Appendix J
Human Health Risks

This appendix presents detailed information on the potential impacts to humans associated with incident-free (normal) releases of radioactivity from the proposed surplus plutonium disposition facilities. This information supports the human health risk assessments described in Chapter 4. In addition, site-specific input data used in the evaluation of these human health impacts are also provided or referenced where appropriate. The proposed facilities would be at one or more of four candidate U.S. Department of Energy (DOE) sites: the Hanford Site (Hanford), Idaho National Engineering and Environmental Laboratory (INEEL), the Pantex Plant (Pantex), and the Savannah River Site (SRS). Information is also presented on the human health impacts of mixed oxide (MOX) fuel lead assembly fabrication activities at five potential DOE sites: Argonne National Laboratory–West (ANL–W) at INEEL, Hanford, Lawrence Livermore National Laboratory (LLNL), Los Alamos National Laboratory (LANL), and SRS.

J.1 HANFORD

J.1.1 Assessment Data

To perform the dose assessments for the *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS), different types of data were collected and generated. In addition, calculational assumptions were made. Appendix F.10 provides a summary of the methods and tools (e.g., the GENII computer code) used for the assessments.

J.1.1.1 Meteorological Data

The meteorological data used for the Hanford dose assessments was in the form of a joint frequency data (JFD) file. A JFD file is a table that lists the percentages of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The JFD file was based on measurements taken over a period of several years at a specific location and height. Average annual meteorological conditions, averaged over the measurement period, were used for normal operations. Table J–1 presents the JFD used in the dose assessments for Hanford.

J.1.1.2 Population Data

The Hanford population distribution was based on the *1990 Census of Population and Housing Data* (DOC 1992). Projections were determined for the year 2010 (about midlife of operations) for areas within 80 km (50 mi) of the locations for the proposed surplus plutonium disposition facilities. The site population in 2010 was assumed to be representative of the population over the operational period evaluated. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances out to an 80-km (50-mi) distance. The grid was centered at the Fuels and Materials Examination Facility (FMEF) in the 400 Area, the location from which radionuclides are assumed to be released during incident-free operations. Table J–2 presents the population data used for the dose assessments at Hanford.

J.1.1.3 Agricultural Data

The 1987 Census of Agriculture was the source used to generate site-specific data for food production. Food production was spatially distributed on a circular grid similar to that used for the population distribution described previously. This food grid (or wheel) was generated by combining the fraction of a county in each segment (e.g., south, southwest, north-northeast) and the county production of the eight food categories analyzed by GENII—leafy vegetables, root vegetables, fruits, grains, beef, poultry, milk, and eggs. Each

Table J–1. Hanford 1983–1991 Joint Frequency Distributions at 61-m Height

Wind Speed (m/s)	Stability Class	Wind Blows Toward															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
0.89	A	0.12	0.1	0.08	0.11	0.14	0.15	0.1	0.08	0.14	0.08	0.05	0.06	0.07	0.05	0.05	0.07
	B	0.05	0.05	0.05	0.05	0.06	0.05	0.04	0.03	0.07	0.03	0.02	0.02	0.03	0.02	0.03	0.03
	C	0.06	0.04	0.04	0.04	0.06	0.04	0.07	0.05	0.04	0.04	0.03	0.01	0.05	0.03	0.04	0.04
	D	0.32	0.23	0.2	0.18	0.25	0.26	0.24	0.28	0.36	0.26	0.19	0.15	0.22	0.19	0.22	0.21
	E	0.19	0.14	0.1	0.1	0.13	0.13	0.14	0.19	0.37	0.22	0.18	0.17	0.23	0.19	0.19	0.19
	F	0.22	0.14	0.1	0.09	0.13	0.11	0.15	0.2	0.34	0.2	0.2	0.12	0.2	0.14	0.16	0.16
	G	0.13	0.08	0.06	0.03	0.06	0.07	0.07	0.18	0.22	0.13	0.09	0.07	0.12	0.09	0.12	0.09
2.7	A	0.32	0.28	0.28	0.28	0.39	0.37	0.37	0.34	0.55	0.32	0.16	0.09	0.17	0.13	0.13	0.15
	B	0.12	0.09	0.08	0.06	0.12	0.07	0.1	0.11	0.15	0.12	0.05	0.05	0.05	0.04	0.06	0.07
	C	0.13	0.08	0.08	0.05	0.09	0.08	0.1	0.11	0.16	0.08	0.04	0.03	0.05	0.03	0.06	0.08
	D	0.58	0.41	0.37	0.26	0.38	0.33	0.46	0.59	0.85	0.49	0.25	0.15	0.33	0.36	0.47	0.41
	E	0.32	0.2	0.19	0.12	0.21	0.21	0.25	0.45	0.68	0.46	0.31	0.24	0.37	0.29	0.38	0.33
	F	0.35	0.23	0.15	0.07	0.12	0.09	0.18	0.36	0.64	0.31	0.23	0.16	0.18	0.18	0.23	0.22
	G	0.18	0.12	0.06	0.03	0.04	0.04	0.08	0.2	0.3	0.16	0.1	0.04	0.08	0.1	0.15	0.16
4.7	A	0.39	0.31	0.21	0.1	0.13	0.13	0.15	0.19	0.77	0.51	0.17	0.13	0.19	0.15	0.16	0.17
	B	0.14	0.09	0.06	0.04	0.04	0.04	0.04	0.07	0.2	0.16	0.06	0.04	0.03	0.02	0.06	0.06
	C	0.1	0.1	0.06	0.03	0.03	0.03	0.04	0.06	0.16	0.16	0.04	0.02	0.05	0.04	0.06	0.07
	D	0.59	0.38	0.26	0.14	0.16	0.14	0.32	0.55	0.97	0.75	0.27	0.15	0.34	0.46	0.63	0.55
	E	0.41	0.21	0.15	0.09	0.1	0.11	0.28	0.6	1.02	0.71	0.37	0.27	0.5	0.53	0.6	0.43
	F	0.37	0.22	0.11	0.06	0.07	0.06	0.17	0.48	0.73	0.44	0.21	0.11	0.16	0.2	0.37	0.29
	G	0.19	0.11	0.05	0.02	0.02	0.01	0.04	0.19	0.26	0.14	0.06	0.02	0.04	0.07	0.19	0.13
7.2	A	0.22	0.17	0.08	0.02	0.02	0.01	0.03	0.05	0.32	0.63	0.28	0.17	0.23	0.11	0.19	0.15
	B	0.07	0.05	0.01	0.01	0	0	0.02	0.01	0.1	0.22	0.06	0.05	0.05	0.03	0.07	0.03
	C	0.04	0.05	0.02	0.01	0	0.01	0.02	0.02	0.07	0.18	0.06	0.04	0.03	0.03	0.05	0.04
	D	0.27	0.19	0.09	0.04	0.02	0.04	0.1	0.25	0.65	0.86	0.37	0.2	0.29	0.5	0.75	0.4
	E	0.27	0.18	0.07	0.02	0.02	0.04	0.15	0.43	0.73	0.74	0.34	0.2	0.39	0.73	0.94	0.44
	F	0.21	0.14	0.06	0.02	0.02	0.01	0.09	0.33	0.52	0.39	0.14	0.07	0.09	0.16	0.45	0.26
	G	0.13	0.08	0.04	0.01	0.01	0.01	0.03	0.11	0.19	0.13	0.04	0.02	0.01	0.04	0.14	0.13

Table J–1. Hanford 1983–1991 Joint Frequency Distributions at 61-m Height (Continued)

Wind Speed (m/s)	Stability Class	Wind Blows Toward															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
9.8	A	0.05	0.05	0.03	0.01	0	0	0	0.01	0.08	0.29	0.21	0.12	0.12	0.08	0.12	0.04
	B	0.02	0.01	0.01	0	0	0	0	0	0.02	0.08	0.04	0.04	0.04	0.02	0.03	0.02
	C	0.02	0.02	0.01	0	0	0	0	0.01	0.02	0.08	0.06	0.03	0.03	0.03	0.03	0.01
	D	0.09	0.08	0.02	0.01	0	0.01	0.03	0.04	0.24	0.58	0.32	0.16	0.19	0.33	0.57	0.14
	E	0.1	0.12	0.04	0.01	0	0.01	0.06	0.17	0.37	0.51	0.26	0.13	0.17	0.43	0.73	0.22
	F	0.1	0.11	0.03	0.01	0.01	0	0.03	0.14	0.21	0.2	0.07	0.02	0.03	0.08	0.23	0.16
	G	0.05	0.04	0.02	0	0	0	0.01	0.07	0.09	0.05	0.03	0	0	0.02	0.1	0.07
13.0	A	0.01	0.02	0	0	0	0	0	0	0.02	0.09	0.1	0.1	0.08	0.03	0.07	0.01
	B	0	0.01	0	0	0	0	0	0	0.01	0.03	0.04	0.04	0.02	0.01	0.03	0.01
	C	0	0.01	0	0	0	0	0	0	0.01	0.02	0.04	0.02	0.02	0.01	0.02	0.01
	D	0.03	0.03	0.01	0	0	0	0.01	0.02	0.07	0.27	0.24	0.12	0.09	0.19	0.32	0.05
	E	0.04	0.08	0.03	0.01	0	0	0.02	0.05	0.13	0.32	0.25	0.1	0.07	0.2	0.33	0.07
	F	0.04	0.05	0.02	0.01	0	0	0.02	0.06	0.08	0.13	0.05	0.01	0.01	0.02	0.1	0.06
	G	0.01	0.01	0	0	0	0	0	0.02	0.02	0.03	0.01	0	0	0.01	0.05	0.04
16.0	A	0	0.01	0	0	0	0	0	0	0	0.02	0.06	0.03	0.02	0.01	0.01	0
	B	0	0.01	0	0	0	0	0	0	0	0.01	0.02	0.01	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0.01	0.02	0.01	0.01	0	0.01	0
	D	0.02	0.03	0.01	0.01	0	0	0	0.01	0.01	0.11	0.19	0.06	0.03	0.06	0.1	0.01
	E	0.01	0.04	0.03	0	0	0	0.01	0.02	0.05	0.16	0.16	0.04	0.02	0.04	0.09	0.01
	F	0.01	0.03	0	0	0	0	0	0.03	0.04	0.05	0.02	0	0.01	0	0.01	0.02
	G	0	0	0	0	0	0	0	0.02	0.02	0.02	0	0	0	0	0.02	0
19.0	A	0.02	0.03	0	0	0	0	0	0	0	0.01	0.05	0.01	0.01	0	0.01	0
	B	0	0.03	0	0	0	0	0	0	0	0	0.02	0	0	0	0	0
	C	0.01	0.02	0	0	0	0	0	0	0	0	0.03	0	0	0	0	0
	D	0.03	0.09	0	0	0	0	0	0	0	0.09	0.22	0.04	0.03	0.01	0.02	0
	E	0.03	0.1	0.02	0	0	0	0	0.02	0.02	0.1	0.14	0.02	0.01	0.01	0.01	0
	F	0.02	0.04	0.01	0	0	0	0	0.03	0.03	0.04	0.02	0	0	0	0.01	0
	G	0	0.01	0	0	0	0	0	0.02	0.02	0.02	0	0	0	0	0.01	0

Source: Neitzel 1996.

county's food production was assumed to be distributed uniformly over the given county's land area. These categorized food wheels were then used in the assessment of doses to the Hanford population from the ingestion pathway. The consumption rates used in the dose assessments were those for the maximally exposed individual (MEI) and average exposed individual. People living within the 80-km (50-mi) assessment area were assumed to consume only food grown in that area. Hanford food production and consumption data used for the dose assessments in the SPD EIS were obtained from the *Health Risk Data for Storage and Disposition Final PEIS* (HNUS 1996).

