{-14}	0.22 0.000022 8.8×10^{-12}	0.15 0.000015 7.5×10^{-9}
Aircraft Crash w/Fire[c]	NA	---	NA	NA	NA	NA	NA	NA	NA
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.13 0.021 1.1×10^{-8}	0.0033 0.00053 2.7×10^{-10}	1.6 0.26 0.00013	0.0064 0.0010 5.0×10^{-10}	0.00035 0.000056 2.8×10^{-11}	0.25 0.040 1.6×10^{-8}	0.078 0.013 6.5×10^{-6}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	64 0.20 1.0×10^{-7}	14 0.044 2.2×10^{-8}	740 2.3 0.0012	4.8 0.015 7.5×10^{-9}	12 0.037 1.9×10^{-8}	3600 11 4.4×10^{-6}	55 0.17 0.000085
Aircraft Crash[c]	NA	---	NA	NA	NA	NA	NA	NA	NA

Table F-111A Summary of the Accident Analysis Dose Assessments at the Hanford Site Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	0.30 0.048 2.4×10^{-8}	6.5 1.0 0.00050	0.015 0.0024 1.2×10^{-9}	0.31 0.050 0.000025
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.0039 3.9×10^{-7} 2.0×10^{-13}	0.029 2.9×10^{-6} 1.5×10^{-9}	0.000071 7.1×10^{-9} 3.6×10^{-15}	0.0015 1.5×10^{-7} 7.5×10^{-11}
Aircraft Crash w/Fire[c]	NA	---	NA	NA	NA	NA
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.0016 0.00026 1.3×10^{-10}	0.032 0.0051 2.6×10^{-6}	0.000079 0.000013 6.5×10^{-12}	0.0018 0.00029 1.5×10^{-7}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	7.9 0.025 1.3×10^{-8}	180 0.56 0.00028	2.0 0.0062 3.1×10^{-9}	27 0.084 0.000042
Aircraft Crash[c]	NA	---	NA	NA	NA	NA

NA = Not Applicable

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

[c] *Aircraft crash accidents are not applicable to the Hanford Site since their frequency of occurrence is less than 10^{-7}/yr.*

Table F-112 Summary of the Accident Analysis Dose Assessments at the Oak Ridge Reservation Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			50th-Percentile Meteorology			
	Frequency (event/yr)	Risk	MEI (mrem)[a]	NPAI (mrem)	Population (person-rem)	MEI (mrem)	NPAI (mrem)	Worker (mrem)	Population (person-rem)
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	22 3.5 1.8×10^{-6}	42 6.7 3.4×10^{-6}	55 8.8 0.0044	2.1 0.34 1.7×10^{-7}	9.4 1.5 7.5×10^{-7}	140 22 8.8×10^{-6}	8.4 1.3 0.00065
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	1.4 0.00014 7.0×10^{-11}	0.18 0.000018 9.0×10^{-12}	15 0.0015 7.5×10^{-7}	0.14 0.000014 7.0×10^{-12}	0.042 4.2×10^{-6} 2.1×10^{-12}	0.61 0.000061 2.4×10^{-11}	2.3 0.00023 1.2×10^{-7}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event Dose/yr LCF	2300 0.0023 1.2×10^{-9}	180 0.00018 9.0×10^{-11}	2900 0.0029 1.5×10^{-6}	220 0.00022 1.1×10^{-10}	41 0.000041 2.1×10^{-11}	610 0.00061 2.4×10^{-10}	440 0.00044 2.2×10^{-7}
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.71 0.11 5.5×10^{-8}	0.20 0.0032 1.6×10^{-8}	16 2.6 0.0013	0.068 0.011 5.5×10^{-9}	0.046 0.0074 3.7×10^{-9}	0.68 0.11 4.4×10^{-8}	2.5 0.40 0.00020
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	1500 4.7 2.4×10^{-6}	3300 10 5.0×10^{-6}	1400 4.3 0.0022	230 0.71 3.6×10^{-7}	910 2.8 1.4×10^{-6}	6800 21 8.4×10^{-6}	210 0.65 0.00033
Aircraft Crash	1×10^{-6}	Dose/event Dose/yr LCF	380 0.00038 1.9×10^{-10}	600 0.00060 3.0×10^{-10}	2900 0.0029 1.5×10^{-6}	29 0.000029 1.5×10^{-10}	130 0.00013 6.5×10^{-11}	1900 0.0019 7.6×10^{-10}	120 0.00012 6.0×10^{-8}

Table F-112A Summary of the Accident Analysis Dose Assessments at the Oak Ridge Reservation Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		50th-Percentile Meteorology	
	Frequency (event/yr)	Risk	MEI (mrem)[a]	Population (person-rem)	MEI (mrem)	Population (person-rem)
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	9.8 1.6 8.0×10^{-7}	29 4.6 0.0023	0.96 0.15 7.5×10^{-8}	4.4 0.70 0.00035
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.038 3.8×10^{-6} 1.9×10^{-12}	0.13 0.000013 6.5×10^{-9}	0.0039 3.9×10^{-7} 2.0×10^{-13}	0.021 2.1×10^{-6} 1.1×10^{-9}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event Dose/yr LCF	180 0.00018 9.0×10^{-11}	500 0.00050 2.5×10^{-7}	17 0.000017 8.5×10^{-12}	76 0.000076 3.8×10^{-8}
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.042 0.0067 3.4×10^{-9}	0.14 0.022 0.000011	0.0043 0.00069 3.5×10^{-10}	0.023 0.0037 1.9×10^{-6}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	1100 3.4 1.7×10^{-6}	1100 3.4 0.0017	180 0.56 2.8×10^{-7}	150 0.47 0.00024
Aircraft Crash	1×10^{-6}	Dose/event Dose/yr LCF	140 0.00014 7.0×10^{-11}	420 0.00042 2.1×10^{-7}	13 0.000013 6.5×10^{-12}	61 0.000061 3.1×10^{-8}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-113 Summary of the Accident Analysis Dose Assessments at the Nevada Test Site Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			50th-Percentile Meteorology			
	Frequency (event/yr)	Risk	MEI (mrem)[a]	NPAI (mrem)	Population (person-rem)	MEI (mrem)	NPAI (mrem)	Worker (mrem)	Population (person-rem)
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	1.7 0.27 1.4×10^{-7}	0.31 0.050 2.5×10^{-8}	1.5 0.24 0.00012	0.052 0.0083 4.2×10^{-9}	0.0046 0.00074 3.7×10^{-10}	20 3.2 1.3×10^{-6}	0.038 0.0060 3.0×10^{-9}
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.11 0.000011 5.5×10^{-12}	0.0014 1.4×10^{-7} 7.0×10^{-14}	0.40 0.000040 2.0×10^{-8}	0.0033 3.3×10^{-7} 1.7×10^{-13}	0.000026 2.6×10^{-9} 1.3×10^{-15}	0.089 8.9×10^{-6} 3.6×10^{-12}	0.010 1.0×10^{-6} 5.0×10^{-10}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event Dose/yr LCF	180 0.00018 9.0×10^{-11}	1.2 1.2×10^{-6} 6.0×10^{-13}	250 0.00025 1.3×10^{-7}	5.6 5.6×10^{-6} 2.8×10^{-12}	0.020 2.0×10^{-8} 1.0×10^{-14}	87 0.000087 3.5×10^{-11}	6.2 6.2×10^{-6} 3.1×10^{-9}
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.054 0.0086 4.2×10^{-9}	0.0016 0.00026 1.3×10^{-10}	0.33 0.053 0.000026	0.0017 0.00027 1.4×10^{-10}	0.000029 4.6×10^{-6} 2.3×10^{-12}	0.10 0.016 6.4×10^{-9}	0.0084 0.0013 6.5×10^{-7}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	88 0.27 1.4×10^{-7}	15 0.047 2.3×10^{-8}	54 0.17 0.000084	6.9 0.021 1.1×10^{-8}	1.1 0.0034 1.7×10^{-9}	1300 4.0 0.000016	1.9 0.0059 3.0×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event Dose/yr LCF	29 0.000029 1.5×10^{-11}	4.2 4.2×10^{-6} 2.1×10^{-12}	61 0.000061 3.1×10^{-8}	0.92 9.2×10^{-7} 4.6×10^{-13}	0.067 6.7×10^{-8} 3.4×10^{-14}	290 0.00029 1.2×10^{-10}	1.6 1.6×10^{-6} 8.0×10^{-10}

Table F-113A Summary of the Accident Analysis Dose Assessments at the Nevada Test Site Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		50th-Percentile Meteorology	
	Frequency (event/yr)	Risk	MEI (mrem)[a]	Population (person-rem)	MEI (mrem)	Population (person-rem)
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	0.78 0.13 6.2×10^{-8}	0.26 0.042 0.000021	0.024 0.0038 1.9×10^{-9}	0.0066 0.0011 5.3×10^{-7}
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.0031 3.1×10^{-7} 1.6×10^{-13}	0.0011 1.1×10^{-7} 5.5×10^{-11}	0.00011 1.1×10^{-8} 5.5×10^{-15}	0.000033 3.3×10^{-9} 1.7×10^{-12}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event Dose/yr LCF	13 0.000013 6.5×10^{-12}	4.5 4.5×10^{-6} 2.3×10^{-9}	0.41 4.1×10^{-7} 2.1×10^{-13}	0.12 1.2×10^{-7} 6.0×10^{-11}
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.0036 0.00058 2.9×10^{-10}	0.0013 0.00021 1.1×10^{-7}	0.00012 0.000019 9.5×10^{-12}	0.000037 5.9×10^{-6} 3.0×10^{-9}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	55 0.17 8.5×10^{-8}	5.4 0.017 8.5×10^{-6}	5.8 0.018 9.0×10^{-9}	0.70 0.0022 1.1×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event Dose/yr LCF	11 0.000011 5.5×10^{-12}	3.7 3.7×10^{-6} 1.9×10^{-9}	0.35 3.5×10^{-7} 1.8×10^{-13}	0.096 9.6×10^{-8} 4.8×10^{-11}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-114 Summary of the Accident Analysis Dose Assessments at the Barnwell Nuclear Fuels Plant Wet Storage Facility[a] at the Savannah River Site - All Pathways

Table F-115 Summary of the Accident Analysis Dose Assessments at the Receiving Basin for Offsite Fuels and L-Reactor Basin Wet Storage Facilities at the Savannah River Site-All Pathways

	95th-Percentile Meteorology			50th-Percentile Meteorology		
						Population

Table F-116 Summary of the Accident Analysis Dose Assessments for the Fuel Material Examination Facility Dry Storage and WNP-4 Wet Storage Facilities at the Hanford Site - All Pathways