Table J–2. Projected Hanford Population Surrounding FMEF for Year 2010

Direction	Distance (mi)										Total
	0–1	1–2	2–3	3–4	4–5	5–10	10–20	20–30	30–40	40–50	
S	0	0	0	0	0	4,265	44,747	1,141	7,041	19,608	**76,802**
SSW	0	0	0	0	2	1,515	2,758	438	2,976	3,951	**11,640**
SW	0	0	0	0	42	1,388	4,788	316	227	2,047	**8,808**
WSW	0	0	0	0	0	54	2,387	17,154	3,588	325	**23,508**
W	0	0	0	0	0	0	766	6,201	28,142	15,966	**51,075**
WNW	0	0	0	0	0	0	5	879	1,233	9,074	**11,191**
NW	0	0	0	0	0	0	0	645	411	178	**12,34**
NNW	0	0	0	0	0	0	0	1,097	1,437	1,491	**4,025**
N	0	0	0	0	0	0	0	1,153	3,773	2,749	**7,675**
NNE	0	0	0	0	0	18	468	5,523	1,514	25,879	**33,402**
NE	0	0	0	0	0	95	827	7,348	3,019	1,256	**12,545**
ENE	0	0	0	0	0	345	1,544	3,737	423	446	**6,495**
E	0	0	0	0	0	425	948	451	351	327	**2,502**
ESE	0	0	0	0	0	434	655	347	266	326	**2,028**
SE	0	0	0	0	0	419	1,313	1,736	396	1,459	**5,323**
SSE	0	0	0	0	0	6,989	87,249	33,689	608	986	**129,521**
Total	**0**	**0**	**0**	**0**	**44**	**15,947**	**148,455**	**81,855**	**55,405**	**86,068**	**387,774**

Key: FMEF, Fuels and Materials Examination Facility.
Source: DOC 1992.

J.1.1.4 Source Term Data

Estimated incident-free radiological releases associated with the pit conversion, immobilization, and MOX facilities are presented in Tables J–3 through J–5. Stack heights and release locations are provided in the facility data reports (DOE 1999; UC 1998a, 1998b, 1999a, 1999b).

Table J–3. Estimated Incident-Free Annual Radiological Releases From the Pit Conversion Facility at Hanford

Isotope	(μCi/yr)
Plutonium 236	9.3×10^{-11}
Plutonium 238	0.065
Plutonium 239	0.69
Plutonium 240	0.18
Plutonium 241	0.69
Plutonium 242	4.8×10^{-5}
Americium 241	0.37
Hydrogen 3	1.1×10^{9}

Source: UC 1998a.

Table J–4. Estimated Incident-Free Annual Radiological Releases From the Immobilization Facility at Hanford

Isotope	Ceramic (17 t) (μCi/yr)	Ceramic (50 t) (μCi/yr)	Glass (17 t) (μCi/yr)	Glass (50 t) (μCi/yr)
Plutonium 236	–	–	–	–
Plutonium 238	–	0.57	–	0.52
Plutonium 239	3.7	9.5	3.4	8.6
Plutonium 240	1.7	3.1	1.6	2.8
Plutonium 241	110	100	98	93
Plutonium 242	1.3×10^{-3}	1.6×10^{-3}	1.2×10^{-3}	1.5×10^{-3}
Americium 241	2.3	5.4	2.2	5.0
Uranium 234	–	–	–	–
Uranium 235	1.1×10^{-5}	4.5×10^{-5}	2.3×10^{-6}	2.3×10^{-6}
Uranium 238	8.8×10^{-5}	3.5×10^{-4}	1.9×10^{-5}	1.9×10^{-5}

Source: UC 1999a, 1999b.

Table J–5. Estimated Incident-Free Annual Radiological Releases From the MOX Facility at Hanford

Isotope	(μCi/yr)
Plutonium 236	1.3×10^{-8}
Plutonium 238	8.5
Plutonium 239	91
Plutonium 240	23
Plutonium 241	101
Plutonium 242	6.1×10^{-3}
Americium 241	48
Uranium 234	5.1×10^{-3}
Uranium 235	2.1×10^{-4}
Uranium 238	0.012

Source: UC 1998b.

J.1.1.5 Other Calculational Assumptions

To estimate radiological impacts of incident-free operation of the proposed facilities at Hanford, the following additional assumptions and factors were considered, in accordance with the guidelines established in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.109 (NRC 1977).

Ground surfaces were assumed to have no previous deposition of radionuclides for the purposes of modeling the incremental radiological impacts associated with surplus plutonium disposition activities. However, doses associated with true instances of prior deposition are accounted for in the Affected Environment and Cumulative Impacts sections.

The annual external exposure time to the plume and to soil contamination was 0.7 year for the MEI (NRC 1977).

The annual external exposure time to the plume and to soil contamination was 0.5 year for the population (NRC 1977).

The annual inhalation exposure time to the plume was 1 year for the MEI and general population (NRC 1977).

The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of the adult human.

A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of contaminated animal products. Drinking water, aquatic food ingestion, and any other pathway that may involve liquid exposure were not examined because all releases are to the air.

Reported stack heights were used for atmospheric releases. The resultant doses were conservative as use of the actual stack height instead of the effective stack height negates plume rise.

The calculated doses are 50-year committed doses from 1 year of intake.

J.1.2 Facilities

The following sections present all viable radiological impact scenarios that could be associated with different combinations of incident-free facility operations at Hanford.

J.1.2.1 Pit Conversion Facility

J.1.2.1.1 Construction of Pit Conversion Facility

No radiological risk would be incurred by members of the public from construction and modification of a pit conversion facility at Hanford. According to recent surveys conducted in the 400 Area, a construction worker would not be expected to receive any additional dose above natural background levels (Antonio 1998). Nonetheless, if deemed necessary, workers may be monitored (badged) as a precautionary measure.

J.1.2.1.2 Operation of Pit Conversion Facility

Tables J–6 and J–7 present the incident-free radiological impacts of the operation of a pit conversion facility at Hanford.

Table J–6. Potential Radiological Impacts on the Public of Operation of Pit Conversion Facility in FMEF at Hanford

Population within 80 km for year 2010	
Dose (person-rem)	6.9
Percent of natural background[a]	5.9×10^{-3}
10-year latent fatal cancers	0.034
Maximally exposed individual	
Annual dose (mrem)	0.017
Percent of natural background[a]	5.7×10^{-3}
10-year latent fatal cancer risk	8.5×10^{-8}
Average exposed individual within 80 km[b]	
Annual dose (mrem)	0.017
10-year latent fatal cancer risk	8.5×10^{-8}

[a] The annual natural background radiation level at Hanford is 300 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 116,300 person-rem.
[b] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of Hanford in 2010 (387,800).
Key: FMEF, Fuels and Materials Examination Facility.
Source: Model results.

Table J–7. Potential Radiological Impacts on Involved Workers of Operation of Pit Conversion Facility in FMEF at Hanford

Number of badged workers	383
Total dose (person-rem/yr)	192
10-year latent fatal cancers	0.77
Average worker dose (mrem/yr)	500
10-year latent fatal cancer risk	2.0×10^{-3}

Key: FMEF, Fuels and Materials Examination Facility.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

J.1.2.2 Immobilization Facility

J.1.2.2.1 Construction of Immobilization Facility

No radiological risk would be incurred by members of the public from the construction and modification of an immobilization (ceramic or glass) facility at Hanford. According to recent radiation surveys conducted in the 400 Area, a construction worker would not be expected to receive any additional dose above natural background levels (Antonio 1998). Nonetheless, if deemed necessary, workers may be monitored (badged) as a precautionary measure.

J.1.2.2.2 Operation of Immobilization Facility

Tables J–8 and J–9 present all possible incident-free radiological impact scenarios for the operation of a ceramic or glass immobilization facility at Hanford.

Table J–8. Potential Radiological Impacts on the Public of Operation of Immobilization Facility in FMEF at Hanford

Impact	17 t		50 t	
	Ceramic	Glass	Ceramic	Glass
Population within 80 km for year 2010				
Dose (person-rem)	7.8×10^{-3}	7.1×10^{-3}	0.016	0.015
Percent of natural background[a]	6.7×10^{-6}	6.1×10^{-6}	1.4×10^{-5}	1.3×10^{-5}
10-year latent fatal cancers	3.9×10^{-5}	3.6×10^{-5}	8.0×10^{-5}	7.5×10^{-5}
Maximally exposed individual				
Annual dose (mrem)	1.1×10^{-4}	9.7×10^{-5}	2.2×10^{-4}	2.0×10^{-4}
Percent of natural background[a]	3.7×10^{-5}	3.2×10^{-5}	7.3×10^{-5}	6.7×10^{-5}
10-year latent fatal cancer risk	5.5×10^{-10}	4.9×10^{-10}	1.1×10^{-9}	1.0×10^{-9}
Average exposed individual within 80 km[b]				
Annual dose (mrem)	2.0×10^{-5}	1.8×10^{-5}	4.1×10^{-5}	3.9×10^{-5}
10-year latent fatal cancer risk	1.0×10^{-10}	9.0×10^{-11}	2.1×10^{-10}	2.0×10^{-10}

[a] The annual natural background radiation level at Hanford is 300 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 116,300 person-rem.

[b] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of Hanford in 2010 (387,800).

Key: FMEF, Fuels and Materials Examination Facility.

Source: Model results.

Table J–9. Potential Radiological Impacts on Involved Workers of Operation of Immobilization Facility in FMEF at Hanford[a]

Impact	17 t		50 t	
	Ceramic	Glass	Ceramic	Glass
Number of badged workers	365	365	397	397
Total dose (person-rem/yr)	274	274	298	298
10-year latent fatal cancers	1.1	1.1	1.2	1.2
Average worker dose (mrem/yr)	750	750	750	750
10-year latent fatal cancer risk	3.0×10^{-3}	3.0×10^{-3}	3.0×10^{-3}	3.0×10^{-3}

[a] The presented values are representative of the largest possible number of workers regardless of collocation considerations.

Key: FMEF, Fuels and Materials Examination Facility.

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: UC 1999a, 1999b.

J.1.2.3 MOX Facility

J.1.2.3.1 Construction of MOX Facility

No radiological risk would be incurred by members of the public from the construction and modification of a MOX facility at Hanford. According to recent radiation surveys conducted in the 400 Area, a construction worker would not be expected to receive any additional dose above natural background levels (Antonio 1998). Nonetheless, if deemed necessary, workers may be monitored (badged) as a precautionary measure.

J.1.2.3.2 Operation of MOX Facility

Tables J–10 and J–11 present the incident-free radiological impacts of the operation of a MOX facility at Hanford. The facility would either be located within the existing FMEF or a new facility would be built adjacent to FMEF.

Table J–10. Potential Radiological Impacts on the Public of Operation of MOX Facility in FMEF or New Construction at Hanford

Impact	FMEF[a]	New[a]
Population dose within 80 km for year 2010		
Dose (person-rem)	0.14	0.29
Percent of natural background[b]	1.2×10^{-4}	2.5×10^{-4}
10-year latent fatal cancers	6.9×10^{-4}	1.5×10^{-3}
Maximally exposed individual		
Annual dose (mrem)	1.8×10^{-3}	4.8×10^{-3}
Percent of natural background[b]	6.1×10^{-4}	1.6×10^{-3}
10-year latent fatal cancer risk	9.3×10^{-9}	2.4×10^{-8}
Average exposed individual within 80 km[c]		
Annual dose (mrem)	3.5×10^{-4}	7.5×10^{-4}
10-year latent fatal cancer risk	1.7×10^{-9}	3.7×10^{-9}

[a] The difference in impacts is attributable to different stack heights. As described in Section 4.26.1.2.2, Water Resources, no component was attributed to liquid pathways because it is not expected that significant contamination could reach these pathways given the site's groundwater and surface-water characteristics.

[b] The annual natural background radiation level at Hanford is 300 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 116,300 person-rem.

[c] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of Hanford in 2010 (387,800).

Key: FMEF, Fuels and Materials Examination Facility.

Source: Model results.

Table J–11. Potential Radiological Impacts on Involved Workers of Operation of MOX Facility in FMEF or New Construction at Hanford

Number of badged workers	331
Total dose (person-rem/yr)	22
10-year latent fatal cancers	0.088
Average worker dose (mrem/yr)	65
10-year latent fatal cancer risk	2.6×10^{-4}

Key: FMEF, Fuels and Materials Examination Facility.

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: DOE 1999; UC 1998b.

J.1.2.4 Pit Conversion and Immobilization Facilities

J.1.2.4.1 Construction of Pit Conversion and Immobilization Facilities

No radiological risk would be incurred by members of the public from the construction and modification of pit conversion and immobilization (ceramic or glass) facilities at Hanford. According to recent radiation surveys conducted in the 400 Area, a construction worker would not be expected to receive any additional dose above

natural background levels (Antonio 1998). Nonetheless, if deemed necessary, workers may be monitored (badged) as a precautionary measure.

J.1.2.4.2 Operation of Pit Conversion and Immobilization Facilities

Tables J–12 and J–13 present all possible incident-free radiological impact scenarios for the operation of the pit conversion and immobilization facilities at Hanford.

Table J–12. Potential Radiological Impacts on the Public of Operation of Pit Conversion and Immobilization Facilities in FMEF at Hanford

Impact	Pit Conversion	Immobilization (50 t) Ceramic	Immobilization (50 t) Glass	Total[a]
Population within 80 km for year 2010				
Dose (person-rem)	6.9	0.016	0.015	6.9
Percent of natural background[b]	5.9×10^{-3}	1.4×10^{-5}	1.3×10^{-5}	5.9×10^{-3}
10-year latent fatal cancers	0.034	8.0×10^{-5}	7.5×10^{-5}	0.034
Maximally exposed individual				
Annual dose (mrem)	0.017	2.2×10^{-4}	2.0×10^{-4}	0.017
Percent of natural background[b]	5.7×10^{-3}	7.3×10^{-5}	6.7×10^{-5}	5.8×10^{-3}
10-year latent fatal cancer risk	8.5×10^{-8}	1.1×10^{-9}	1.0×10^{-9}	8.6×10^{-8}
Average exposed individual within 80 km[c]				
Annual dose (mrem)	0.017	4.1×10^{-5}	3.9×10^{-5}	0.017
10-year latent fatal cancer risk	8.5×10^{-8}	2.1×10^{-10}	2.0×10^{-10}	8.5×10^{-8}

[a] Totals represent the largest possible sums for each public category. Totals are additive in all cases because the same groups or individuals would receive doses from both facilities.
[b] The annual natural background radiation level at Hanford is 300 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 116,300 person-rem.
[c] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of Hanford in 2010 (387,800).
Key: FMEF, Fuels and Materials Examination Facility.
Source: Model results.

Table J–13. Potential Radiological Impacts on Involved Workers of Operation of Pit Conversion and Immobilization Facilities in FMEF at Hanford

Impact	Pit Conversion	Immobilization (50 t)[a] Ceramic or Glass	Total
Number of badged workers	383	397	780
Total dose (person-rem/yr)	192	298	490
10-year latent fatal cancers	0.77	1.2	2.0
Average worker dose (mrem/yr)	500	750	628[b]
10-year latent fatal cancer risk	2.0×10^{-3}	3.0×10^{-3}	2.5×10^{-3}

[a] The presented values are representative of the largest possible number of workers regardless of collocation considerations.
[b] Represents an average of the doses for both facilities.
Key: FMEF, Fuels and Materials Examination Facility.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: UC 1998a, 1999a, 1999b.

J.1.2.5 Pit Conversion and MOX Facilities

J.1.2.5.1 Construction of Pit Conversion and MOX Facilities

No radiological risk would be incurred by members of the public from the modification of FMEF for pit disassembly and conversion and MOX fuel fabrication or construction of new MOX facility at Hanford. According to recent radiation surveys conducted in the 400 Area, a construction worker would not be expected to receive any additional dose above natural background levels (Antonio 1998). Nonetheless, if deemed necessary, workers may be monitored (badged) as a precautionary measure.

J.1.2.5.2 Operation of Pit Conversion and MOX Facilities

Tables J–14 and J–15 present the incident-free radiological impacts of the operation of the pit conversion and MOX facilities at Hanford.

Table J–14. Potential Radiological Impacts on the Public of Operation of Pit Conversion and MOX Facilities in FMEF or New MOX Facility at Hanford

Impact	Pit Conversion	MOX[a]		Total[b]
		FMEF	New	
Population within 80 km for year 2010				
Dose (person-rem)	6.9	0.14	0.29	7.2
Percent of natural background[c]	5.9×10^{-3}	1.2×10^{-4}	2.5×10^{-4}	6.2×10^{-3}
10-year latent fatal cancers	0.034	7.0×10^{-4}	1.5×10^{-3}	0.036
Maximally exposed individual				
Annual dose (mrem)	0.017	1.8×10^{-3}	4.8×10^{-3}	0.022
Percent of natural background[c]	5.7×10^{-3}	6.1×10^{-4}	1.6×10^{-3}	7.3×10^{-3}
10-year latent fatal cancer risk	8.5×10^{-8}	9.3×10^{-9}	2.4×10^{-8}	1.1×10^{-7}
Average exposed individual within 80 km[d]				
Annual dose (mrem)	0.017	3.5×10^{-4}	7.5×10^{-4}	0.018
10-year latent fatal cancer risk	8.5×10^{-8}	1.7×10^{-9}	3.7×10^{-9}	8.9×10^{-8}

[a] As described in Section 4.26.1.2.2, Water Resources, no component was attributed to liquid pathways because it is not expected that significant contamination could reach these pathways given the site's groundwater and surface-water characteristics.
[b] Totals represent the largest possible sums for each public category. Totals are additive in all cases because the same groups or individuals would receive doses from both facilities.
[c] The annual natural background radiation level at Hanford is 300 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 116,300 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of Hanford in 2010 (387,800).
Key: FMEF, Fuels and Materials Examination Facility.
Source: Model results.

Table J–15. Potential Radiological Impacts on Involved Workers of Operation of Pit Conversion and MOX Facilities in FMEF or New MOX Facility at Hanford

Impact	Pit Conversion	MOX (FMEF or New)	Total
Number of badged workers	383	331	714
Total dose (person-rem/yr)	192	22	214
10-year latent fatal cancers	0.77	0.088	0.86
Average worker dose (mrem/yr)	500	65	300[a]
10-year latent fatal cancer risk	2.0×10^{-3}	2.6×10^{-4}	1.2×10^{-3}

[a] Represents an average of the doses for both facilities.

Key: FMEF, Fuels and Materials Examination Facility.

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: DOE 1999; UC 1998a, 1998b.

J.1.2.6 Immobilization and MOX Facilities

J.1.2.6.1 Construction of Immobilization and MOX Facilities

No radiological risk would be incurred by members of the public from the modification of FMEF for collocating plutonium conversion and immobilization (ceramic or glass) and MOX fuel fabrication or construction of a new MOX facility at Hanford. According to recent radiation surveys conducted in the 400 Area, a construction worker would not be expected to receive any additional dose above natural background levels (Antonio 1998). Nonetheless, if deemed necessary, workers may be monitored (badged) as a precautionary measure.

J.1.2.6.2 Operation of Immobilization and MOX Facilities

Tables J–16 and J–17 present the incident-free radiological impacts of the operation of the immobilization and MOX facilities at Hanford.

Table J–16. Potential Radiological Impacts on the Public of Operation of Collocating Immobilization and MOX Facilities in FMEF or New MOX Facility at Hanford

Impact	Immobilization (17 t)		MOX[a]		Total[b]
	Ceramic	Glass	FMEF	New	
Population within 80 km for year 2010					
Dose (person-rem)	7.8×10^{-3}	7.1×10^{-3}	0.14	0.29	0.30
Percent of natural background[c]	6.7×10^{-6}	6.1×10^{-6}	1.2×10^{-4}	2.5×10^{-4}	2.6×10^{-4}
10-year latent fatal cancers	3.9×10^{-5}	3.6×10^{-5}	6.9×10^{-4}	1.5×10^{-3}	1.5×10^{-3}
Maximally exposed individual					
Annual dose (mrem)	1.1×10^{-4}	9.7×10^{-5}	1.8×10^{-3}	4.8×10^{-3}	4.9×10^{-3}
Percent of natural background[c]	3.7×10^{-5}	3.2×10^{-5}	6.1×10^{-4}	1.6×10^{-3}	1.6×10^{-3}
10-year latent fatal cancer risk	5.5×10^{-10}	4.9×10^{-10}	9.3×10^{-9}	2.4×10^{-8}	2.5×10^{-8}
Average exposed individual within 80 km[d]					
Annual dose (mrem)	2.0×10^{-5}	1.8×10^{-5}	3.5×10^{-4}	7.5×10^{-4}	7.7×10^{-4}
10-year latent fatal cancer risk	1.0×10^{-10}	9.0×10^{-11}	1.7×10^{-9}	3.7×10^{-9}	3.9×10^{-9}

[a] As described in Section 4.26.1.2.2, Water Resources, no component was attributed to liquid pathways because it is not expected that significant contamination could reach these pathways given the site's groundwater and surface-water characteristics.
[b] Totals represent the largest possible sums for each public category. Totals are additive in all cases because the same groups or individuals would receive doses from both facilities.
[c] The annual natural background radiation level at Hanford is 300 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 116,300 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of Hanford in 2010 (387,800).
Key: FMEF, Fuels and Materials Examination Facility.
Source: Model results.

Table J–17. Potential Radiological Impacts on Involved Workers of Operation of Collocating Immobilization and MOX Facilities in FMEF or New MOX Facility at Hanford

Impact	Immobilization (17 t)[a] Ceramic or Glass	MOX (FMEF or New)	Total
Number of badged workers	365	331	696
Total dose (person-rem/yr)	274	22	296
10-year latent fatal cancers	1.1	0.088	1.2
Average worker dose (mrem/yr)	750	65	425[b]
10-year latent fatal cancer risk	3.0×10^{-3}	2.6×10^{-4}	1.7×10^{-3}

[a] The presented values are representative of the largest possible number of workers regardless of collocation considerations.
[b] Represents an average of the doses for both facilities.
Key: FMEF, Fuels and Materials Examination Facility.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: DOE 1999; UC 1998b, 1999a, 1999b.

J.1.2.7 Pit Conversion, Immobilization, and MOX Facilities

J.1.2.7.1 Construction of Pit Conversion, Immobilization, and MOX Facilities

No radiological risk would be incurred by members of the public from the modification of FMEF for pit disassembly and conversion and plutonium conversion and immobilization (ceramic or glass) and construction of a new MOX facility at Hanford. According to recent radiation surveys conducted at the 400 Area, a construction worker would not be expected to receive any additional dose above natural background levels (Antonio 1998). Nonetheless, if deemed necessary, workers may be monitored (badged) as a precautionary measure.

J.1.2.7.2 Operation of Pit Conversion, Immobilization, and MOX Facilities

Tables J–18 and J–19 present all possible incident-free radiological impact scenarios for operating all three facilities at Hanford.

Table J–18. Potential Radiological Impacts on the Public of Operation of Pit Conversion and Immobilization Facilities in FMEF and New MOX Facility at Hanford

Impact	Pit Conversion	Immobilization (17 t)		MOX[a]		Total[b]
		Ceramic	Glass	FMEF	New	
Population within 80 km for year 2010						
Dose (person-rem)	6.9	7.8×10^{-3}	7.1×10^{-3}	0.14	0.29	7.2
Percent of natural background[c]	5.9×10^{-3}	6.7×10^{-6}	6.1×10^{-6}	1.2×10^{-4}	2.5×10^{-4}	6.2×10^{-3}
10-year latent fatal cancers	0.034	3.9×10^{-5}	3.6×10^{-5}	6.9×10^{-4}	1.5×10^{-3}	0.036
Maximally exposed individual						
Annual dose (mrem)	0.017	1.1×10^{-4}	9.7×10^{-5}	1.8×10^{-3}	4.8×10^{-3}	0.022
Percent of natural background[c]	5.7×10^{-3}	3.7×10^{-5}	3.2×10^{-5}	6.1×10^{-4}	1.6×10^{-3}	7.3×10^{-3}
10-year latent fatal cancer risk	8.5×10^{-8}	5.5×10^{-10}	4.9×10^{-10}	9.3×10^{-9}	2.4×10^{-8}	1.1×10^{-7}
Average exposed individual within 80 km[d]						
Annual dose (mrem)	0.017	2.0×10^{-5}	1.8×10^{-5}	3.5×10^{-4}	7.5×10^{-4}	0.018
10-year latent fatal cancer risk	8.5×10^{-8}	1.0×10^{-10}	9.0×10^{-11}	1.7×10^{-9}	3.7×10^{-9}	8.9×10^{-8}

[a] As described in Section 4.26.1.2.2, Water Resources, no component was attributed to liquid pathways because it is not expected that significant contamination could reach these pathways given the site's groundwater and surface-water characteristics.
[b] Totals represent the largest possible sums for each public category. Totals are additive in all cases because the same groups or individuals would receive doses from all three facilities.
[c] The annual natural background radiation level at Hanford is 300 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 116,300 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of Hanford in 2010 (387,800).
Key: FMEF, Fuels and Materials Examination Facility.
Source: Model results.