			95th-Percentile Meteorology			50th-Percentile Meteorology			
	Frequency (event/yr)	Risk	MEI (mrem)[a]	NPAI (mrem)	Population (person-rem)	MEI (mrem)	NPAI (mrem)	Worker (mrem)	Population (person-rem)
Dry Storage Accidents at FMEF[b]									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[c]	4.7 0.75 3.7×10^{-7}	2.1 0.34 1.7×10^{-7}	46 7.4 0.0037	0.42 0.067 3.4×10^{-8}	0.25 0.040 2.0×10^{-8}	0.99 0.16 6.4×10^{-8}	5.7 0.91 0.00046
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.2 0.00002 8×10^{-12}	0.032 3.2×10^{-6} 1.6×10^{-12}	3.2 0.00032 3.2×10^{-7}	0.017 1.7×10^{-6} 8.5×10^{-13}	0.0017 1.7×10^{-7} 8.5×10^{-14}	0.0049 4.9×10^{-7} 2.5×10^{-13}	0.41 0.000041 2.1×10^{-8}
Aircraft Crash w/Fire[d]	NA	---	NA	NA	NA	NA	NA	NA	NA
Wet Storage Accidents at WNP-4[b]									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.15 0.024 1.2×10^{-8}	0.0033 0.00053 2.7 x 10-10	1.3 0.21 0.00011	0.018 0.0029 1.5×10^{-9}	0.00060 0.000096 4.8×10^{-11}	0.00024 0.000038 1.5×10^{-11}	0.13 0.021 0.000011
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	97 0.3 1.5×10^{-7}	76 0.24 1.2×10^{-7}	620 1.9 0.00096	20 0.062 3.1×10^{-8}	45 0.14 7.0×10^{-8}	120 0.37 1.5×10^{-7}	160 0.50 0.00025
Aircraft Crash[d]	NA	---	NA	NA	NA	NA	NA	NA	NA

Table F-116A Summary of the Accident Analysis Dose Assessments for the Fuel Material Examination Facility Dry Storage and WNP-4 Wet Storage Facilities at the Hanford Site - External and Inhalation Pathways

			95th-Percentile Meteorology		50th-Percentile Meteorology	
	Frequency (event/yr)	Risk	MEI (mrem)[a]	Population (person-rem)	MEI (mrem)	Population (person-rem)
Dry Storage Accidents at FMEF[b]						
Fuel Assembly Breach	0.016	Dose/event Dose/yr LCFc	0.46 0.074 3.7×10^{-8}	6.6 1.1 0.00055	0.041 0.0066 3.3×10^{-9}	0.79 0.12 0.000060
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.0028 2.8×10^{-7} $1 4 \times 10^{-13}$	0.04 4.0×10^{-6} 2.0×10^{-9}	0.00025 2.5×10^{-8} 1.2×10^{-14}	0.0057 5.7×10^{-7} 2.9×10^{-10}
Aircraft Crash w/Fire[d]	NA	---	NA	NA	NA	NA
Wet Storage Accidents at WNP-4[b]						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.0023 0.00037 1.8×10^{-9}	0.032 0.0051 2.6×10^{-6}	0.00028 0.000045 2.2×10^{-11}	0.0034 0.00054 2.7×10^{-7}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	32 0.099 5.0×10^{-9}	180 0.56 0.00028	12 0.037 1.9×10^{-8}	120 0.37 0.00019
Aircraft Crash[d]	NA	---	NA	NA	NA	NA

NA = Not Applicable

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Emissions will be released through an elevated stack for Fuel Assembly Breach, Dropped Fuel Cask, and Accidental Criticality Accidents.*

[c] *Point Estimate of Latent Cancer Fatalities event/yr.*

[d] *Aircraft Crash accidents are not applicable to the Hanford Site since their frequency of occurrence is less than 10^{-7} event/yr.*

Table F-117 provides a summary of the consequences of radiation exposure to the public and to the MEI from emissions in wet storage (generic and existing), and dry storage (generic and existing).

Table F-117 Normal Release Dose Assessments and Latent Cancer Fatalities at Storage Sites

	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Savannah River Site				
Receipt/Unloading at:				
RBOF	1.1×10^{-4}	5.5×10^{-11}	5.7×10^{-3}	2.8×10^{-6}
L-Reactor Basin	7.3×10^{-5}	3.7×10^{-11}	4.6×10^{-3}	2.3×10^{-6}
BNFP	6.5×10^{-4}	3.3×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
New Dry Storage Facility	1.8×10^{-4}	9.0×10^{-11}	8.6×10^{-3}	4.3×10^{-6}
New Wet Storage Facility	1.1×10^{-4}	5.5×10^{-11}	5.7×10^{-3}	2.8×10^{-6}
Storage at:				
RBOF	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
L-Reactor Basin[a]	3.6×10^{-4}	1.8×10^{-10}	2.2×10^{-2}	1.1×10^{-5}
BNFP	7.5×10^{-9}	3.8×10^{-15}	4.8×10^{-8}	2.4×10^{-11}
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
Idaho National Engineering Laboratory				
Receipt/Unloading at:				
IFSF (dry storage)	5.6×10^{-4}	2.8×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
FAST (wet storage)	3.8×10^{-4}	1.9×10^{-10}	3.1×10^{-3}	1.6×10^{-6}
CPP-749 (dry storage)	5.6×10^{-4}	2.8×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
New Dry Storage Facility	5.6×10^{-4}	2.8×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
New Wet Storage Facility	3.8×10^{-4}	1.9×10^{-10}	3.1×10^{-3}	1.6×10^{-6}
Storage at:				
IFSF (dry storage)	0	0	0	0
FAST (wet storage)	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
CPP-749 (dry storage)	0	0	0	0
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
Hanford Site				
Receipt/Unloading at:				
FMEF (dry storage)	2.0×10^{-4}	1.0×10^{-10}	1.1×10^{-2}	5.5×10^{-6}
WNP-4 Spray Pond (wet storage)	2.2×10^{-4}	1.1×10^{-10}	5.8×10^{-3}	2.9×10^{-6}
New Dry Storage Facility	2.5×10^{-4}	1.3×10^{-10}	1.5×10^{-2}	7.5×10^{-6}
New Wet Storage Facility	2.0×10^{-4}	1.0×10^{-10}	1.2×10^{-2}	6.0×10^{-6}
Storage at:				
FMEF (dry storage)	0	0	0	0
WNP-4 Spray Pond (wet storage)	5.9×10^{-10}	3.0×10^{-16}	1.6×10^{-8}	8.0×10^{-12}
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	8.8×10^{-10}	4.4×10^{-16}	6.9×10^{-8}	3.5×10^{-11}
Oak Ridge Reservation				
Receipt/Unloading at:				
New Dry Storage Facility	8.9×10^{-2}	4.5×10^{-8}	8.5×10^{-2}	4.3×10^{-5}
New Wet Storage Facility	6.0×10^{-2}	3.0×10^{-8}	6.1×10^{-2}	3.1×10^{-5}
Storage at:				
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	4.6×10^{-7}	2.3×10^{-13}	5.0×10^{-7}	2.5×10^{-10}

	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
		Nevada Test Site		
Receipt/Unloading at:				
E-MAD (dry storage)	7.6×10^{-4}	3.8×10^{-10}	9.3×10^{-4}	4.7×10^{-7}
New Dry Storage Facility	7.6×10^{-4}	3.8×10^{-10}	9.3×10^{-4}	4.7×10^{-7}
New Wet Storage Facility	5.2×10^{-4}	2.6×10^{-10}	5.2×10^{-4}	2.6×10^{-7}
Storage at:				
E-MAD (dry storage)	0	0	0	0
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	4.0×10^{-9}	2.0×10^{-15}	4.7×10^{-9}	2.0×10^{-12}

[a] *L-Reactor basin doses are due to existing conditions; the foreign research reactor spent nuclear fuel contribution would be six orders of magnitude smaller*

F.6.7 Accident Scenarios Involving Target Materials

A review of the hypothetical accident scenarios analyzed for spent nuclear fuel indicates that only the aircraft crash with fire accident is applicable to the target materials. The frequency of occurrence of an accident involving target materials is estimated to be 3 percent of the 1×10^{-6} per year frequency figure used in the spent nuclear fuel accident analysis. This is because the number of transfer casks that would involve target material is less than 3 percent of that used for 22,700 spent nuclear fuel elements. Therefore, the frequency of this scenario is less than 10^{-7} per year, and is considered to be unforeseeable. Nonetheless, this accident was analyzed and its consequences at potential storage locations were summarized in Table F-118. The frequency of this accident is set conservatively at 10^{-7} per year.

The process by which target materials are prepared for shipment [i.e., drying and canning of the target material solutions, (see Appendix B, Section B.1.5)] releases all gaseous fission products (noble gases and halogens). In addition, the cans in which target materials would be packed do not require any further cutting when they are received in a storage facility. A review of the hypothetical accident scenarios analyzed for spent nuclear fuel indicates that only the aircraft crash with fire accident would be applicable to the target materials. The cans are never cut, and there are no gaseous fission products; therefore, fuel element breach and fuel cask drop scenarios would not be applicable. In addition, should there be an aircraft crash into the wet storage pool where the target material is stored; or, if an accidental criticality in the pool were to occur, the radioactivity releases would be bound by that of the spent nuclear fuel analyzed for these accidents. This is because the amount of radioactive inventory per target material can is very small compared to that in the bounding spent nuclear fuel. In addition, any releases from the target cans would be absorbed in the pool.

Therefore, a scenario involving an aircraft crash into a dry storage facility with ensuing fire was analyzed for the target materials. The scenario assumptions are similar to those described in Section F.6.4.4.3. Because of the size of each can, it was assumed that the transfer cask involved in the accident would contain 40 cans of target materials containing maximum radionuclide inventories, (i.e., 40 cans of 200 grams of ^{235}U per can cooled for at least 3 years). The overall respirable release fraction is assumed to be 5×10^{-3} (Neuhauser and Kanipe, 1993). Table F-119 shows the radioactivity release source terms for this accident.