Table J–19. Potential Radiological Impacts on Involved Workers of Operation of Pit Conversion and Immobilization Facilities in FMEF and New MOX Facility at Hanford

Impact	Pit Conversion	Immobilization (17 t)[a] Ceramic or Glass	MOX (FMEF or New)	Total
Number of badged workers	383	365	331	1,079
Total dose (person-rem/yr)	192	274	22	488
10-year latent fatal cancers	0.77	1.1	0.088	2.0
Average worker dose (mrem/yr)	500	750	65	452[b]
10-year latent fatal cancer risk	2.0×10^{-3}	3.0×10^{-3}	2.6×10^{-4}	1.8×10^{-3}

[a] The presented values are representative of the largest possible number of workers regardless of collocation considerations.
[b] Represents an average of the doses for all three facilities.

Key: FMEF, Fuels and Materials Examination Facility.

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: DOE 1999; UC 1998b, 1999a, 1999b.

J.2 INEEL

J.2.1 Assessment Data

To perform the dose assessments for the SPD EIS, different types of data were collected and generated. In addition, calculational assumptions were made. Appendix F.10 provides a summary of the methods and tools (e.g., the GENII computer code) that were used for the assessments.

J.2.1.1 Meteorological Data

The meteorological data used for the INEEL dose assessments was in the form of JFD file. A JFD file is a table listing the percentages of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The JFD file was based on measurements taken over a period of several years at a specific location and height. Average annual meteorological conditions, averaged over the measurement period, were used for normal operations. Table J–20 presents the JFD used in the dose assessments for INEEL.

J.2.1.2 Population Data

The INEEL population distribution was based on the *1990 Census of Population and Housing Data* (DOC 1992). Projections were determined for the year 2010 (about midlife of operations) for areas within 80 km (50 mi) of the locations for the proposed surplus plutonium disposition facilities. The site population in 2010 was assumed to be representative of the population over the operational period evaluated. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances out to an 80-km (50-mi) distance. The grid was centered at the Idaho Nuclear Technology and Engineering Center (INTEC), the location from which radionuclides are assumed to be released during incident-free operations. Table J–21 presents the population data used for the dose assessments at INEEL.

J.2.1.3 Agricultural Data

The 1987 Census of Agriculture was the source used to generate site-specific data for food production. Food production was spatially distributed on a circular grid similar to that used for the population distribution described previously. This food grid (or wheel) was generated by combining the fraction of a county in each segment (e.g., south, southwest, north-northeast) and the county production of the eight food categories analyzed by GENII–leafy vegetables, root vegetables, fruits, grains, beef, poultry, milk, and eggs. Each county's food production was assumed to be distributed uniformly over the given county's land area. These categorized food wheels were then used in the assessment of doses to the INEEL population from the ingestion pathway. The consumption rates used in the dose assessments were those for the MEI and average exposed individual. People living within the 80-km (50-mi) assessment area were assumed to consume only food grown in that area. INEEL food production and consumption data used for the dose assessments in the SPD EIS were obtained from the *Health Risk Data for Storage and Disposition Final PEIS* (HNUS 1996).

Table J–20. INEEL 1987–1991 Joint Frequency Distributions at 61-m Height

Wind Speed (m/s)	Stability Class	Wind Blows Toward															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
1.0	A	0.2	0.31	0.28	0.21	0.2	0.19	0.24	0.22	0.17	0.16	0.11	0.11	0.1	0.11	0.09	0.15
	B	0.04	0.06	0.03	0.01	0.01	0.01	0.01	0.02	0.03	0.02	0.01	0.01	0.01	0	0	0.01
	C	0.04	0.07	0.07	0.02	0.01	0.01	0.01	0.01	0.01	0.01	0.02	0.01	0.01	0.01	0.01	0.01
	D	0.15	0.26	0.15	0.08	0.03	0.05	0.04	0.07	0.07	0.07	0.04	0.05	0.05	0.05	0.05	0.08
	E	0.14	0.17	0.15	0.08	0.07	0.07	0.04	0.06	0.05	0.07	0.06	0.04	0.04	0.05	0.06	0.06
	F	0.4	0.46	0.44	0.3	0.23	0.2	0.16	0.18	0.13	0.16	0.15	0.16	0.17	0.16	0.18	0.27
2.5	A	0.25	0.45	0.58	0.49	0.4	0.34	0.31	0.49	0.63	0.66	0.57	0.32	0.24	0.14	0.18	0.18
	B	0.06	0.18	0.21	0.11	0.03	0.02	0.02	0.05	0.08	0.12	0.08	0.05	0.03	0.01	0.01	0.02
	C	0.15	0.35	0.4	0.09	0.02	0.01	0.02	0.05	0.11	0.1	0.12	0.03	0.04	0.02	0.01	0.03
	D	0.55	1.78	1.05	0.2	0.07	0.04	0.08	0.1	0.17	0.3	0.32	0.2	0.1	0.07	0.08	0.12
	E	0.32	0.75	0.52	0.15	0.07	0.04	0.06	0.09	0.09	0.17	0.15	0.18	0.07	0.06	0.07	0.09
	F	0.77	1.65	1.38	0.67	0.34	0.24	0.21	0.27	0.31	0.51	0.47	0.48	0.35	0.32	0.34	0.38
4.5	A	0.02	0.05	0.05	0.03	0.02	0.01	0.02	0.04	0.08	0.1	0.09	0.08	0.02	0.02	0.02	0.01
	B	0.07	0.12	0.16	0.09	0.04	0.03	0.04	0.12	0.2	0.39	0.4	0.2	0.1	0.05	0.08	0.06
	C	0.07	0.19	0.33	0.13	0.02	0.02	0.02	0.08	0.14	0.33	0.58	0.21	0.07	0.05	0.03	0.06
	D	0.45	2.59	2.36	0.33	0.07	0.05	0.08	0.22	0.36	0.91	1.18	0.7	0.22	0.12	0.12	0.21
	E	0.34	1.26	0.93	0.17	0.04	0.03	0.06	0.11	0.21	0.34	0.49	0.38	0.15	0.08	0.12	0.17
	F	0.35	1.2	1.25	0.37	0.12	0.06	0.04	0.15	0.17	0.33	0.43	0.34	0.18	0.08	0.12	0.16
6.9	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0.01	0	0	0	0	0	0
	C	0.06	0.07	0.08	0.03	0.02	0.01	0.02	0.07	0.1	0.23	0.46	0.27	0.1	0.04	0.05	0.04
	D	0.67	1.47	1.6	0.35	0.06	0.03	0.08	0.26	0.4	1.28	2.95	1.78	0.44	0.16	0.08	0.4
	E	0.15	0.8	0.8	0.16	0.03	0.01	0.06	0.13	0.13	0.33	0.88	0.69	0.11	0.02	0.01	0.08
	F	0.05	0.2	0.25	0.07	0.01	0.01	0	0.02	0.02	0.01	0.1	0.11	0.01	0.01	0	0.01
9.6	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0.01	0.01	0.01	0	0	0
	D	0.64	0.61	0.74	0.16	0.02	0.01	0.04	0.16	0.29	1.1	3.53	1.98	0.38	0.12	0.07	0.26
	E	0.03	0.12	0.17	0.07	0	0	0.01	0.03	0.03	0.06	0.37	0.28	0.04	0.01	0	0
	F	0	0	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0
13.2	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	D	0.25	0.25	0.18	0.05	0	0	0.02	0.08	0.16	0.55	2.88	2.13	0.18	0.11	0.01	0.05
	E	0	0	0	0	0	0	0	0	0	0	0.01	0.01	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Table J–20. INEEL 1987–1991 Joint Frequency Distributions at 61-m Height (Continued)

Wind Speed (m/s)	Stability Class	Wind Blows Toward															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
19.0	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	D	0.01	0.05	0.01	0.01	0	0	0	0	0	0.04	0.47	0.48	0.01	0.01	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
25.0	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	D	0	0	0	0	0	0	0	0	0	0	0	0.01	0	0	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Source: Sagendorf 1992.

Table J–21. Projected INEEL Population Surrounding INTEC for Year 2010

Direction	Distance (mi)										Total
	0–1	1–2	2–3	3–4	4–5	5–10	10–20	20–30	30–40	40–50	
S	0	0	0	0	0	32	204	340	1,222	3,624	**5,422**
SSW	0	0	0	0	0	22	92	182	335	445	**1,076**
SW	0	0	0	0	0	22	87	117	163	304	**693**
WSW	0	0	0	0	0	0	87	136	149	262	**634**
W	0	0	0	0	0	0	87	180	392	280	**939**
WNW	0	0	0	0	0	0	269	519	445	311	**1,544**
NW	0	0	0	0	0	6	384	620	772	720	**2,502**
NNW	0	0	0	0	0	6	96	97	315	173	**687**
N	0	0	0	0	0	0	25	45	77	100	**247**
NNE	0	0	0	0	0	0	25	48	170	161	**404**
NE	0	0	0	0	0	0	0	285	652	342	**1,279**
ENE	0	0	0	0	0	0	0	332	575	1,057	**1,964**
E	0	0	0	0	0	0	0	506	1,203	12,055	**13,764**
ESE	0	0	0	0	0	0	208	947	1,536	103,127	**105,818**
SE	0	0	0	0	0	0	219	374	16,764	11,931	**29,288**
SSE	0	0	0	0	0	20	212	346	7,427	8,500	**16,505**
Total	**0**	**0**	**0**	**0**	**0**	**108**	**1,995**	**5,074**	**32,197**	**143,392**	**182,766**

Key: INTEC, Idaho Nuclear Technology and Engineering Center.
Source: DOC 1992.

J.2.1.4 Source Term Data

Estimated incident-free radiological releases associated with the pit conversion and MOX facilities are presented in Tables J–22 and J–23. Stack heights and release locations are provided in the facility data reports (DOE 1999; UC 1998c, 1998d).

Table J–22. Estimated Incident-Free Annual Radiological Releases From the Pit Conversion Facility at INEEL

Isotope	(μCi/yr)
Plutonium 236	9.3×10^{-11}
Plutonium 238	0.065
Plutonium 239	0.69
Plutonium 240	0.18
Plutonium 241	0.69
Plutonium 242	4.8×10^{-5}
Americium 241	0.37
Hydrogen 3	1.1×10^{9}

Source: UC 1998c.

Table J–23. Estimated Incident-Free Annual Radiological Releases From the MOX Facility at INEEL

Isotope	(μCi/yr)
Plutonium 236	1.3×10^{-8}
Plutonium 238	8.5
Plutonium 239	91
Plutonium 240	23
Plutonium 241	101
Plutonium 242	6.1×10^{-3}
Americium 241	48
Uranium 234	5.1×10^{-3}
Uranium 235	2.1×10^{-4}
Uranium 238	0.012

Source: UC 1998d.

J.2.1.5 Other Calculational Assumptions

To estimate radiological impacts of incident-free operation of the proposed facilities at INEEL, the following additional assumptions and factors were considered, in accordance with the guidelines established in NRC Regulatory Guide 1.109 (NRC 1977).

Ground surfaces were assumed to have no previous deposition of radionuclides for the purposes of modeling the incremental radiological impacts associated with surplus plutonium disposition activities. However, doses associated with true instances of prior deposition are accounted for in the Affected Environment and Cumulative Impacts sections.

The annual external exposure time to the plume and to soil contamination was 0.7 year for the MEI (NRC 1977).

The annual external exposure time to the plume and to soil contamination was 0.5 year for the population (NRC 1977).

The annual inhalation exposure time to the plume was 1 year for the MEI and general population (NRC 1977).

The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of the adult human.

A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of contaminated animal products. Drinking water, aquatic food ingestion, and any other pathway that may involve liquid exposure were not examined because all releases are to the air.

Reported stack heights were used for atmospheric releases. The resultant doses were conservative as use of the actual stack height instead of the effective stack height negates plume rise.

The calculated doses are 50-year committed doses from 1 year of intake.

J.2.2 Facilities

The following sections present all viable radiological impact scenarios that could be associated with different combinations of incident-free facility operations at INEEL.

J.2.2.1 Pit Conversion Facility

J.2.2.1.1 Construction of Pit Conversion Facility

No radiological risk would be incurred by members of the public from construction and modification of a pit conversion facility in the Fuel Processing Facility (FPF) at INEEL. According to a recent radiation survey (Mitchell et al. 1997) conducted in the INTEC area, a construction worker could receive about 5 mrem/yr above natural background levels from exposure to radiation deriving from other activities, past or present, at the site. Construction worker exposures would be kept as low as is reasonably achievable, and workers would be monitored (badged) as appropriate.