Table F-118 Summary of the Accident Analysis Dose Assessments for the Aircraft Crash Accident with Fire Involving Target Material - All Pathways

			95 Percent Meteorology			*50 Percent Meteorology*			
Site[c]	*Frequency (event/yr)*	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
NTS		Dose/event	180	28	120	5.6	0.45	2000	3.0
		Dose/yr	0.000018	2.8×10^{-6}	0.000012	5.6×10^{-7}	4.5×10^{-8}	0.00020	3.0×10^{-7}
	1×10^{-7}	LCF[b]	9.0×10^{-12}	1.4×10^{-12}	6.0×10^{-9}	2.8×10^{-13}	2.3×10^{-14}	8.0×10^{-11}	1.5×10^{-10}
ORR		Dose/event	2400	4000	3700	230	910	14000	560
		Dose/yr	0.00024	0.00040	0.00037	0.000023	0.000091	0.0014	0.000056
	1×10^{-7}	LCF	1.2×10^{-10}	2.0×10^{-10}	1.9×10^{-7}	1.2×10^{-11}	4.6×10^{-11}	5.6×10^{-10}	2.8×10^{-8}
INEL		Dose/event	130	63	1500	9.3	5.7	2700	84
		Dose/yr	0.000013	6.3×10^{-6}	0.00015	9.3×10^{-7}	5.7×10^{-7}	0.00027	8.4×10^{-6}
	1×10^{-7}	LCF	6.5×10^{-12}	6.3×10^{-12}	7.5×10^{-8}	4.7×10^{-13}	2.9×10^{-13}	1.1×10^{-10}	4.2×10^{-9}
SRS		Dose/event	26	6.3	970	5.8	0.41	2700	66
		Dose/yr	2.6×10^{-6}	6.3×10^{-7}	0.000097	5.8×10^{-7}	4.1×10^{-8}	0.00027	6.6×10^{-6}
	1×10^{-7}	LCF	1.3×10^{-12}	3.2×10^{-13}	4.9×10^{-8}	2.9×10^{-13}	2.1×10^{-14}	1.1×10^{-10}	3.3×10^{-9}

Table F-118A Summary of the Accident Analysis Dose Assessments for the Aircraft Crash Accident with Fire Involving Target Material - External and Inhalation Pathways

			95 Percent Meteorology		*50 Percent Meteorology*	
Site[c]	*Frequency (event/yr)*	*Risk*	*MEI (mrem)*[a]	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
NTS		Dose/event	64	22	2.1	0.57
		Dose/yr	6.4×10^{-6}	2.2×10^{-6}	2.1×10^{-7}	5.7×10^{-8}
	1×10^{-7}	LCF[b]	3.2×10^{-12}	1.1×10^{-9}	1.1×10^{-13}	2.9×10^{-11}
ORR		Dose/event	870	2500	83	560
		Dose/yr	0.000087	0.00025	8.3×10^{-6}	0.000056
	1×10^{-7}	LCF	4.4×10^{-11}	1.3×10^{-7}	4.2×10^{-12}	2.8×10^{-8}
INEL		Dose/event	20	230	1.4	12
		Dose/yr	2.0×10^{-6}	0.000023	1.4×10^{-7}	1.2×10^{-6}
	1×10^{-7}	LCF	1.0×10^{-12}	1.2×10^{-8}	7.0×10^{-14}	6.0×10^{-10}
SRS		Dose/event	4.4	270	1.0	19
		Dose/yr	4.4×10^{-7}	0.000027	1.0×10^{-7}	1.9×10^{-6}
	1×10^{-7}	LCF	2.2×10^{-13}	1.4×10^{-8}	5.0×10^{-12}	9.5×10^{-10}

NTS = Nevada Test Site; ORR = Oak Ridge Reservation; INEL = Idaho National Engineering Laboratory; SRS = Savannah River Site

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/ yr.*

[c] *Aircraft crash accidents are not applicable to the Hanford Site since their frequency of occurrence is much less than 10^{-7} event/yr.*

Table F-119 Target Materials Aircraft Crash with Fire Accident Source Terms

Isotope	*Curies*
Strontium-89	2.4×10^{-4}
Strontium-90	3.1×10^{0}
Yttrium-90	3.1×10^{0}
Yttrium-91	1.4×10^{-3}
Zirconium-95	5.2×10^{-3}
Niobium-95	1.1×10^{-2}
Rubidium-103	2.1×10^{-6}
Rubidium-106	7.9×10^{-1}
Ruthenium-103m	2.1×10^{-6}
Tin-123	1.1×10^{-3}
Antimony-125	8.2×10^{-2}
Tellurium-125m	2.0×10^{-2}
Tellurium-127m	2.2×10^{-3}
Tellurium-129m	6.0×10^{-9}
Cesium-134	6.5×10^{-3}
Cesium-137	3.0×10^{-1}
Cerium-141	7.4×10^{-8}
Cerium-144	7.7×10^{0}
Presidium-144	7.7×10^{0}
Promethium-147	6.3×10^{0}
Promethium-148m	2.5×10^{-9}
Europium-154	1.4×10^{-3}
Europium-155	5.2×10^{-2}
Uranium-234	1.4×10^{-7}
Uranium-235	8.3×10^{-5}
Uranium-238	1.5×10^{-6}
Plutonium-238	3.3×10^{-6}
Plutonium-239	6.2×10^{-4}
Plutonium-240	1.4×10^{-5}
Plutonium-241	1.3×10^{-4}
Americium-241	6.9×10^{-7}
Americium-242m	4.4×10^{-12}
Americium-243	3.1×10^{-12}
Curium-242	6.8×10^{-12}
Curium-244	3.2×10^{-12}

F.7 Costs

The cost of implementing the proposed action is analyzed in this section. For the purpose of the cost analysis, the alternatives described in Section 2.1 of the EIS were adjusted to reflect the Record of Decision on the Programmatic SNF&INEL Final EIS (DOE, 1995g) issued in May 1995. According to this Record of Decision, if foreign research reactor spent nuclear fuel is managed in the United States, the aluminum-based portion would be managed at the Savannah River Site and the TRIGA portion would be managed at the Idaho National Engineering Laboratory. The cost analysis also considers the financing arrangements discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS that would affect the cost to the United States. The cost information is presented as follows:

F.7.1 Summary of Cost Information

F.7.2 Costs of Individual Program Components

F.7.1 Summary of Cost Information

This section presents total costs for the proposed policy and implementation alternatives that would impact the costs. The costs are presented in two parts: 1) minimum discounted costs (base case) for the well-defined program components and integration approaches, and 2) "other cost factors" that are likely but sufficiently uncertain that they cannot be directly included in the minimum discounted costs. The costs are shown as net present values in a consistent accounting framework.

Several important factors are used when estimating costs. These factors are as follows:

- *Site- and Implementation-Specific Facilities* - All costs for management in the United States are for facilities that exist or are planned at either the Savannah River Site or the Idaho National Engineering Laboratory. Costs are allocated to the program in proportion to the share of foreign research reactor spent nuclear fuel managed or transferred at each facility. This allocation of capital and operating costs within larger programs results in lower costs to the program than would be the case for the use of facilities dedicated to foreign research reactor spent nuclear fuel.

- *Schedule of Activities* - For all management alternatives (except total management overseas), all spent nuclear fuel is shipped, managed for 40 years, and disposed (either as spent nuclear fuel or as reprocessing waste) on schedules that are appropriate for the selected facilities.

- *Discount Rate* - The base case costs are discounted to 1996 at the rate specified by the Office of Management and Budget for the year ending February 1996. This rate is 4.9 percent real. The base case costs for management outside the United States are discounted at a 3 percent real rate of interest. This rate is estimated to be the long-term real rate of interest that can be expected on a trust fund outside the United States. If the net present value of the costs of the program are received in 1996, a hypothetical trust fund invests the money at the real discount rate so that future expenditures are made out of principal and accrued interest.

- *Net Present Value* - Net present value is a figure-of-merit for decision-making on the basis of life-cycle cost, not a value used for establishing budgets or cash flows. All costs are shown in constant 1996 dollars discounted to 1996. This means that the costs for the duration of the program, expressed as a net present value, are due and payable on January 1, 1996, not in the year the costs are incurred.

- *Timing of Expenses* - All costs are assumed to be incurred on the last day of each year of the 40-year management period. The principal and accrued interest in the trust funds (at the net present value of the program costs) are exactly sufficient to meet the costs as they are incurred.

- *Timing of Payments* - Deferring payments beyond January 1, 1996 increases the payments required (either from reactor operators or the United States Congress) by a factor based on the discount rate and the deferral. Pro-forma full-cost recovery fees are shown for payments made on December 31 of each of the 13 receipt years (1996 through 2008).

- *Inflation and Escalation* - Costs are expressed in constant 1996 dollars in this analysis, so the effects of inflation are eliminated. No costs are escalated in real terms.
- *Ultimate Disposition* - Estimated costs for geologic disposal of intact spent nuclear fuel or waste from chemical separation are included to provide a complete life-cycle cost analysis.

F.7.1.1 Scenarios Analyzed

For the purpose of the cost analysis, six scenarios were analyzed. The scenarios reflect the alternatives that affect cost directly, are consistent with the Record of Decision of the Programmatic SNF&INEL Final EIS (DOE, 1995g) and include the costs for ultimate disposal. The six cost scenarios are:

1. *Management Alternative 1 (Storage)* - Storage of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site with new dry or wet storage facilities; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory at existing wet or dry storage facilities.
2. *Management Alternative 1 (revised to incorporate chemical separation)* - Chemical separation of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.
3. *Management Alternative 1 (revised to incorporate a new technology)* — Implementation of a new treatment and/or packaging technology for aluminum-based foreign research reactor spent nuclear fuel and target material at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.
4. *Target Material* - Storage of target material at the Savannah River Site. This scenario provides the cost differential that can be used to assess the cost of managing target material in addition to the foreign research reactor spent nuclear fuel in Management Alternative 1 storage and chemical separation scenarios.
5. *Management Alternative 2* - Management of all foreign research reactor spent nuclear fuel overseas. This scenario reflects a combination of reprocessing and dry storage overseas. Countries with the capability to accept the waste from reprocessing are assumed to have their spent nuclear fuel reprocessed. The rest use dry storage.
6. *Management Alternative 3* - Chemical separation of a portion of the aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; reprocessing of the remainder of aluminum-based foreign research reactor spent nuclear fuel overseas; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

By varying the quantities of material managed in different ways in the United States and overseas, different cost scenarios can be generated. The costs of these variations are bounded by the costs of the scenarios described above. For instance, a management alternative that includes acceptance of target material into the United States would be represented by a combination of Scenarios 1 and 4 or 2 and 4.

The implementation alternatives under Management Alternative 1 related to alternative amounts of foreign research reactor spent nuclear fuel eligible under the policy (Section 2.2.2.1), and alternative policy durations (Section 2.2.2.2), were not considered separately in the cost analysis because they are bounded by the cost scenarios analyzed. These implementation alternatives reduce the amount of foreign research reactor spent nuclear fuel eligible under the policy.

The implementation alternative under Management Alternative 1 related to alternative locations for taking title to the foreign research reactor spent nuclear fuel (Section 2.2.2.4) was not considered because it does not affect the cost analysis.

F.7.1.2 Minimum Program Costs

Table F-120 shows the minimum discounted program costs (base case) for the six scenarios defined above. These costs cover all foreign research reactor spent nuclear fuel shipments, management over 40 years, and geologic disposal. Uncertainties (risks) and escalation are zero. The schedule for activities in Europe under Management Alternative 3 is similar to that in the United States but not exactly the same. Reprocessing takes place over 13 years at Dounreay (the same timespan used for chemical separation at the Savannah River Site) although it could be completed at Dounreay in 9 or 10 years. Dounreay's charges for reprocessing are based on 1996 costs, not costs for 1996 through 2008 averaged over the 13-year period (as was done for the Savannah River Site). Geologic disposal takes place in 2025 through 2030 in Europe and 2030 through 2035 in the United States. Costs are discounted at 3 percent for the portion to be managed overseas and at 4.9 percent for the portion to be managed in the United States.

Table F-120 Minimum Program Costs
(Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	Net Present Value
1. Management Alternative 1 (Storage)	725/775[a]
2. Management Alternative 1 (revised to incorporate Chemical Separation)	625
3. Management Alternative 1 (revised to incorporate a New Technology)[b]	625-950
4. Target Material	35
5. Management Alternative 2	1,250
6. Management Alternative 3	675

[a] *Dry/Wet new storage facilities*

[b] *Includes target material*

Because of the uncertainties involved with the implementation of the new technology, the cost for Scenario 3 is presented as a range as discussed in Appendix F, Section F.7.2.9. Also, shipping costs in Scenario 3 include the assumption that of the total number of cask shipments, only 38 cask shipments would be accepted at the West Coast.

F.7.1.3 Other Cost Factors

There are four important sources of cost risk (excluding escalation) that are not part of the minimum costs in Table F-120. Table F-121 shows the likely values (risks) for these factors, taking into account the absolute values of the uncertainties and their probability of occurrence. A brief summary of these cost factors follows the table.

The other cost factors summarized in Table F-121 are as follows:

1. *Systems Integration and Logistics Risks* - Significant risks exist in the details of the policy implementation. The implementation of the policy would involve up to 41 foreign countries, up to 13 years of receipts, dozens of foreign ports, up to ten domestic ports, two U.S.

Table F-121 Other Cost Factors
(Net Present Value, Millions of 1996 Dollars in 1996)

	Cost Factors				
	Systems Integration &	Component	Non-Program	3% Discount	

F.7.1.4 Potential Total Costs

Table F-122 combines the base case costs with the "other cost factors" to provide a realistic expectation of the potential total costs of the program, excluding escalation. The "other cost factors" are divided into technical factors and discount rate-related factors. This table also shows the cumulative percentage effect on the minimum discounted program costs of real escalation at a rate of 1 percent per year over 40 years.

Table F-122 Potential Total Costs
(Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	Minimum Program Cost	Other Cost Factors (Technical)	Other Cost Factors (Discount Rate)	Potential Total Cost, No Escalation	1% Real Escalation, Cumulative
1. Management Alternative 1 (Storage)	725/775[a]	210	175	≈1,100	+11%
2. Management Alternative 1 (revised to incorporate chemical separation)	625	85-145	125	≈900	+9%
3. Management Alternative 1 (revised to incorporate a new technology)[c]	625-950	210	225	≈1,050-1,400	10%-11%
5. Management Alternative 2	1250	600-1600	250	2,100-3,100	+13%
6. Management Alternative 3[b]	675	225-275	75	≈1,000	+9%

[a] *Dry/Wet new storage facilities.*

[b] *The total cost risk to the United States is less than 1/2 the total cost risk since a large portion of the activities under this alternative would occur overseas.*

1. United States bears the full cost of the program for developing countries and charges a competitive fee to developed countries.

2. United States bears the full cost for all countries (*no* fee).

3. United States charges a *full-cost-recovery* fee to all countries.

4. United States bears the full cost of the program for developing countries and charges a *full-cost-recovery* fee to developed countries.

From a practical standpoint, the U.S. cost under financing arrangement 3 above would be zero. The issue would be whether any foreign countries would participate in the program if full-cost recovery exceeded a competitive fee. The first and fourth arrangements are functionally similar, the U.S. cost resulting from the difference in the *competitive versus the full-cost-recovery fee.* The U.S. cost under the second arrangement (no fee) would be the total program cost as discussed earlier. Any fees established by the United States will take place pursuant to a Federal Register notice after the Record of Decision for this EIS.

Table F-123 shows costs to the United States for the minimum program in each of the cost scenarios analyzed (except target material) under a variety of fee schedules. Adding target material to Scenarios 1, 2, 5, or 6 would increase the cost by 3 to 4 percent. Fees of $2,000/kgTM, $5,000/kgTM, $7,500/kgTM, and $10,000/kgTM, including a pass-through of shipping charges (all expressed in constant 1996 dollars and levelized over 13 years), are used to provide a range of estimates for the cost to the United States. These fees do not imply that reactor operators would pay them for management in Europe or the United States, or that the fee established by the United States will be one of these values. They are used for illustration only and suggest a bounding range, exclusive of technical risk factors, discount rate adjustments, and escalation. The cost to the United States, presented in Table F-123, is the sum of: 1) the cost of managing the foreign research reactor spent nuclear fuel from the developing countries, including shipping, and 2) the difference between the revenues received for management of developed country foreign research reactor spent nuclear fuel and the total program cost of managing developed country foreign research reactor spent nuclear fuel, excluding shipping. Including shipping in the U.S. management costs allows management costs for the United States and the United Kingdom to be presented on a comparable basis.

Table F-123 shows that for minimum discounted program costs and fees charged to developed country reactor operators levelized over 13 years, costs to the United States for management of foreign research reactor spent nuclear fuel (and target material in Scenario 3) could range from several hundred million dollars at a fee of $2,000/kgTM to a profit for fees of $7,500/kgTM to $10,000/kgTM. The cost of managing the spent nuclear fuel from the developing countries (including shipping) adds roughly $100 million more to the cost borne by the United States. Excluding Scenario 5, for which all costs and fees are speculative, the table shows that costs to the United States in Management Alternative 3 are significantly lower than for Management Alternative 1. The savings to the United States exist because the United States bears none of the cost of Spent Nuclear Fuel Management in Europe except the cost of blending down the HEU at Dounreay.

If fees in the $2,000 to $10,000 per kgTM range (levelized $1996 dollars) are established and charged over 13 years, the costs to the United States would be as estimated in Table F-123 (excluding target materials) plus any additional cost factors not incorporated in the minimum program costs. These additional cost factors are: 1) technical risks, 2) discount rate-related risks, and 3) escalation. Table F-122 shows that

Table F-123 Costs to the United States for the Minimum Program Under Various Scenarios and Fee Structures (Millions of 1996 Dollars, Net Present Value of Costs in 1996, Fees Levelized Over 1996-2008 Period)

Scenario[a]	Full-Cost Recovery[b]	Levelized Shipping Fee $/kgTM	Levelized Management Fee (excluding shipping) $/kgTM	Net Present Value For Levelized Fee[c] (Developed Countries Only) $2,000/ kgTM	$5,000/ kgTM	$7,500/ kgTM	$10,000/ kgTM	No Fee[d] Developed Countries	Total (excluding shipping)
1. Management Alternative 1 (Storage)	100	1,500	6,500	325	100	(75)	(250)	475	575
2. Management Alternative 1 (revised to incorporate Chemical Separation)	90	1,500	5,800	275	50	(125)	(300)	425	525
3. Management Alternative 1 (revised to incorporate a New Technology)	90-110	1,700	5,600-9,200	275-550	50-325	(150)-125	(325)-(50)	425-700	500-800
5. Management Alternative 2[e]	500+							1,250 +	1,750+
6. Management Alternative 3[f]	85	1,500	6,000	225	75	(50)	(175)	300	375

[a] *The total mass (kgTM) of foreign research reactor spent nuclear fuel in the various scenarios is approximately as follows: Aluminum-based plus TRIGA: 115,000 kgTM; from developing countries: 15,000 kgTM; from developed countries: 100,000 kgTM; to Dounreay in Management Alternative 3: 37,000 kgTM. The total mass of target material is approximately 3,400 kgTM aluminum-based equivalent and essentially all .from developed countries.*

[b] *Full-cost recovery from developed countries only. The United States bears the costs of the developing countries in these cases.*

[c] *Net present value of costs to the United States for management fees paid in 13 equal annual installments on December 31 of the years 1996 through 2008. Add costs in column labeled "Full-Cost Recovery" to generate total cost to the United States (developed and developing countries).*

[d] *As above, implicitly paid by the taxpayers in 13 equal annual installments (to maintain consistency with the payment period of the reactor operators), excluding shipping. The net present value of shipping in Scenarios 1 [Management Alternative 1 (Storage)] and 2 [Management Alternative 1 (revised to incorporate chemical separation)] is $140 Million. The net present value of shipping to the United States only in Scenario 6 is $90 Million. The net present value of shipping in Scenario 3 [Management Alternative 1 (revised to incorporate a new technology)] is $160 Million.*

[e] *There is no defined basis for the charges to the United States for non-U.S. management. Costs to the United States under Management Alternative 2 assume that the United States absorbs the cost to construct and operate independent foreign research reactor spent nuclear fuel storage installations (including all supporting safety, security, transport, health physics, etc. infrastructure) for the 22 countries with no commercial nuclear power programs and that the United States partially subsidizes the other countries, depending on developmental status, commercial nuclear power infrastructure, and other factors.*

[f] *U.S. component of Management Alternative 3 only. Revenues paid to the United States exclude shipping charges. Costs to the United States for management in Europe consist only of the charge to blend down the HEU to LEU ($20 million). European reactor operators using Dounreay are assumed to bear all other costs.*

technical risks could add roughly $100 to $200 million to the costs borne by the United States. Discount rate-related risks are of a similar size. Escalation risks are more uncertain but could be in the same range.

F.7.2 Costs of Individual Program Components

This section provides details on program costs for each of the scenarios outlined in section F.7.1.

F.7.2.1 Programmatic Cost Assumptions

Table F-124 shows programmatic assumptions about costs and the basis for the cost calculations.

Table F-124 Programmatic Assumptions and Bases

Variable	*Assumption*	*Basis*
Year Dollars	1996	Standardized to first year of program.
Discount Rate for Management in the United States	4.9 percent real	Required by Office of Management and Budget for programs beginning between February 1995 and February 1996.
Discount Rate for Management in Europe	3.0 percent real	Representative of long-run average in larger Western European economies.
Rounding of Totals	$25 million	Highlights differences between programs that typically differ by $100 million. No implication of precision.
Component Contingencies	Included in base costs	Standard costing assumption
Program Risks	Not included in base costs	Logistical complexity of program could add 10-15 percent to total costs.
Uncertainties	Not included in base costs	
Risk-adjustment	Not included in base costs	
Escalation	Not included in base costs	
Costs incurred over what period	40 years (1996 to 2035) in United States 35 years (1996 to 2020) in United Kingdom	Maximum length of interim storage
Repository Shipping	2030 to 2035 in United States 2025 to 2030 in United Kingdom	Storage maximum in United States and United Kingdom
Qualification of fuel types for disposal	$10M per type allocated to the program, 5 types in program	Idaho National Engineering Laboratory estimate. The foreign research reactor spent nuclear fuel program is estimated to be responsible for three types of aluminum-based spent nuclear fuel and two types of TRIGA spent nuclear fuel. The types are related to the repository program characteristics.