J.2.2.1.2 Operation of Pit Conversion Facility

Tables J–24 and J–25 present the incident-free radiological impacts of the operation of a pit conversion facility at INEEL.

Table J–24. Potential Radiological Impacts on the Public of Operation of Pit Conversion Facility in FPF at INEEL

Population within 80 km for year 2010	
Dose (person-rem)	2.2
Percent of natural background[a]	3.3×10^{-3}
10-year latent fatal cancers	0.011
Maximally exposed individual	
Annual dose (mrem)	0.015
Percent of natural background[a]	4.2×10^{-3}
10-year latent fatal cancer risk	7.5×10^{-8}
Average exposed individual within 80 km[b]	
Annual dose (mrem)	0.012
10-year latent fatal cancer risk	6.0×10^{-8}

[a] The annual natural background radiation level at INEEL is 361 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 66,000 person-rem.
[b] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of INEEL in 2010 (182,800).
Key: FPF, Fuel Processing Facility.
Source: Model results.

Table J–25. Potential Radiological Impacts on Involved Workers of Operation of Pit Conversion Facility in FPF at INEEL

Number of badged workers	341
Total dose (person-rem/yr)	170
10-year latent fatal cancers	0.68
Average worker dose (mrem/yr)	500
10-year latent fatal cancer risk	2.0×10^{-3}

Key: FPF, Fuel Processing Facility.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: UC 1998c.

J.2.2.2 MOX Facility

J.2.2.2.1 Construction of MOX Facility

No radiological risk would be incurred by members of the public from the construction of a new MOX facility at INEEL. According to a recent radiation survey (Mitchell et al. 1997) conducted in the INTEC area, a construction worker could receive about 5 mrem/yr above natural background levels from exposure to radiation deriving from other activities, past or present, at the site. Construction worker exposures would be kept as low as is reasonably achievable, and workers would be monitored (badged) as appropriate.

J.2.2.2.2 Operation of MOX Facility

Tables J–26 and J–27 present the incident-free radiological impacts of the operation of a new MOX facility at INEEL.

Table J–26. Potential Radiological Impacts on the Public of Operation of New MOX Facility at INEEL[a]

Population within 80 km for year 2010	
Dose (person-rem)	0.037
Percent of natural background[b]	5.6×10^{-5}
10-year latent fatal cancers	1.9×10^{-4}
Maximally exposed individual	
Annual dose (mrem)	3.2×10^{-3}
Percent of natural background[b]	8.8×10^{-4}
10-year latent fatal cancer risk	1.6×10^{-8}
Average exposed individual within 80 km[c]	
Annual dose (mrem)	2.1×10^{-4}
10-year latent fatal cancer risk	1.0×10^{-9}

[a] As described in Section 4.26.2.2.2, Water Resources, no component was attributed to liquid pathways because it is not expected that significant contamination could reach these pathways given the site's groundwater and surface-water characteristics.

[b] The annual natural background radiation level at INEEL is 361 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 66,000 person-rem.

[c] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of INEEL in 2010 (182,800).

Source: Model results.

Table J–27. Potential Radiological Impacts on Involved Workers of Operation of New MOX Facility at INEEL

Number of badged workers	331
Total dose (person-rem/yr)	22
10-year latent fatal cancers	0.088
Average worker dose (mrem/yr)	65
10-year latent fatal cancer risk	2.6×10^{-4}

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: DOE 1999; UC 1998d.

J.2.2.3 Pit Conversion and MOX Facilities

J.2.2.3.1 Construction of Pit Conversion and MOX Facilities

No radiological risk would be incurred by members of the public from the construction and modification of a pit conversion facility in FPF and construction of a new MOX facility at INEEL. According to a recent radiation survey (Mitchell et al. 1997) conducted in the INTEC area, a construction worker could receive about 5 mrem/yr above natural background levels from exposure to radiation deriving from other activities, past or present, at the site. Construction worker exposures would be kept as low as is reasonably achievable, and workers would be monitored (badged) as appropriate.

J.2.2.3.2 Operation of Pit Conversion and MOX Facilities

Tables J–28 and J–29 present the incident-free radiological impacts of operation of pit conversion and MOX facilities at INEEL.

Table J–28. Potential Radiological Impacts on the Public of Operation of Pit Conversion Facility in FPF and New MOX Facility at INEEL

Impact	Pit Conversion	MOX[a]	Total[b]
Population within 80 km for year 2010			
Dose (person-rem)	2.2	0.037	2.2
Percent of natural background[c]	3.3×10^{-3}	5.6×10^{-5}	3.4×10^{-3}
10-year latent fatal cancers	0.011	1.9×10^{-4}	0.011
Maximally exposed individual			
Annual dose (mrem)	0.015	3.2×10^{-3}	0.018
Percent of natural background[c]	4.2×10^{-3}	8.8×10^{-4}	5.1×10^{-3}
10-year latent fatal cancer risk	7.5×10^{-8}	1.6×10^{-8}	9.1×10^{-8}
Average exposed individual within 80 km[d]			
Annual dose (mrem)	0.012	2.1×10^{-4}	0.012
10-year latent fatal cancer risk	6.0×10^{-8}	1.0×10^{-9}	6.1×10^{-8}

[a] As described in Section 4.26.2.2.2, Water Resources, no component was attributed to liquid pathways because it is not expected that significant contamination could reach these pathways given the site's groundwater and surface-water characteristics.
[b] Totals are additive in all cases because the same groups or individuals would receive doses from both facilities.
[c] The annual natural background radiation level at INEEL is 361 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 66,000 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of INEEL in 2010 (182,800).
Key: FPF, Fuel Processing Facility.
Source: Model results.

Table J–29. Potential Radiological Impacts on Involved Workers of Operation of Pit Conversion Facility in FPF and New MOX Facility at INEEL

Impact	Pit Conversion	MOX	Total
Number of badged workers	341	331	672
Total dose (person-rem/yr)	170	22	192
10-year latent fatal cancers	0.68	0.088	0.77
Average worker dose (mrem/yr)	500	65	286[a]
10-year latent fatal cancer risk	2.0×10^{-3}	2.6×10^{-4}	1.1×10^{-3}

[a] Represents an average of the doses for both facilities.
Key: FPF, Fuel Processing Facility.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: DOE 1999; UC 1998c, 1998d.

J.3 PANTEX

J.3.1 Assessment Data

To perform the dose assessments for the SPD EIS, different types of data were collected and generated. In addition, calculational assumptions were made. Appendix F.10 provides a summary of the methods and tools (e.g., the GENII computer code) that were used for the assessments.

J.3.1.1 Meteorological Data

The meteorological data used for the Pantex dose assessments was in the form of a JFD file. A JFD file is a table listing the percentages of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The JFD file was based on measurements taken over a period of several years at a specific location

and height. Average annual meteorological conditions, averaged over the measurement period, were used for normal operations. Table J–30 presents the JFD used in the dose assessments for Pantex.

J.3.1.2 Population Data

The Pantex population distribution was based on the *1990 Census of Population and Housing Data* (DOC 1992). Projections were determined for the year 2010 (about midlife of operations) for areas within 80 km (50 mi) of the locations for the proposed plutonium disposition facilities. The site population in 2010 was assumed to be representative of the population over the operational period evaluated. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances out to an 80-km (50-mi) distance. The grid was centered at Zone 4, the location from which radionuclides are assumed to be released during incident-free operations. Table J–31 presents the population data used for the dose assessments at Pantex.

J.3.1.3 Agricultural Data

The 1987 Census of Agriculture was the source used to generate site-specific data for food production. Food production was spatially distributed on a circular grid similar to that used for the population distribution described previously. This food grid (or wheel) was generated by combining the fraction of a county in each segment (e.g., south, southwest, north-northeast) and the county production of the eight food categories analyzed by GENII—leafy vegetables, root vegetables, fruits, grains, beef, poultry, milk, and eggs. Each county's food production was assumed to be distributed uniformly over the given county's land area. These categorized food wheels were then used in the assessment of doses to the Pantex population from the ingestion pathway. The consumption rates used in the dose assessments were those for the MEI and average exposed individual. People living within the 80-km (50-mi) assessment area were assumed to consume only food grown in that area. Pantex food production and consumption data used for the dose assessments in the SPD EIS were obtained from the *Health Risk Data for Storage and Disposition Final PEIS* (HNUS 1996).

Table J–30. 1985–1989 Joint Frequency Distributions at 7-m Height for Pantex[a]

Wind Speed (m/s)	Stability Class	Wind Blows Toward															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
0.89	A	0.02	0	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.02	0.01	0.01
	B	0.02	0.01	0.01	0.02	0.03	0.02	0.02	0.02	0.05	0.01	0.03	0.02	0.04	0.02	0.03	0.02
	C	0.02	0	0.01	0.01	0.01	0	0.01	0	0.01	0.01	0.01	0.01	0.02	0.01	0.01	0.01
	D	0.03	0.01	0.03	0.02	0.02	0.02	0.02	0.03	0.02	0.02	0.01	0.02	0.03	0.02	0.02	0.03
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0.12	0.04	0.04	0.05	0.04	0.04	0.07	0.08	0.17	0.11	0.16	0.09	0.13	0.13	0.11	0.08
2.5	A	0.03	0.01	0.02	0.02	0.03	0.02	0.02	0.01	0.02	0.02	0.01	0.03	0.02	0.02	0.02	0.01
	B	0.12	0.06	0.08	0.06	0.14	0.06	0.07	0.05	0.13	0.06	0.09	0.05	0.11	0.09	0.11	0.07
	C	0.12	0.05	0.07	0.07	0.06	0.05	0.04	0.05	0.12	0.11	0.09	0.11	0.13	0.13	0.15	0.09
	D	0.22	0.12	0.13	0.14	0.18	0.12	0.12	0.16	0.19	0.16	0.12	0.14	0.18	0.13	0.16	0.16
	E	0.23	0.1	0.09	0.1	0.12	0.14	0.16	0.14	0.31	0.21	0.23	0.18	0.21	0.15	0.19	0.12
	F	0.41	0.16	0.13	0.14	0.18	0.2	0.25	0.23	0.62	0.49	0.64	0.39	0.48	0.49	0.43	0.28
4.5	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0.08	0.04	0.07	0.07	0.07	0.06	0.06	0.09	0.17	0.13	0.13	0.09	0.1	0.08	0.07	0.08
	C	0.45	0.21	0.18	0.2	0.27	0.16	0.22	0.22	0.63	0.45	0.54	0.39	0.47	0.37	0.48	0.32
	D	1.14	0.72	0.64	0.59	0.72	0.66	1.02	1.1	2.19	1.21	1	0.5	0.41	0.32	0.6	0.5
	E	0.72	0.33	0.28	0.27	0.41	0.39	0.79	1.16	2.75	1.85	1.83	0.93	0.55	0.56	0.79	0.38
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6.9	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0.13	0.1	0.07	0.05	0.04	0.04	0.05	0.13	0.52	0.5	0.39	0.22	0.16	0.08	0.05	0.04
	D	3.07	1.76	1	0.67	0.9	0.83	1.73	2.59	7.3	4.2	3.32	1.83	1.19	0.57	0.89	0.95
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
9.6	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0.03	0.02	0.03	0.01	0	0.01	0.01	0.03	0.18	0.19	0.09	0.04	0.03	0.01	0	0.01
	D	1.49	0.82	0.29	0.13	0.11	0.13	0.33	0.48	2.24	1.48	1.01	0.76	0.49	0.12	0.15	0.34
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
12.1	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0.01	0.01	0	0	0	0	0	0	0.04	0.01	0.01	0.02	0.01	0	0	0
	D	0.73	0.32	0.05	0.03	0.01	0.02	0.05	0.1	0.41	0.22	0.2	0.25	0.24	0.05	0.09	0.2
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

[a] Joint frequency distribution data was compiled by the National Weather Service Station at Amarillo Airport; it was assumed that this data satisfactorily represented the atmospheric conditions at the Pantex site.
Source: NWS 1997.

Table J–31. Projected Pantex Population Surrounding Zone 4 for Year 2010

Direction	Distance (mi)										Total
	0–1	1–2	2–3	3–4	4–5	5–10	10–20	20–30	30–40	40–50	
S	0	0	0	4	5	41	100	96	104	268	**618**
SSW	0	0	0	0	5	117	441	1,095	361	1,013	**3,032**
SW	0	0	0	3	3	901	18,330	14,816	13,199	1,137	**48,389**
WSW	0	0	3	2	3	49	88,209	65,959	1,189	528	**15,5942**
W	0	0	2	2	3	25	3,372	683	227	897	**5,211**
WNW	0	0	3	2	3	25	148	360	517	834	**1,892**
NW	0	2	3	3	3	25	98	253	547	542	**1,476**
NNW	0	2	3	4	5	30	88	344	519	16,924	**17,919**
N	0	2	3	4	5	41	151	5,476	176	225	**6,083**
NNE	0	2	3	4	5	41	162	18,764	2,998	233	**22,212**
NE	0	2	3	4	5	41	163	396	295	165	**1,074**
ENE	0	2	3	4	5	41	324	724	22,852	176	**24,131**
E	0	2	3	4	5	961	2,016	884	372	1,085	**5,332**
ESE	0	2	3	4	5	41	273	512	248	401	**1,489**
SE	0	0	3	4	5	41	303	370	115	2,182	**3,023**
SSE	0	0	0	4	5	41	677	311	69	109	**1,216**
Total	**0**	**16**	**35**	**52**	**70**	**2,461**	**114,855**	**111,043**	**43,788**	**26,719**	**299,039**

Source: DOC 1992.

J.3.1.4 Source Term Data

Estimated incident-free radiological releases associated with the new pit conversion and MOX facilities at Pantex are presented in Tables J–32 and J–33. Stack heights and release locations are provided in the facility data reports (DOE 1999; UC 1998e, 1998f).

Table J–32. Estimated Incident-Free Annual Radiological Releases From the New Pit Conversion Facility at Pantex

Isotope	(μCi/yr)
Plutonium 236	9.3×10^{-11}
Plutonium 238	0.065
Plutonium 239	0.69
Plutonium 240	0.18
Plutonium 241	0.69
Plutonium 242	4.8×10^{-5}
Americium 241	0.37
Hydrogen 3	1.1×10^{9}

Source: UC 1998e.

Table J–33. Estimated Incident-Free Annual Radiological Releases From the New MOX Facility at Pantex

Isotope	(μCi/yr)
Plutonium 236	1.3×10^{-8}
Plutonium 238	8.5
Plutonium 239	91
Plutonium 240	23
Plutonium 241	101
Plutonium 242	6.1×10^{-3}
Americium 241	48
Uranium 234	5.1×10^{-3}
Uranium 235	2.1×10^{-4}
Uranium 238	0.012

Source: UC 1998f.

J.3.1.5 Other Calculational Assumptions

To estimate radiological impacts of incident-free operation of the proposed facilities at Pantex, the following additional assumptions and factors were considered, in accordance with the guidelines established in NRC Regulatory Guide 1.109 (NRC 1977).

Ground surfaces were assumed to have no previous deposition of radionuclides for the purposes of modeling the incremental radiological impacts associated with surplus plutonium disposition activities. However, doses associated with true instances of prior deposition are accounted for in the Affected Environment and Cumulative Impacts sections.

The annual external exposure time to the plume and to soil contamination was 0.7 year for the MEI (NRC 1977).

The annual external exposure time to the plume and to soil contamination was 0.5 year for the population (NRC 1977).

The annual inhalation exposure time to the plume was 1 year for the MEI and general population (NRC 1977).

The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of the adult human.

A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of contaminated animal products. Drinking water, aquatic food ingestion, and any other pathway that may involve liquid exposure were not examined because all releases were to the air.

Reported stack heights were used for atmospheric releases. The resultant doses were conservative as use of the actual stack height instead of the effective sack height negates plume rise.

The calculated doses are 50-year committed doses from 1 year of intake.

J.3.2 Facilities

The following sections present all viable radiological impact scenarios that could be associated with different combinations of incident-free facility operations at Pantex.

J.3.2.1 Pit Conversion Facility

J.3.2.1.1 Construction of Pit Conversion Facility

No radiological risk would be incurred by members of the public from the construction of a new pit conversion facility at Pantex. According to a recent radiation survey (DOE 1997) conducted in Zone 4, a construction worker would not be expected to receive any additional radiation exposure above natural background levels in the area. Nonetheless, construction workers may be monitored (badged) as a precautionary measure.

J.3.2.1.2 Operation of Pit Conversion Facility

Tables J–34 and J–35 present the incident-free radiological impacts of the operation of a new pit conversion facility at Pantex.

Table J–34. Potential Radiological Impacts on the Public of Operation of New Pit Conversion Facility at Pantex

Population within 80 km for year 2010	
Dose (person-rem)	0.58
Percent of natural background[a]	5.8×10^{-4}
10-year latent fatal cancers	2.9×10^{-3}
Maximally exposed individual	
Annual dose (mrem)	0.062
Percent of natural background[a]	0.019
10-year latent fatal cancer risk	3.1×10^{-7}
Average exposed individual within 80 km[b]	
Annual dose (mrem)	1.9×10^{-3}
10-year latent fatal cancer risk	9.5×10^{-9}

[a] The annual natural background radiation level at Pantex is 332 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 99,300 person-rem.

[b] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of Pantex in 2010 (299,000).

Source: Model results.

Table J–35. Potential Radiological Impacts on Involved Workers of Operation of New Pit Conversion Facility at Pantex

Number of badged workers	383
Total dose (person-rem/yr)	192
10-year latent fatal cancers	0.77
Average worker dose (mrem/yr)	500
10-year latent fatal cancer risk	2.0×10^{-3}

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: UC 1998e.

J.3.2.2 MOX Facility

J.3.2.2.1 Construction of MOX Facility

No radiological risk would be incurred by members of the public from construction of a new MOX facility at Pantex. According to a recent radiation survey (DOE 1997) conducted in Zone 4, a construction worker would not be expected to receive any additional radiation exposure above natural background levels in the area. Nonetheless, construction workers may be monitored (badged) as a precautionary measure.

J.3.2.2.2 Operation of MOX Facility

Tables J–36 and J–37 present the incident-free radiological impacts of the operation of a new MOX facility at Pantex.

Table J–36. Potential Radiological Impacts on the Public of Operation of New MOX Facility at Pantex[a]

Population within 80 km for year 2010	
Dose (person-rem)	0.027
Percent of natural background[b]	2.7×10^{-5}
10-year latent fatal cancers	1.3×10^{-4}
Maximally exposed individual	
Annual dose (mrem)	0.015
Percent of natural background[b]	$4.5\text{x}10^{-3}$
10-year latent fatal cancer risk	7.5×10^{-8}
Average exposed individual within 80 km[c]	
Annual dose (mrem)	8.8×10^{-5}
10-year latent fatal cancer risk	4.5×10^{-10}

[a] As described in Section 4.26.3.2.2, Water Resources, no component was attributed to liquid pathways because it is not expected that significant contamination could reach these pathways given the site's groundwater and surface-water characteristics.
[b] The annual natural background radiation level at Pantex is 332 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 99,300 person-rem.
[c] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of Pantex in 2010 (299,000).
Source: Model results.

Table J–37. Potential Radiological Impacts on Involved Workers of Operation of New MOX Facility at Pantex

Number of badged workers	331
Total dose (person-rem/yr)	22
10-year latent fatal cancers	0.088
Average worker dose (mrem/yr)	65
10-year latent fatal cancer risk	2.6×10^{-4}

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: DOE 1999; UC 1998f.

J.3.2.3 Pit Conversion and MOX Facilities

J.3.2.3.1 Construction of Pit Conversion and MOX Facilities

No radiological risk would be incurred by members of the public from the construction of new pit conversion and MOX facilities at Pantex. According to a recent radiation survey (DOE 1997) conducted in Zone 4, a construction worker would not be expected to receive any additional radiation exposure above natural background levels in the area. Nonetheless, construction workers may be monitored (badged) as a precautionary measure.

J.3.2.3.2 Operation of Pit Conversion and MOX Facilities

Tables J–38 and J–39 present the incident-free radiological impacts of operation of the new pit conversion and MOX facilities at Pantex.

Table J–38. Potential Radiological Impacts on the Public of Operation of New Pit Conversion and MOX Facilities at Pantex

Impact	Pit Conversion	MOX[a]	Total[b]
Population within 80 km for year 2010			
Dose (person-rem)	0.58	0.027	0.61
Percent of natural background[c]	5.8×10^{-4}	2.7×10^{-5}	6.1×10^{-4}
10-year latent fatal cancers	2.9×10^{-3}	1.3×10^{-4}	3.0×10^{-3}
Maximally exposed individual			
Annual dose (mrem)	0.062	0.015	0.077
Percent of natural background[c]	0.019	4.5×10^{-3}	0.024
10-year latent fatal cancer risk	3.1×10^{-7}	7.5×10^{-8}	3.9×10^{-7}
Average exposed individual within 80 km[d]			
Annual dose (mrem)	1.9×10^{-3}	8.8×10^{-5}	2.0×10^{-3}
10-year latent fatal cancer risk	9.5×10^{-9}	4.4×10^{-10}	9.9×10^{-9}

[a] As described in Section 4.26.3.2.2, Water Resources, no component was attributed to liquid pathways because it is not expected that significant contamination could reach these pathways given the site's groundwater and surface-water characteristics.

[b] Totals are additive in all cases because the same groups or individuals would receive doses from both facilities.

[c] The annual natural background radiation level at Pantex is 332 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive 99,300 person-rem.

[d] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of Pantex in 2010 (299,000).

Source: Model results.

Table J–39. Potential Radiological Impacts on Involved Workers of Operation of New Pit Conversion and MOX Facilities at Pantex

Impact	Pit Conversion	MOX	Total
Number of badged workers	383	331	714
Total dose (person-rem/yr)	192	22	214
10-year latent fatal cancers	0.77	0.088	0.86
Average worker dose (mrem/yr)	500	65	300[a]
10-year latent fatal cancer risk	2.0×10^{-3}	2.6×10^{-4}	1.2×10^{-3}

[a] Represents an average of the doses for both facilities.

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: DOE 1999; UC 1998e, 1998f.

J.4 SRS

J.4.1 Assessment Data

To perform the dose assessments for the SPD EIS, different types of data were collected and generated. In addition, calculational assumptions were made. Appendix F.10 provides a summary of the methods and tools (e.g., the GENII computer code) that were used for the assessments.

J.4.1.1 Meteorological Data

The meteorological data used for the SRS dose assessments was in the form of a JFD file. A JFD file is a table listing the percentages of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The JFD data file was based on measurements taken over a period of several years at a specific location (F-Area) and height. Average annual meteorological conditions, averaged over the measurement period, were used for normal operations. Table J–40 presents the JFD data used in the dose assessments for SRS.

J.4.1.2 Population Data

The SRS population distribution was based on the *1990 Census of Population and Housing Data* (DOC 1992). Projections were determined for the year 2010 (about midlife of operations) for areas within 80 km (50 mi) of the locations for the proposed surplus plutonium disposition facilities. The site population in 2010 was assumed to be representative of the population over the operational period evaluated. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances out to an 80-km (50-mi) distance. The grids were centered at the Actinide Packaging and Storage Facility in F-Area, the locations from which radionuclides are assumed to be released during incident-free operations. Tables J–41 and J–42 present the population data used for the dose assessments at SRS.

J.4.1.3 Agricultural Data

The 1987 Census of Agriculture was the source used to generate site-specific data for food production. Food production was spatially distributed on a circular grid similar to that used for the population distributions described previously. This food grid (or wheel) was generated by combining the fraction of a county in each segment (e.g., south, southwest, north-northeast) and the county production of the eight food categories analyzed by GENII (leafy vegetables, root vegetables, fruits, grains, beef, poultry, milk, and eggs). Each county's food production was assumed to be distributed uniformly over the given county's land area. These categorized food wheels are then used in the assessment of doses to the SRS population from the ingestion pathway. The consumption rates used in the dose assessments were those for the MEI and average exposed individual. People living within the 80-km (50-mi) assessment area were assumed to consume only food grown in that area. SRS food production and consumption data used for the dose assessments in the SPD EIS were obtained from the *Health Risk Data for Storage and Disposition Final PEIS* (HNUS 1996).