F.7.2.2 Individual Program Components

The proposed foreign research reactor spent nuclear fuel program consists of many components. Table F-125 outlines the components of the five cost analysis scenarios described in Section F.7.1. Detailed discussions of the individual program components follow the table.

F.7.2.3 Logistics and Program Management

Under Management Alternative 1, the United States would undertake a program where the maximum requirements begin with the shipping of an estimated 837 casks of foreign research reactor spent nuclear

Table F-125 Applicability of Specific Cost Components to the Cost Analysis Scenarios

			Management Alternative 1 (revised to	Management Alternative 1 (revised to			

F.7.2.4 Shipping to the United States

Shipping the foreign research reactor spent nuclear fuel and target material to the United States requires an estimated 977 cask shipments. Of this, 837 cask shipments would contain spent nuclear fuel and 140 cask shipments would contain target material.[4] The shipping period would be

Australian cask is assumed to be shipped as part of larger shipments from Asia. These shipments would carry 6 casks per vessel and call on three ports per transit to the United States.

- **Japan** — The Japan Atomic Energy Research Institute owns two casks that would be used for spent nuclear fuel accepted by the United States. Because the Japan Atomic Energy Research Institute is near the port of export for Japan, inland freight charges would be negligible. Japan would likely require chartered vessels (at least as far as Europe for shipments of spent nuclear fuel to the United States via Europe). Shipment by chartered vessel would be approximately $450,000, or $225,000 per cask. It is estimated that Japan would ship approximately 110 casks to the United States (99 aluminum-based and 11 TRIGA). Japan could choose to acquire more casks to reduce its cost per ocean transit. As with Australia, a cask charge is assigned to show true program costs.

- **Asia** (excluding Australia and Japan) — Asian nations (excluding Japan) would be expected to have relatively low inland freight costs. It is unclear if Asian nations would require chartered vessels. Asian nations (excluding Australia and Japan) account for an estimated 62 casks (23 containing aluminum-based spent nuclear fuel, 1 containing target material, and 38 containing TRIGA spent nuclear fuel).

- **Canada** — For cost analysis, all Canadian shipments (approximately 116 casks of aluminum-based spent nuclear fuel and 125 casks of target material) are assumed to come by truck to the Savannah River Site. Cask rental and inland freight charges reflect the shipping times and distances for long overland routes. Shipping by rail is also feasible.

- **Other Atlantic** — All other nations nearer the Atlantic Ocean than the Pacific Ocean are assumed to have characteristics similar to those of Asia (excluding Australia and Japan) but lower ocean shipping costs because of greater proximity to the United States. Shipments from Mexico would come by sea, since the Mexican spent nuclear fuel is located in the southern part of the country. The Other Atlantic nations are not likely to require chartered vessels. Other Atlantic nations account for 38 casks, 23 of which would contain aluminum-based spent nuclear fuel and 15 of which would contain TRIGA spent nuclear fuel.

- **Other Pacific** — All other nations nearer the Pacific Ocean than the Atlantic Ocean are assumed to have characteristics similar to those of Asia (excluding Australia and Japan) but lower ocean shipping costs because of greater proximity to the United States. Because the Other Pacific countries are on the western coast of South America (which is significantly closer to the southeastern United States than the northwestern United States) and because all the spent nuclear fuel from these countries is aluminum-based, the EIS assumes that all shipments from Other Pacific countries will go by sea to an East Coast port via the Panama Canal. The Other Pacific nations would not be likely to require chartered vessels. Other Pacific nations account for 12 casks, all of which would contain aluminum-based spent nuclear fuel.

Table F-126 summarizes the cost of shipping a single spent nuclear fuel cask from various parts of the world to the United States in the configuration considered most likely by this EIS. The base case assumes the use of charter ships. The discounted cost of overseas shipping to the United States (including overland shipping from Canada and including target material) is shown in the table as $158 million (summing the bottom row). Of the 977 shipments, 827 originate either in Canada or in ports nearer the U.S. East Coast

Table F-126 Representative Shipping Costs to/from the United States for a Spent Nuclear Fuel Cask (Thousands of 1996 Dollars per Cask and Millions of 1996 Dollars for the Program, including Target Material)

Activity/Cost	*Europe*	*Australia*	*Japan*	*Other Asia*	*Canada*	*Other Atlantic*	*Other Pacific*
Charter	Y	Y	Y	Y	N/A	Y	Y
U.S. Coast	East	West	West	West	N/A	East	East
Charter Cost $k	400	550	550	500	N/A	300	300
Casks/Charter	6	See Other Asia	6	6	N/A	6	6
Ports-of-Call	2	See Other Asia	1	3	N/A	3	3
Total Rental Charges, $k/Cask	51	48	42	66	21	60	66
Inland Freight, Country, Site, and Overland Route Weighted, $k/Cask	37	38	41	30	25	26	38
Insurance, Security, Administration, Cask Return, $k/Cask	51	49	58	70	36	49	49
$k/Cask, Excluding Contingency	224	253	239	246	86	232	246
Number of Casks (Aluminum)	393	9	99	23	116	23	12
Number of Casks (TRIGA)	98	0	11	38	0	15	0
Number of Casks (Target Material)	14	0	0	1	125	0	0
Number of Casks (Total)	505	9	110	62	241	38	12
of which, from Developing Countries	72	0	0	53	0	38	12
Total Cost, including 15% Contingency, $M	130	2	30	18	24	10	3
Discounted Cost ($M)	95	2	22	13	17	7	2

(including 12 cask shipments from the West Coast of South America). The remaining 150 cask shipments originate in ports nearer the U.S. West Coast. Assuming shipments to the nearest U.S. coast, regardless of the type of spent nuclear fuel, an estimated 113 shipments of TRIGA spent nuclear fuel received at East Coast ports and an estimated 132 shipments of

Shipments from the rest of the world (excluding Europe and Canada) are assumed by charter at the rate of 6 casks per vessel and 3 ports-of-call (i.e., two casks per country). Adding ports-of-call increases costs in transit (by about $20,000 per port-of-call and $20,000 per day in transit between ports) but saves money on balance by increasing the number of casks on the ship. Reducing the shipments from Asia (excluding Australia, Japan) and the Other Atlantic and Other Pacific countries to 2 casks and 1 port-of-call would increase program costs by $12 million.

- **Shipping to Distant Coasts and Sites** -- The cost of shipping the foreign research reactor spent nuclear fuel depends on which ports were selected and from where they would be accepting the shipments. The dynamics of the program are that roughly 75 percent of the foreign research reactor spent nuclear fuel is aluminum-based (and therefore would be destined for the Savannah River Site, on the United States' East Coast) and roughly 75 percent of the foreign research reactor spent nuclear fuel (excluding Canadian spent nuclear fuel) is in countries on the Atlantic side of the United States. While the 75 percent aluminum-based spent nuclear fuel and the 75 percent Atlantic spent nuclear fuel are not identical, there is sufficient overlap to create a situation where shipping all the spent nuclear fuel directly to a United States East Coast port and then distributing the TRIGA spent nuclear fuel to the Idaho National Engineering Laboratory by land would be only about 5 percent ($8 million) more expensive than shipping the spent nuclear fuel to the nearest port and then overland to the appropriate site. The cost of overland shipping by truck from an eastern port to the Idaho National Engineering Laboratory for a shipment that would logically arrive at an eastern port is less than the cost of ocean shipping to a western port to minimize the overland transit by truck.

- **Receipt Rates at the Savannah River Site and the Idaho National Engineering Laboratory** -- To accept all the foreign research reactor spent nuclear fuel within the proposed 13-year period requires, on average, cask receipts of almost six casks per month (seven per month if target material is included). Splitting the spent nuclear fuel by fuel type, consistent with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995g), implies receipt of 4 to 5 casks per month of aluminum-based spent nuclear fuel at the Savannah River Site and about one cask per month of TRIGA spent nuclear fuel at the Idaho National Engineering Laboratory.[6] About 1 cask per month of target material would also be received at the Savannah River Site.

- **Cask Rental Charges** -- Truck casks rent for approximately $1,500 per day on long-term lease. Shorter-term rentals are appreciably more expensive (EG&G, 1994b). Table F-126 incorporates the $1,500 per day rate for a long-term lease. The use of the smaller truck casks (compared to rail casks) permits savings in ocean shipping, short overland transport (although this could change in response to high charter costs), and security. The cost to acquire a new truck cask has been increasing steadily and is now approaching $2 million. The time from ordering to delivery exceeds 1 year. Because of the limited market for casks and the risk of constructing a cask for which there is no long-term demand, potential cask owners and lessors would place a high fixed charge rate on an investment in new casks for the foreign research reactor program. For a 20-year operating life, the fixed

6 *The weighted-average number of spent nuclear fuel elements per cask is estimated to be slightly more than 27. The sites are limited by cask receipt rates, not elements per cask. Some casks would have as many as 120 elements. Others would have one element. Most would have about 27 to 30 elements.*

charge rate would be at least 30 percent. For a fixed charge rate of 30 percent, a $2 million cask must rent for $600,000 per year, or approximately $1,650 per day on a yearly lease.

- **Cask Shipment and Rental Periods** -- The average time required to complete a round-trip shipment depends on the area of cask origin, the number of casks shipped at one time, the number of ports-of-call made enroute to the United States, inland shipping in the United States, and turnaround time at the sites. Excluding Canada, round-trip cask shipment periods range from an average of less than 40 days for a cask from the Atlantic coast of South America to the southeast coast of the United States (with an ultimate destination of the Savannah River Site) to more than 60 days for a cask from Australia to the same ultimate destination (either via a Pacific port and an overland transit to the Savannah River Site or via a passage through the Panama Canal to an Atlantic Port).

The base costs cover two ocean transits, port handling in two countries, shipment to and from the cask lessor, and overland transport from the ports to and from the sites and reactor facilities. Cask handling at the sites is estimated separately.

- **Contingencies** -- Over the past few years, the cost of almost all phases of international spent nuclear fuel shipping has risen sharply. Also, European regulations regarding ocean shipping of nuclear cargoes have tightened dramatically. While these costs are built into the values in Table F-126, potentially large additional contingencies are not. These contingencies include escalating cask lease rates; partially filled casks; higher inland freight charges in the United States; dedicated rail shipping in the United States; consolidation limitations in Asia, South America, or Africa; and additional security. On the other hand, the single largest contingency -- the use of charter ships -- has been added to the base case. Consideration of the magnitude of the contingencies suggests a contingency factor of about 15 percent. This factor applies to the shipping component of the program only, not the impacts on the program logistics or integration from delays in shipping, barriers erected by the States, etc. These program-level impacts are discussed separately in Section F.7.4.