Table J–40. SRS 1987–1991 Joint Frequency Distributions at 61-m Height

Wind Speed (m/s)	Stability Class	Wind Blows Toward															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
2.0	A	0.27	0.35	0.39	0.42	0.34	0.31	0.28	0.31	0.31	0.3	0.32	0.34	0.5	0.32	0.29	0.26
	B	0.04	0.05	0.06	0.08	0.05	0.05	0.04	0.05	0.05	0.04	0.06	0.07	0.06	0.06	0.06	0.04
	C	0.02	0.03	0.1	0.07	0.02	0.04	0.03	0.06	0.05	0.05	0.07	0.07	0.09	0.06	0.03	0.02
	D	0.01	0.03	0.07	0.02	0.02	0.03	0.05	0.05	0.04	0.04	0.05	0.05	0.03	0.02	0.04	0.03
	E	0	0	0.02	0	0	0.01	0.02	0.01	0.01	0.02	0.02	0.02	0.02	0.01	0.01	0.02
	F	0	0	0.01	0	0	0	0	0	0	0	0	0	0	0	0	0
4.0	A	0.64	0.63	0.7	0.77	0.76	0.63	0.54	0.66	0.58	0.64	0.73	1.15	1	0.69	0.52	0.44
	B	0.22	0.3	0.33	0.4	0.33	0.26	0.21	0.22	0.28	0.26	0.51	0.67	0.59	0.3	0.16	0.2
	C	0.08	0.52	0.57	0.77	0.51	0.37	0.33	0.39	0.44	0.45	0.7	0.77	0.69	0.33	0.28	0.15
	D	0.06	0.52	1.49	1.12	0.5	0.51	0.62	0.78	0.77	0.62	0.7	0.75	0.77	0.47	0.31	0.15
	E	0.04	0.2	0.8	0.35	0.18	0.28	0.42	0.55	0.57	0.43	0.51	0.42	0.49	0.33	0.25	0.15
	F	0.02	0.02	0.1	0.05	0.03	0.03	0.07	0.09	0.06	0.07	0.09	0.06	0.06	0.07	0.06	0.04
6.0	A	0.49	0.15	0.1	0.09	0.1	0.09	0.08	0.14	0.11	0.14	0.17	0.17	0.19	0.18	0.1	0.21
	B	0.12	0.22	0.17	0.22	0.19	0.09	0.08	0.15	0.17	0.2	0.3	0.42	0.37	0.28	0.11	0.08
	C	0.08	0.4	0.42	0.63	0.35	0.18	0.19	0.34	0.38	0.43	0.6	0.77	0.64	0.39	0.17	0.11
	D	0.06	0.8	2.28	1.39	0.62	0.44	0.67	1.31	1.21	0.75	0.94	0.87	1.01	0.66	0.29	0.18
	E	0.06	0.51	1.36	1.07	0.56	0.48	0.64	1.25	1.29	0.97	1.08	1.14	1.22	0.77	0.38	0.21
	F	0.02	0.04	0.18	0.28	0.23	0.21	0.2	0.23	0.23	0.26	0.25	0.26	0.21	0.19	0.1	0.08
8.0	A	0.11	0.03	0.01	0.01	0.01	0.01	0	0.02	0.01	0.04	0.02	0.02	0.03	0.03	0.02	0.03
	B	0	0.06	0.02	0.01	0	0	0	0.01	0.03	0.04	0.08	0.06	0.04	0.08	0.03	0.01
	C	0.01	0.11	0.11	0.13	0.06	0.04	0.05	0.07	0.13	0.17	0.27	0.28	0.33	0.29	0.06	0.01
	D	0.04	0.3	0.6	0.41	0.08	0.03	0.1	0.25	0.21	0.15	0.2	0.24	0.63	0.35	0.05	0.02
	E	0.02	0.29	0.25	0.16	0.06	0.02	0.02	0.06	0.08	0.05	0.16	0.12	0.15	0.06	0.02	0.02
	F	0	0.01	0.04	0.06	0.04	0.01	0.02	0.02	0.04	0.05	0.02	0.01	0.01	0	0	0
12.0	A	0.01	0	0	0	0	0	0	0	0	0	0.01	0.01	0	0.01	0	0.01
	B	0	0	0	0	0	0	0	0	0	0	0	0	0.01	0.02	0	0
	C	0	0.01	0	0	0	0	0	0.02	0.03	0.03	0.04	0.06	0.2	0.18	0.01	0
	D	0.01	0.06	0.08	0.08	0.01	0.01	0.01	0.03	0.05	0.03	0.06	0.03	0.39	0.2	0.01	0
	E	0	0.01	0.02	0.01	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
14.1	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0.01	0	0.01	0	0
	D	0	0	0	0	0	0	0	0	0	0	0	0	0.01	0	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Source: Simpkins 1997.

Table J–41. Projected SRS Population Surrounding APSF (Pit Conversion and MOX Facilities) for Year 2010

Direction	Distance (mi) 0–1	1–2	2–3	3–4	4–5	5–10	10–20	20–30	30–40	40–50	Total
S	0	0	0	0	0	0	600	2,109	3,312	3,447	**9,468**
SSW	0	0	0	0	0	36	935	1,853	4,732	2,501	**10,057**
SW	0	0	0	0	0	73	1,239	8,333	2,023	4,318	**15,986**
WSW	0	0	0	0	0	228	3,762	4,014	3,742	7,194	**18,940**
W	0	0	0	0	0	355	7,786	47,484	21,880	18,192	**95,697**
WNW	0	0	0	0	0	2,439	11,335	205,958	53,232	6,694	**279,658**
NW	0	0	0	0	0	1,455	18,694	38,351	2,884	3,123	**64,507**
NNW	0	0	0	0	0	3,279	40,843	20,468	9,466	5,766	**79,822**
N	0	0	0	0	0	1,012	7,787	6,010	5,928	20,994	**41,731**
NNE	0	0	0	0	0	145	1,934	2,959	6,794	20,775	**32,607**
NE	0	0	0	0	0	0	3,168	3,786	5,985	11,236	**24,175**
ENE	0	0	0	0	0	0	3,077	5,828	7,625	33,477	**50,007**
E	0	0	0	0	0	0	6,188	5,442	7,342	3,952	**22,924**
ESE	0	0	0	0	0	0	996	3,497	4,455	7,253	**16,201**
SE	0	0	0	0	0	0	572	2,555	4,695	7,667	**15,489**
SSE	0	0	0	0	0	0	390	648	4,122	2,975	**8,135**
Total	**0**	**0**	**0**	**0**	**0**	**9,022**	**109,306**	**359,295**	**148,217**	**159,564**	**785,404**

Key: APSF, Actinide Packaging and Storage Facility.
Source: DOC 1992.

Table J–42. Projected SRS Population Surrounding APSF (Immobilization Facility) for Year 2010

Direction	Distance (mi) 0–1	1–2	2–3	3–4	4–5	5–10	10–20	20–30	30–40	40–50	Total
S	0	0	0	0	0	0	576	2,124	3,368	3,437	**9,505**
SSW	0	0	0	0	0	33	914	1,849	4,750	2,508	**10,054**
SW	0	0	0	0	0	59	1,204	8,412	2,043	4,640	**16,358**
WSW	0	0	0	0	0	241	3,930	4,188	3,771	6,887	**19,017**
W	0	0	0	0	0	543	7,632	51,313	22,422	18,246	**100,156**
WNW	0	0	0	0	0	2,344	11,777	204,567	51,659	6,581	**276,928**
NW	0	0	0	0	0	1,479	19,053	36,367	2,990	3,123	**63,012**
NNW	0	0	0	0	0	3,394	43,236	17,846	9,567	5,783	**79,826**
N	0	0	0	0	0	961	7,818	5,691	6,005	21,037	**41,512**
NNE	0	0	0	0	0	171	1,936	3,000	6,811	21,327	**33,245**
NE	0	0	0	0	0	0	3,137	3,756	6,043	11,279	**24,215**
ENE	0	0	0	0	0	0	3,202	5,735	7,434	34,686	**51,057**
E	0	0	0	0	0	0	6,264	5,509	7,575	3,991	**23,339**
ESE	0	0	0	0	0	0	1,023	2,892	4,016	7,077	**15,008**
SE	0	0	0	0	0	0	569	3,116	5,213	7,848	**16,746**
SSE	0	0	0	0	0	0	380	636	3,953	3,002	**7,971**
Total	**0**	**0**	**0**	**0**	**0**	**9,225**	**112,651**	**357,001**	**147,620**	**161,452**	**787,949**

Key: APSF, Actinide Packaging and Storage Facility.
Source: DOC 1992.

J.4.1.4 Source Term Data

Estimated incident-free radiological releases associated with the new pit conversion, immobilization, and MOX facilities are presented in Tables J–43 through J–45. Stack heights and release locations are provided in the facility data reports (DOE 1999; UC 1998g, 1998h, 1999c, 1999d).

Table J–43. Estimated Incident-Free Annual Radiological Releases From the Pit Conversion Facility at SRS

Isotope	(μCi/yr)
Plutonium 236	9.3×10^{-11}
Plutonium 238	0.065
Plutonium 239	0.69
Plutonium 240	0.18
Plutonium 241	0.69
Plutonium 242	4.8×10^{-5}
Americium 241	0.37
Hydrogen 3	1.1×10^{9}

Source: UC 1998g.

Table J–44. Estimated Incident-Free Annual Radiological Releases From the New Immobilization Facility at SRS

Isotope	Ceramic (17 t) (μCi/yr)	Ceramic (50 t) (μCi/yr)	Glass (17 t) (μCi/yr)	Glass (50 t) (μCi/yr)
Plutonium 236	–	–	–	–
Plutonium 238	–	0.57	–	0.52
Plutonium 239	3.7	9.5	3.4	8.6
Plutonium 240	1.7	3.1	1.6	2.8
Plutonium 241	110	100	98	93
Plutonium 242	1.3×10^{-3}	1.6×10^{-3}	1.2×10^{-3}	1.5×10^{-3}
Americium 241	2.3	5.4	2.2	5.0
Uranium 234	–	–	–	–
Uranium 235	1.1×10^{-5}	4.5×10^{-5}	2.3×10^{-6}	2.3×10^{-6}
Uranium 238	8.8×10^{-5}	3.5×10^{-4}	1.9×10^{-5}	1.9×10^{-5}

Source: UC 1999c, 1999d.

Table J–45. Estimated Incident-Free Annual Radiological Releases From the New MOX Facility at SRS

Isotope	Airborne (μCi/yr)	Liquid (μCi/yr)
Plutonium 236	$1.3.\times10^{-8}$	9.3×10^{-8}
Plutonium 238	8.5	64
Plutonium 239	91	670
Plutonium 240	23	170
Plutonium 241	101	750
Plutonium 242	6.1×10^{-3}	0.046
Americium 241	48	350
Uranium 234	5.1×10^{-3}	0.037
Uranium 235	2.1×10^{-4}	1.6×10^{-3}
Uranium 238	0.012	0.089

Source: UC 1998h.

J.4.1.5 Other Calculational Assumptions

To estimate radiological impacts of incident-free operation of the facilities at SRS, the following additional assumptions and factors were considered, in accordance with the guidelines established in NRC Regulatory Guide 1.109 (NRC 1977).

Ground surfaces were assumed to have no previous deposition of radionuclides for the purposes of modeling the incremental radiological impacts associated with surplus plutonium disposition activities. However, doses associated with true instances of prior deposition are accounted for in the Affected Environment and Cumulative Impacts sections.

The annual external exposure time to the plume and to soil contamination was 0.7 year for the MEI (NRC 1977).

The annual external exposure time to the plume and to soil contamination was 0.5 year for the population (NRC 1977).

The annual inhalation exposure time to the plume was 1 year for the MEI and general population (NRC 1977).

The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of the adult human.

A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of contaminated animal products. Drinking water, aquatic food ingestion, and any other pathway that may involve liquid exposure were also examined for the MOX facility because it is the only facility with expected liquid releases at SRS.

Reported stack heights were used for atmospheric releases. The resultant doses were conservative as use of the actual stack height instead of the effective stack height negates plume rise.

The calculated doses are 50-year committed doses from 1 year of intake.

J.4.2 Facilities

The following sections present all viable radiological impact scenarios that could be associated with different combinations of incident-free facility operations at SRS.

J.4.2.1 Pit Conversion Facility

J.4.2.1.1 Construction of Pit Conversion Facility

No radiological risk would be incurred by members of the public from the construction of a new pit conversion facility at SRS. Construction worker exposures to radiation that derives from other activities at the site, past and present, would also be kept as low as is reasonably achievable. Construction workers would be monitored (badged) as appropriate. Summaries of radiological impacts of these activities are presented in Table J–46 for workers at risk.