F.7.2.5 Shipping to the United Kingdom

Shipping to the United Kingdom is less expensive than shipping to the United States. Cost estimates provided by the United Kingdom Atomic Energy Authority for this EIS are about $30,000 per cask from Europe (Scullion, 1995). This compares to more than $200,000 per cask estimated for shipments from Europe to the United States. The estimates for shipping to the United Kingdom reflect the large savings from the very short ocean transit from continental Europe (and thus the vessel charter cost), the ocean transit and site turn-around periods (and thus the cask rental time), the inland freight charges for shipping a short distance in the United Kingdom, and the reduced administrative, insurance, and security costs for the shorter activity.

It is possible that the estimated cost for shipments to the United Kingdom is understated in comparison to the U.S. costs for at least two reasons. First, no detailed analysis of the cost components similar to that in Table F-126 was conducted and thus some costs, especially indirect costs, such as administration, may have been omitted. Second, costs for shipments to the United States have increased sharply in recent years. Costs for recent shipments to the United States were higher than anticipated and may not be reflected in the estimated costs to ship from Europe to the United Kingdom.

F.7.2.6 Storage at the Savannah River Site

Consistent with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995g), approximately 17,800 aluminum-based spent nuclear fuel elements could be received and managed at the Savannah River Site. These elements would be stored or chemically separated. Under Implementation Alternative 1c to Management Alternative 1, target material equal to about 600 aluminum-based elements could also be received and stored at the Savannah River Site. The cost to receive and store the target material is proportional to the ratio of target material (expressed in element-equivalents, e.g., cans) to spent nuclear fuel elements (i.e., about 3.4 percent). Costs in this section refer to the basic implementation of Management Alternative 1 (17,800 spent nuclear fuel elements and no targets).

Storage at the Savannah River Site would consist of two phases: Phase-1 storage in existing facilities and Phase-2 storage in new facilities. Logistically, the base case for Management Alternative 1 (storage) is as follows:

- At the start of the implementation period, aluminum-based spent nuclear fuel would be shipped to the Savannah River Site and wet-stored in RBOF and the L-Reactor disassembly basin.
- At about the same time, construction would begin on a staging and characterization facility and an interim dry or wet storage facility at the Savannah River Site. The staging facility would be designed to receive and transfer all the foreign research reactor spent nuclear fuel (and other nuclear materials, including domestic research reactor spent nuclear fuels). The dry or wet storage facility would be designed to store the spent nuclear fuel (and possibly target material) until the spent nuclear fuel and target material were prepared for shipment to the repository. The new facilities would be commissioned in 2003, accept off-site receipts of foreign research reactor spent nuclear fuel through 2008, and on-site transfers (of all aluminum-based materials, not just foreign research reactor spent nuclear fuel) from the RBOF and the L-Reactor disassembly basin through about 2008 or 2009. If commissioning of the new storage facility is delayed to 2005, transfers from existing basins would continue through about 2010.
- At some point in the 2015 to 2035 time period, the stored spent nuclear fuel would be prepared for repository disposal in as-yet unspecified repository-qualified canisters. Cost estimates are based on a repository packaging and shipping period of 2030 to 2035.

Table F-127 shows the annual costs for storage of 17,800 foreign research reactor spent nuclear fuel elements at the Savannah River Site during Phase 1 and Phase 2, where Phase 2 storage is dry (WSRC, 1995c). Receiving and storing target material would add $20 million (discounted) to expenditures at the Savannah River Site and $35 million (discounted) to the total costs. The key assumptions used to generate the costs in Table F-127 are discussed below.

- Annual operating costs for round-the-clock operations at RBOF and L-Reactor disassembly basin are allocated to the foreign research reactor spent nuclear fuel program in proportion to the share of foreign research reactor spent nuclear fuel mass transferred to or from the basins relative to total cask transfers at RBOF, and L-Reactor disassembly basin in each year until all of the foreign research reactor spent nuclear fuel has been transferred to dry storage (about 2008 or 2009). Unit costs are assumed fixed in each year. Thus, allocable costs scale in proportion to the amount of foreign research reactor material received at the basins.

Table F-127 Storage of Aluminum-Based Spent Nuclear Fuel at the Savannah River Site, Including Phase 2 Dry Storage (Millions of 1996 Dollars)

				Capital	Capital	Operating	Decontamination

River Site over the 1996 through 2035 period (by MTR-equivalents) but only 28 percent of the total aluminum-based spent nuclear fuel received initially at the new staging and characterization facility. The unweighted average of these two percentages is 43 percent. Because most of the foreign research reactor spent nuclear fuel arrives prior to the operation of the new staging and characterization facility, the foreign research reactor spent nuclear fuel bears a disproportionately high share of the operating costs of RBOF and L-Reactor disassembly basin and a disproportionately low share of the capital and operating costs of the new staging and characterization facility.

- A new dry or wet storage facility would be constructed for operation in 2003. A dry storage facility would consist of a pad, fence, canisters, and storage overpacks. It would operate through 2035. The canisters used at the dry facility would not necessarily be

F.7.2.7 Storage at the Idaho National Engineering Laboratory

In the base scenarios involving United States acceptance of foreign research reactor spent nuclear fuel, approximately 4,900 TRIGA elements would be shipped to the Idaho National Engineering Laboratory for storage in existing facilities.

The Idaho National Engineering Laboratory would store the TRIGA spent nuclear fuel in the IFSF until the spent nuclear fuel was transferred to canisters for shipping to the repository. Table F-128 shows annual operating costs for the Idaho National Engineering Laboratory to dry-store approximately 4,900 TRIGA elements at the IFSF. (The Idaho National Engineering Laboratory could also wet-store the TRIGA elements at the FAST facility for about twice the cost as at IFSF.) The discounted total cost using the IFSF facility for storage is approximately $30 million. Qualification of TRIGA spent nuclear fuel for repository disposal would add another $15 million.

To complete the transfers from existing storage facilities to repository-qualified dry storage canisters, the Idaho National Engineering Laboratory might eventually require a new staging facility similar to that at the Savannah River Site. The Idaho National Engineering Laboratory is deferring construction of this facility until the repository waste acceptance criteria are available some time after 2000. Based on the share of TRIGA spent nuclear fuel relative to all material to be dry-stored at the Idaho National Engineering Laboratory and geologically disposed, the foreign research reactor spent nuclear fuel program would be allocated no more than $10 million of the capital cost of a staging facility whose discounted total cost would be less than $150 million. Allocable operating costs would be the same as shown in the column for repository canister loading. For cost analysis, repository loading and shipping is assumed to take place in 2030. Actual loading could take place earlier but cannot be specified at present.

F.7.2.8 Chemical Separation at the Savannah River Site

Implementation of Management Alternative 1 (revised to incorporate chemical separation) at the Savannah River Site could take place in different ways. One bounding case is to assume that existing and new facilities are used in essentially the same way as in Management Alternative 1 (storage). RBOF and L-Reactor disassembly basin are used for receiving and lag-storage, a new staging and characterization facility is required for repository loading of aluminum-based material received after the completion of chemical separation operations, one of the Canyons (F- or H-Canyon) is used at a moderate rate, new dry or wet storage facilities are required, etc. This option can be viewed as the separation of foreign research reactor spent nuclear fuel within a larger program to store and directly dispose non-foreign aluminum-based spent nuclear fuel.

The other bounding case can be viewed as the separation of foreign research reactor spent nuclear fuel within an accelerated program to chemically separate all of the accumulated aluminum-based materials and medium-term receipts. In this case, RBOF continues to be used for 40 years for receipts, characterization, storage and repository loading; no new staging and characterization facility is constructed; receipts of aluminum-based spent nuclear fuel from domestic sources are accelerated; one of the Canyons (F- or H-Canyon) is used at an accelerated pace; and no new dry or wet storage facilities are required.

In either case, chemical separation continues to around 2008 to 2010, at which point the canyons are shut down. In the first case, however, enough material remains on site and due to be received that a large-scale storage program (including a new staging and characterization facility) is required. In the second case, very little separable material remains on site and only about 5 casks per year are due to be received at the

Table F-128 Storage of TRIGA Spent Nuclear Fuel at the Idaho National Engineering Laboratory (Millions of 1996 Dollars)

Year	Capital Costs -Staging	Transfers	IFSF Operations	Repository Canister Loading	Repository Canisters	Operations & Maintenance	Decontamination &Decommissioning
1996		.4	1				
1997	1	.4	1				
1998		.4	1				
1999		.4	1				
2000		.4	1				
2001		.4	1				
2002		.4	1				
2003		.4	1				
2004		.4	1				
2005		.4	1				
2006		.4	1				
2007		.4	1				
2008		.4	1				
2009			1				
2010			1				
2011			1				
2012			1				
2013	3		1				
2014	4		1				
2015	3		1				
2016-2029			1/yr.				
2030			1	5	10	.1/yr	
2031-2035							2
Total Costs (Undiscounted)	11	5	35	5	10	4	2
NPV	5	4	17	1	2	2	0

time the Canyons are shut down. In this latter case, existing facilities can handle all program functions, including repository canister loading.

The first case is used for cost analysis purposes in this Appendix. This case is more probable, since it is more conservative with respect to selection of separation as an alternative and more conservative with respect to costs.

In either case, uranium (but not plutonium) is chemically separated from fission products at one of the canyons at the Savannah River Site.[8] In this EIS, costs for operations at F-Canyon are used since they are slightly higher than costs at H-Canyon (about $25 million). The credit for recovered uranium is the same in either case.

For either Management Alternative 1 (revised to incorporate chemical separation) (17,800 elements) or Management Alternative 3 (12,200 elements), the following assumptions apply:

8 *HEU cannot be chemically separated from LEU. Plutonium can be chemically separated from uranium and fission products but it is not the intention of the Savannah River Site or this EIS to do so. The amount of plutonium in the foreign research reactor spent nuclear fuel is negligible.*

- Basin operations continue until all material can be transferred out of the basins to the Canyons. Foreign research reactor spent nuclear fuel is out of the basins in 2006 under Management Alternative 1 (revised to incorporate chemical separation) and 2005 under Management Alternative 3. From 2003 forward, all receipts take place at the new staging and characterization facility.

- Canyon operations would take place over a maximum of 13 years (1998 through 2010). [illegible]

[illegible]

Table F-129 Chemical Separation Costs at the Savannah River Site Under Management Alternative 1 (Revised to Incorporate Chemical Separation)

Year	*Allocated Receiving, Lag Storage, Facilities Support*	*Spare Dissolver Capacity (MTR-equivalents)*	*Approximate Foreign Research Reactor Share of Dissolver (percent)*	*Allocated Canyon Operations Cost*	*Allocated Canyon Deferral Penalty*	*Allocated HEU Credit*
1996	16	0				
1997	22	0				
1998	25	720	72	13		3
1999	26	720	72	13		3
2000	27	2880	72	4		10
2001	22	2880	72	2		10
2002	23	2880	72	10		10
2003	17	2880	72	23	8	10
2004	16	2880	72	23	16	10
2005	19	2880	72	23	24	10
2006	8	2880	72	23	24	10
2007	0	2880	72	23	24	10
2008	1	2880	72	23	24	10
2009	1	2880	60	19	20	9
2010	1	2880	0	21	20	3
2011-2035	0-2/yr.					
Total Costs (Undiscounted)	242			199	139	107
NPV	178			127	81	70

separation). The foreign research reactor spent nuclear fuel program incurs costs at the percentage shown in the fourth column. This percentage is approximately the spare dissolver capacity allocated to the foreign research reactor program in each year. The change in the percentage at the end occurs because less than proportional dissolver capacity is required to complete the foreign research reactor spent nuclear fuel processing.

Table F-130 shows the same information as Table F-129, adjusted for the shipment of 5,600 aluminum-based elements to Dounreay, Scotland (Management Alternative 3). Table F-130 shows

Table F-130 Chemical Separation Costs at the Savannah River Site Under Management Alternative 3

Year	Allocated Receiving, Lag Storage, Facilities Support	Spare Dissolver Capacity (MTR-equivalents)	Approximate Foreign Research Reactor Share of Dissolver Capacity (Percent)	Allocated Canyon Operations Cost	Allocated Canyon Deferral Penalty	Allocated HEU Credit
1996	12	0				
1997	17	0				
1998	20	720	62	11		
1999	22	720	62	11		2
2000	22	2880	62	3		2
2001	16	2880	62	2		9
2002	16	2880	62	8		9
2003	13	2880	62	20	7	9
2004	11	2880	62	20	14	9
2005	7	2880	62	20	20	9
2006	0	2880	62	20	20	9
2007	0	2880	6	2	2	1
2008	0	2880	0	0	0	0
2009	0	2880	0	0	0	0
2010	0	2880	0	0	0	0
2011-2035	.1-.5/yr.					
Total Costs (Undiscounted)	173			117	63	68
NPV	129			80	39	47

which would then be constructed and operated to manage the foreign research reactor fuel. A number of different technologies will be considered before one or more are selected for further development.

In addition to the uncertainty as to which technology(ies) will be chosen, there are other cost uncertainties including: the repository disposal fee, the need for new facilities and the requirements needed for managing domestic fuel. To account for these uncertainties, a range of costs have been developed. The costs range from about $950 million (undiscounted) or $625 million (discounted) to about $1.75 billion (undiscounted) or $950 million (discounted).

F.7.2.10 Reprocessing in the United Kingdom

Under Management Alternative 3, approximately 5,600 aluminum-based spent nuclear fuel elements would be shipped to the United Kingdom Atomic Energy Authority's facility at Dounreay, Scotland for reprocessing.[11] The remaining 12,200 aluminum-based elements would be chemically separated at the Savannah River Site (Section F.7.2.8).[12] The TRIGA spent nuclear fuel would be stored at the Idaho National Engineering Laboratory (Section F.7.2.7).

11 Equal to about 7,900 MTR-equivalents, including 85 RHF elements at 20 MTR-equivalents apiece.

12 Equal to about 13,600 MTR-equivalents, including the 2,650 Canadian NRU elements and all other elements (excluding the French RHF elements) at 1.12 MTR- equivalents apiece.

The number of elements to be reprocessed at Dounreay is based on the number of spent nuclear fuel elements in countries with commercial nuclear power programs and the clear capability to manage the reprocessing wastes.[13] The reprocessing waste from Dounreay is returned to the countries of origin. More generally, Management Alternative 3 can be viewed as chemical separation of approximately 2/3 of the aluminum-based foreign research reactor spent nuclear fuel elements in the United States and 1/3 in the United Kingdom.

Table F-131 shows the United Kingdom Atomic Energy Authority's currently estimated costs to reprocess aluminum-based spent nuclear fuel elements. The cost for conversion assumes a downblending ratio of 2:1 (i.e., one unit of depleted uranium at 0 percent enrichment is added to each unit of separated uranium at 40 percent enrichment to produce two units of uranium at 20 percent enrichment). The costs in Table F-131 are converted from British Pounds to United States Dollars at a rate of 1.55 dollars per pound. Using these costs, the discounted cost to ship, receive, reprocess, and dispose of the wastes from 5,600 aluminum-based spent nuclear fuel elements on a schedule similar to that at the Savannah River Site and to obtain LEU metal fuel is approximately $265 million. At a discount rate of 3 percent, this is equivalent to about $7,000/kgTM, including a charge of about $700/kgTM for blending down the separated HEU to LEU.

Table F-131 Costs at Dounreay (1996 Dollars)

Activity	*Cost*
Transport Casks to/from Dounreay	$31,000/cask @ 2 casks per shipment
Receive & Unload Casks	$7,700/cask
Reprocess and produce cementous intermediate-level waste	$5,750/kgTM (HEU only)
Convert Uranyl Nitrate to Metal	$4,500/kg uranium metal (or $2,800/kg UO_2 for oxide)
Value of Metallic Uranium	$15,000/kg uranium
Store U-235	$1,550/kg U-235
Store intermediate-level waste	$1,550 per 500 l (132 gal) drum per year (containing 10 kg (22 lb) of spent nuclear fuel wastes)
Transport intermediate-level waste to originating country	$2,600/drum
Geologic disposal of intermediate-level waste	$31,000/drum

Source: All Costs (except value of metallic uranium) (Scullion, 1995).

Foreign research reactor operators may prefer to view their costs as the sum of the undiscounted current costs for shipping, reprocessing, and uranyl nitrate conversion to metal (without downblending to LEU) plus the discounted costs for interim storage of uranium, interim storage of reprocessing waste, and geologic disposal of reprocessing waste. Assuming a 3 percent discount rate for the outyear costs, and excluding the value of recovered metal uranium, the reactor operator would estimate a current cost of about $9,500/kgTM, excluding the value of the recovered uranium and $7,200/kgTM including the value of the recovered uranium. At a zero percent discount rate, which is reasonable if the reactor operator wants to incorporate a risk-adjustment for long-term unknowns like geologic disposal, the current costs are about $12,700/kgTM. The value of recovered metal HEU is credited to Dounreay to make it consistent with the value of the recovered LEU at the Savannah River Site. Blend-down at Dounreay would cost about $700/kgTM on a current cost basis. Since these cost estimates are based on current costs (i.e., 1996 dollars in 1996) rather than the current fraction of a series of costs (i.e., 1/13 of 13 years' worth of constant costs over the 1996 through 2008 period at the Savannah River Site), they are exposed to escalation.

13 Belgium, France, Germany, Italy, Spain, Switzerland, United Kingdom.

The discounted canister-related cost of the packaging strategy displayed in Table F-133 is $11 million. The cost to dispose of the canisters depends on the size of the canisters and the loading levels. An estimate for disposal of full-size (i.e., commercial-type) spent nuclear fuel canisters prepared by the Idaho National Engineering Laboratory equated to $1.8 million per canister in 1994$ in 1994 (Stroupe, 1995), including transportation to the repository. This translated into $2.07 million per canister in 1996$ in 1996, the baseline cost for this EIS. These canisters contained 120 MTR-equivalents of aluminum-based spent nuclear fuel and 500 TRIGA spent nuclear fuel elements.

For the much smaller canisters and lower loading levels shown in Table F-133, a total undiscounted disposal cost (excluding transportation) of $373 million is estimated (TRW, 1995). This translates into an implied charge per canister of approximately $100 thousand for canisters containing aluminum-based spent nuclear fuel and $150 thousand for canisters containing TRIGA spent nuclear fuel. Assuming that repository development costs (1/3 of total repository charges) are incurred from 1996 through 2029 and repository emplacement costs (2/3 of total repository charges) are incurred in 2030 through 2035, the discounted cost of the disposal program (excluding the canisters) is approximately $110 million. About 95 percent of this charge is for disposal of aluminum-based spent nuclear fuel. Discounted total costs for intact disposal of aluminum-based spent nuclear fuel and TRIGA spent nuclear fuel including canister are approximately $125 million.

Total program costs are highly sensitive to the timing of disposal. Accelerating disposal to the 2015 to 2020 time period (rather than 2030 to 2035) reduces undiscounted costs by $50 million but increases discounted costs by $50 million. The savings arise from fewer years of storage prior to repository loading. The discounted cost penalty arises because the large outyear costs for repository development and emplacement lose 15 years of discounting.

F.7.2.14 Disposal of Vitrified High-level Waste

High-level waste is vitrified in the Savannah River Site Defense Waste Processing Facility. The borosilicate glass logs are inserted into waste packages (i.e., metal canisters similar to that used to dispose of commercial spent nuclear fuel) and disposed geologically. The cost to dispose of each waste package is estimated at $1.61 million, including transportation to the repository. At four Defense Waste Processing

Discounted costs for Management Alternative 2 are estimated (very roughly) at $1.25 billion. Costs for this alternative are highly speculative since there is no basis for estimating how most countries would manage their spent nuclear fuel individually or collectively or what types of facilities or approaches they would (or could) select. Of the 41 countries in the proposed foreign research reactor spent nuclear fuel program, 22 have no commercial nuclear power infrastructure to support either a storage or a reprocessing program. These 22, and most of the remaining 19, have no clear program for geologic disposal. Since no country inside or outside the proposed foreign research reactor spent nuclear fuel program has offered to store or dispose of the spent nuclear fuel from other countries, there is no obvious method by which most of the countries in the program could manage their spent nuclear fuel. The costs shown here assume substantial cost penalties from the establishment of up to 22 new spent nuclear fuel storage installations, including all supporting infrastructure.

Reprocessing at the United Kingdom Atomic Energy Authority's facility at Dounreay, Scotland is already an option for Euratom countries that can accept the return of the reprocessing waste. If the United Kingdom Atomic Energy Authority were to reprocess all the material in the foreign research reactor spent nuclear fuel program, including fuels for which it has no current commercial capability, direct costs would exceed $1 billion. Logistics would be highly problematic, however, since the United Kingdom Atomic Energy Authority would require at least 35 years to complete the task at its currently offered capacity. The limited number of other facilities that could reprocess commercial spent nuclear fuel, e.g., the French facility at Marcoule, have not made any commitments to do so. The technical and cost uncertainties associated with disposal of either spent nuclear fuel or high-level waste are entirely speculative but must be considered extremely high.

Overall, there is no basis for assuming that distributed management of the spent nuclear fuel and, in particular, distributed geologic disposal of the spent nuclear fuel or high-level waste, could be accomplished at a cost remotely resembling that of the United States or any other country with a large-scale commercial nuclear power infrastructure.

F.7.3 Interpreting the Minimum Program Costs

Table F-120 (Section F.7.1.2) showed the minimum discounted program costs for the five bounding scenarios. The table showed that for the discount rates appropriate for the U.K. and U.S. portions of the program, hybrid chemical separation/reprocessing of aluminum-based spent nuclear fuel in the United States and the United Kingdom (Management Alternative 3) was about as costly as chemical separation of aluminum-based spent nuclear fuel in the United States alone. Either of the chemical separation/reprocessing approaches was substantially less costly than storing and directly disposing of all the spent nuclear fuel in the United States.

In interpreting the minimum discounted program costs, note that important components of the costs of multiple alternatives are fixed or nearly fixed. Table F-134 shows this relationship. For example, shipping to the United States is the same whether all the spent nuclear fuel is stored or separated. This means that the differences between the costs for the key management function (i.e., storage and disposal or chemical separation and disposal) are substantially larger (in percentage terms) than the differences between the total costs of an implementation alternative. It also means that risks in the unique components of the various implementation alternatives will have an outsized impact on the relative costs of the alternatives.

Table F-134 shows that the undiscounted costs for Management Alternative 1 (storage) exceed $1.4 billion, excluding target material and all other cost and risk factors. The undiscounted costs for Management Alternative 1 (revised to incorporate chemical separation) are approximately $1 billion. Undiscounted costs for Management Alternative 3 are about $1.1 billion. A substantial portion of the cost all the aluminum-based spent nuclear fuel eliminates one or more fuel types from qualification requirements would depend on when the fuel was received, where each fuel type appeared on the prioritization for separation, and how long separation continues.

- Canyon Operating Costs - Canyon operating costs allocated to the foreign research reactor spent nuclear fuel program are at a minimum during the years when processing is incremental to processing under the Interim Management of Nuclear Materials EIS (1998 to 2002) and higher afterwards. Switching from incremental costing to average variable costing increases annual costs from as little as $1 million for 2,880 MTR-equivalents to about $32 million. Including the phase-down penalty (Section F.7.2.8) increases the cost by approximately $33 million per year. The timing of the switch from incremental costing to average variable costing (and thus the impact on the foreign research reactor spent nuclear fuel program) depends on decisions made under the Interim Management of Nuclear Materials EIS (DOE, 1995b) and a facilities utilization study underway at the Savannah River Site. The timing of any deferral penalty is subjective. It depends on whether other missions for the Canyons have been identified and whether plans to deinventory the Canyon used by the foreign research reactor spent nuclear fuel program have been developed. It is clear that Canyon operations costs allocable to the foreign research reactor spent nuclear fuel program per year or per MTR-equivalent would be much higher after 2002 than before 2002 but it is not certain how much higher or when they would become higher. This uncertainty prevents a linear estimation of separation costs according to the quantity of material processed. Section F.7.4 discusses this issue in more detail.

- Staging and Characterization Facility Capital Costs -- The Savannah River Site plans to construct a staging facility to transfer spent nuclear fuel from the existing wet basins to interim dry storage and ultimately to repository canisters. The unallocated discounted capital cost of this facility exceeds $150 million. There is no necessarily correct way to allocate the capital costs of this facility since it supports multiple components of multiple programs and is sized according to joint requirements of multiple programs. Section F.7.2.6 described the cost allocation approach used in this EIS. Approaches that could increase the costs allocated to the foreign research reactor spent nuclear fuel program are also plausible.

- Basin Operating Costs -- The Savannah River Site has estimated the costs to operate RBOF and L-Reactor disassembly basin over a roughly 10-year period at a round-the-clock operations level but has no generalized relationship that permits continuous variation in basin costs according to the number of elements received or stored. Costs depend on the timing of the receipts, the amount of characterization and canning, intra-site and inter-site transfer requirements, the variability in year-to-year staffing, and other factors.

Section F.7.4 outlines four additional groups of factors of significance in using the minimum program costs in Table F-120 as a decision basis for the program.

F.7.4 Interpreting the Other Cost Factors

Table F-120 showed the minimum discounted cost for the five bounding scenarios. The costs in Table F-120 include component contingencies but they do not include system risks, component and non-component risks, or the effects of discount rate changes.[14] Table F-121 showed these latter factors for the five scenarios. Detailed discussions are presented below. Real escalation is excluded from all costs in both tables.

F.7.4.1 Systems Integration and Logistics

The minimum program costs include the contingencies related to individual components of the program, e.g., shipping, basin operations, storage, transfers, and disposal. The minimum program costs do not include systems integration or logistics risks. The proposed foreign research reactor spent nuclear fuel program involves 41 foreign countries (a majority of which have no commercial nuclear power program), dozens of foreign ports, 13 years of receipts, up to 10 domestic ports, as many as 250 cross-country spent nuclear fuel shipments, at least two management sites, and developmental technologies (especially repository disposal technologies). Substantial systems integration bottlenecks could arise in many technical areas, e.g., insufficient casks to ship at the required rate or at the estimated loadings; vulnerability-related shutdowns at existing facilities; requirements for on-site canning prior to cask loading; unplanned requirements for dry storage characterization or conditioning; unexpected facilities requirements for meeting the repository waste acceptance criteria; delays in repository acceptance; and so forth, including normal project (not component) contingencies.

Substantial bottlenecks could also arise in many procedural areas, e.g., incompatibilities with Naval programs at the Idaho National Engineering Laboratory; requirements for on-site inspections by the International Atomic Energy Agency; constraints on shipments, duration of shipments, shipment routes, or quantities of materials shipped pursuant to agreements with the states, and so forth. Because the list of technical and procedural issues that could delay and complicate the program is both long and highly plausible, it is realistic to expect costs to increase above the component-level minimums that make up Table F-120. This risk is estimated at 10 to 15 percent of minimum discounted program costs.

F.7.4.2 Program Component Risks

Several key components of the foreign research reactor spent nuclear fuel program are uncertain. This section discusses the most important probability-adjusted uncertainties (risks).

- ***The method of disposal of spent nuclear fuel***- The base case assumption is that aluminum-based HEU spent nuclear fuel and HEU TRIGA spent nuclear fuel can be loaded into poisoned canisters and disposed at the equivalent of 14.4 kg (31.7 lb) U-235 per canister. This packing density could be unacceptable to the repository program. Processing the uranium into an isotopically neutral mass (1 percent U-235) would require construction of a new melt-and-dilute facility. Construction and operation of this facility

14 Contingencies refer to costs that are certain to occur based on historical experience with programs of similar maturity. These costs are grouped under the term "contingency" because they cannot be line-itemized. Uncertainties refer to changes in the costs of individual components or the overall program that might occur due to unknown changes in regulations, technical conditions, operational status, etc. They are assigned a probability based on their likelihood. Thus, contingencies will occur--they just cannot be line-itemized; uncertainties may occur-- they are adjusted for their probability of occurrence and expressed as risks.

could add $100 million or more to the cost of spent nuclear fuel disposal. Processing the spent nuclear fuel to avoid severe mass limitations on disposal is considered a high probability event.

- ***The adequacy of limited characterization of the spent nuclear fuel*** - There is technical uncertainty about the requirements for characterizing and conditioning the spent nuclear fuel before storing it. At the Savannah River Site, the characterization stage consists of checking the history of the spent nuclear fuel and its paperwork (documentation), visual inspection, gamma scanning (to verify the presence and amount of fissile material), and a leak detection test ("sipping") to determine if any fission products are escaping from the spent nuclear fuel elements. Canning would be limited to degraded elements only. If more extensive characterization and canning is required, new hot cells may be required. Allocable discounted costs to add and operate a hot cell at the staging facility are on the order of $100 million. The requirement for additional characterization and conditioning is a moderately probable event.

- ***Bottlenecks at the Defense Waste Processing Facility*** - Complete separation of aluminum-based spent nuclear fuel at the Savannah River Site generates about 72 Defense Waste Processing Facility logs at a cost of $1.77 million per log. The Savannah River Site estimates that for capital costs of about $100 million and operating costs of about $40 million per year, it could remove bottlenecks at the Defense Waste Processing Facility such that the cost would decline to $1.0 million per year. For the foreign research reactor spent nuclear fuel program, the allocated cost of the capital and operating requirements to relieve the bottleneck is a few million dollars. The discounted savings would be in the range of $50 million. The likelihood that the foreign research reactor spent nuclear fuel program would realize these savings is low to moderate.

- ***Failure to commercially sell the recovered uranium*** - The Savannah River Site might not be allowed to blend-down the recovered HEU for sale as power reactor fuel. DOE, for example, could choose to safeguard the HEU and isolate its chemical separation operations from the commercial power market. This would cost the foreign research reactor spent nuclear fuel program an additional $70 million. The likelihood that the foreign research reactor spent nuclear fuel program would fail to recover this value is low.

F.7.4.3 Non-Program Risks

The key non-program risk is that the cost of repository disposal increases across the board due to a change in scope (not due to escalation within the existing scope). The repository cost allocation used in this EIS assumes no monitored retrievable storage and one geologic repository. If either of these assumptions is incorrect, the cost of the repository component of the program would increase by about 20 percent. If both

A second non-program risk is that one or more of the EISs that relate to materials management and facilities use at the Savannah River Site or the Idaho National Engineering Laboratory (besides the foreign research reactor spent nuclear fuel EIS) leads to legal or regulatory action that delays all site activities and throws the foreign research reactor spent nuclear fuel program off-schedule or out of the planned facilities.

F.7.4.4 Discount Rates

This EIS uses the real discount rate specified by the Office of Management and Budget for long-term government projects evaluated in the year ending February 1996 (OMB, 1995). The specified rate, 4.9 percent, is historically high. It compares to Office of Management and Budget rates of 3.8 percent, 4.5 percent, and 2.9 percent for the years ending in February of 1993, 1994, and 1995, respectively (OMB 1992; OMB, 1993; OMB, 1994). It also compares to measured real, long-term government interest rates of 3.2 percent, 2.9 percent, 4.1 percent, and 3.4 percent (through 1995 quarter 2), respectively for the years 1992, 1993, 1994, and 1995 (FRB Cleveland, 1995). Finally, it compares to a Congressional Budget Office estimate of 2 percent for government projects independent of the period and duration (Hartman, 1990).

Unlike the United States, the United Kingdom issues some debt instruments that are the equivalent of inflation-adjusted treasury securities. In recent years, these have yielded between 2 and 5 percent. The rate as of mid January 1996 was approximately 3.6 percent. The United Kingdom is also currently

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