Table J–46. Potential Radiological Impacts on Construction Workers of New Pit Conversion Facility at SRS

Annual average number of workers	341
Total dose (person-rem/yr)	1.4
Annual latent fatal cancers[a]	5.6×10^{-4}
Average worker dose (mrem/yr)	4
Annual latent fatal cancer risk	1.6×10^{-6}

[a] Values are based on a risk factor of 400 latent fatal cancers per million person-rem set by the National Research Council's Committee on the Biological Effects of Ionizing Radiations.

Note: The radiological limit for a construction worker is 100 mrem/yr because they are categorized as members of the public (DOE 1993). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: ICRP 1991; NAS 1990; UC 1998g.

J.4.2.1.2 Operation of Pit Conversion Facility

Tables J–47 and J–48 present the incident-free radiological impacts of the operation of a new pit conversion facility at SRS.

Table J–47. Potential Radiological Impacts on the Public of Operation of New Pit Conversion Facility at SRS

Population within 80 km for year 2010	
Dose (person-rem)	1.6
Percent of natural background[a]	6.9×10^{-4}
10-year latent fatal cancers	8.0×10^{-3}
Maximally exposed individual	
Annual dose (mrem)	3.7×10^{-3}
Percent of natural background[a]	1.3×10^{-3}
10-year latent fatal cancer risk	1.9×10^{-8}
Average exposed individual within 80 km[b]	
Annual dose (mrem)	2.0×10^{-3}
10-year latent fatal cancer risk	1.0×10^{-8}

[a] The annual natural background radiation level at SRS is 295 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive about 232,000 person-rem.
[b] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of SRS in 2010 (about 790,000).
Source: Model results.

Table J–48. Potential Radiological Impacts on Involved Workers of Operation of New Pit Conversion Facility at SRS

Number of badged workers	383
Total dose (person-rem/yr)	192
10-year latent fatal cancers	0.77
Average worker dose (mrem/yr)	500
10-year latent fatal cancer risk	2.0×10^{-3}

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: UC 1998g.

J.4.2.2 Immobilization Facility

J.4.2.2.1 Construction of Immobilization Facility

No radiological risk would be incurred by members of the public from the construction of a new immobilization facility at SRS. Construction worker exposures to radiation that derives from other activities at the site, past or present, would also be kept as low as is reasonably achievable. Construction workers would be monitored (badged) as appropriate. Summaries of radiological impacts of these activities are presented in Table J–49 for workers at risk.

Table J–49. Potential Radiological Impacts on Construction Workers of New Immobilization Facility at SRS[a]

Annual average number of workers	374
Total dose (person-rem/yr)	1.5
Annual latent fatal cancers[b]	6.0×10^{-4}
Average worker dose (mrem/yr)	4
Annual latent fatal cancer risk	1.6×10^{-6}

[a] The values would be the same for immobilization in either ceramic or glass.
[b] Values are based on a risk factor of 400 latent fatal cancers per million person-rem set by the National Research Council's Committee on the Biological Effects of Ionizing Radiations.
Note: The radiological limit for a construction worker is 100 mrem/yr because they are categorized as members of the public (DOE 1993). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: ICRP 1991; NAS 1990; UC 1999c, 1999d.

J.4.2.2.2 Operation of Immobilization Facility

Tables J–50 and J–51 present all possible incident-free radiological impact scenarios of the operation of a new immobilization facility at SRS.

Table J–50. Potential Radiological Impacts on the Public of Operation of New Immobilization Facility at SRS

	17 t		50 t	
Impact	**Ceramic**	**Glass**	**Ceramic**	**Glass**
Population within 80 km for year 2010				
Dose (person-rem)	2.8×10^{-3}	2.6×10^{-3}	5.8×10^{-3}	5.3×10^{-3}
Percent of natural background[a]	1.2×10^{-6}	1.1×10^{-6}	2.5×10^{-6}	2.3×10^{-6}
10-year latent fatal cancers	1.4×10^{-5}	1.3×10^{-5}	2.9×10^{-5}	2.7×10^{-5}
Maximally exposed individual				
Annual dose (mrem)	2.8×10^{-5}	2.6×10^{-5}	5.8×10^{-5}	5.3×10^{-5}
Percent of natural background[a]	9.5×10^{-6}	8.8×10^{-6}	2.0×10^{-5}	1.8×10^{-5}
10-year latent fatal cancer risk	1.4×10^{-10}	1.3×10^{-10}	2.9×10^{-10}	2.7×10^{-10}
Average exposed individual within 80 km[b]				
Annual dose (mrem)	3.6×10^{-6}	3.3×10^{-6}	7.4×10^{-6}	6.7×10^{-6}
10-year latent fatal cancer risk	1.8×10^{-11}	1.6×10^{-11}	3.7×10^{-11}	3.4×10^{-11}

[Text deleted.]
[a] The annual natural background radiation level at SRS is 295 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive about 232,000 person-rem.
[b] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of the SRS facilities in 2010 (about 790,000).
Source: Model results.

Table J–51. Potential Radiological Impacts on Involved Workers of Operation of New Immobilization Facility at SRS[a]

Impact	17 t	50 t
Number of badged workers	323	339
Total dose (person-rem/yr)	242	254
10-year latent fatal cancers	0.97	1.0
Average worker dose (mrem/yr)	750	750
10-year latent fatal cancer risk	3.0×10^{-3}	3.0×10^{-3}

[a] The values would be the same for immobilization in either ceramic or glass.

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: UC 1999c, 1999d.

J.4.2.3 MOX Facility

J.4.2.3.1 Construction of MOX Facility

No radiological risk would be incurred by members of the public from the construction of a new MOX facility at SRS. Construction worker exposures to radiation that derives from other activities at the site, past or present, would also be kept as low as is reasonably achievable. Construction workers would be monitored (badged) as appropriate. Summaries of radiological impacts of these activities are presented in Table J–52 for workers at risk.

Table J–52. Potential Radiological Impacts on Construction Workers of New MOX Facility at SRS

Annual average number of workers	292
Total dose (person-rem/yr)	1.2
Annual latent fatal cancers[a]	4.8×10^{-4}
Average worker dose (mrem/yr)	4
Annual latent fatal cancer risk	1.6×10^{-6}

[a] Values are based on a risk factor of 400 latent fatal cancers per million person-rem set by the National Research Council's Committee on the Biological Effects of Ionizing Radiations.

Note: The radiological limit for a construction worker is 100 mrem/yr because they are categorized as members of the public (DOE 1993). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: ICRP 1991; NAS 1990; UC 1998h.

J.4.2.3.2 Operation of MOX Facility

Tables J–53 and J–54 present the incident-free radiological impacts of the operation of a new MOX facility at SRS.

Table J–53. Potential Radiological Impacts on the Public of Operation of New MOX Facility at SRS[a]

Population within 80 km for year 2010	
Dose (person-rem)	0.18
Percent of natural background[b]	7.8×10^{-5}
10-year latent fatal cancers	9.1×10^{-4}
Maximally exposed individual	
Annual dose (mrem)	3.7×10^{-3}
Percent of natural background[b]	1.3×10^{-3}
10-year latent fatal cancer risk	1.9×10^{-8}
Average exposed individual within 80 km[c]	
Annual dose (mrem)	2.3×10^{-4}
10-year latent fatal cancer risk	1.2×10^{-9}

[a] Includes a dose component from liquid pathways because it is possible that liquid releases could reach these pathways at SRS.
[b] The annual natural background radiation level at SRS is 295 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive about 232,000 person-rem.
[c] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of SRS in 2010 (about 790,000).
Source: Model results.

Table J–54. Potential Radiological Impacts on Involved Workers of Operation of New MOX Facility at SRS

Number of badged workers	331
Total dose (person-rem/yr)	22
10-year latent fatal cancers	0.088
Average worker dose (mrem/yr)	65
10-year latent fatal cancer risk	2.6×10^{-4}

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: DOE 1999; UC 1998h.

J.4.2.4 Pit Conversion and Immobilization Facilities

J.4.2.4.1 Construction of Pit Conversion and Immobilization Facilities

No radiological risk would be incurred by members of the public from construction of new pit conversion and immobilization facilities at SRS. Construction worker exposures to radiation that derives from other activities at the site, past or present, would also be kept as low as is reasonably achievable. Construction workers would be monitored (badged) as appropriate. Summaries of radiological impacts of these activities are presented in Table J–55 for workers at risk.

Table J–55. Potential Radiological Impacts on Construction Workers of New Pit Conversion and Immobilization Facilities at SRS

Impact	Pit Conversion	Immobilization[a]	Total
Annual average number of workers	316	374	690
Total dose (person-rem/yr)	1.3	1.5	2.8
Annual latent fatal cancers[b]	5.2×10^{-4}	6.0×10^{-4}	1.1×10^{-3}
Average worker dose (mrem/yr)	4	4	4[c]
Annual latent fatal cancer risk	1.6×10^{-6}	1.6×10^{-6}	1.6×10^{-6}

[a] The values would be the same for immobilization in either ceramic or glass.
[b] Values are based on a risk factor of 400 latent fatal cancers per million person-rem set by the National Research Council's Committee on the Biological Effects of Ionizing Radiations.
[c] Represents an average of the doses for both facilities.
Note: The radiological limit for a construction worker is 100 mrem/yr because they are categorized as members of the public (DOE 1993). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: ICRP 1991; NAS 1990; UC 1998g, 1999c, 1999d.

J.4.2.4.2 Operation of Pit Conversion and Immobilization Facilities

Tables J–56 and J–57 present all possible incident-free radiological impact scenarios of operation of the new pit conversion and immobilization facilities at SRS.

Table J–56. Potential Radiological Impacts on the Public of Operation of New Pit Conversion and Immobilization Facilities at SRS

Impact	Pit Conversion	Immobilization (50 t)		Total[a]
		Ceramic	Glass	
Population within 80 km for year 2010				
Dose (person-rem)	1.6	5.8×10^{-3}	5.3×10^{-3}	1.6
Percent of natural background[b]	6.9×10^{-4}	2.5×10^{-6}	2.3×10^{-6}	6.9×10^{-4}
10-year latent fatal cancers	8.0×10^{-3}	2.9×10^{-5}	2.7×10^{-5}	8.0×10^{-3}
Maximally exposed individual				
Annual dose (mrem)	3.7×10^{-3}	5.8×10^{-5}	5.3×10^{-5}	3.8×10^{-3}
Percent of natural background[b]	1.3×10^{-3}	2.0×10^{-5}	1.8×10^{-5}	1.3×10^{-3}
10-year latent fatal cancer risk	1.9×10^{-8}	2.9×10^{-10}	2.7×10^{-10}	1.9×10^{-8}
Average exposed individual within 80 km[c]				
Annual dose (mrem)	2.0×10^{-3}	7.4×10^{-6}	6.7×10^{-6}	2.0×10^{-3}
10-year latent fatal cancer risk	1.0×10^{-8}	3.7×10^{-11}	3.4×10^{-11}	1.0×10^{-8}

[Text deleted.]
[a] Totals represent the largest possible sums for each public category. Totals are additive in all cases because the same groups or individuals would receive doses from both facilities.
[b] The annual natural background radiation level at SRS is 295 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive about 232,000 person-rem.
[c] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of the SRS facilities in 2010 (about 790,000).
Source: Model results.

Table J–57. Radiological Impacts on Involved Workers of Operation of New Pit Conversion and Immobilization Facilities at SRS

Impact	Pit Conversion	Immobilization (50 t)[a]	Total
Number of badged workers	383	339	772
Total dose (person-rem/yr)	192	254	446
10-year latent fatal cancers	0.77	1.0	1.8
Average worker dose (mrem/yr)	500	750	618[b]
10-year latent fatal cancer risk	2.0×10^{-3}	3.0×10^{-3}	2.5×10^{-3}

[a] The values would be the same for immobilization in either ceramic or glass.
[b] Represents an average of the doses for both facilities.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved with operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: UC 1998g, 1999c, 1999d.

J.4.2.5 Pit Conversion and MOX Facilities

J.4.2.5.1 Construction of Pit Conversion and MOX Facilities

No radiological risk would be incurred by members of the public from the construction of new pit conversion and MOX facilities at SRS. Construction worker exposures to radiation that derives from other activities at the site, past or present, would also be kept as low as is reasonably achievable. Construction workers would be monitored (badged) as appropriate. Summaries of radiological impacts of these activities are presented in Table J–58 for workers at risk.

Table J–58. Potential Radiological Impacts on Construction Workers of New Pit Conversion and MOX Facilities at SRS

Impact	Pit Conversion	MOX	Total
Annual average number of workers	341	292	633
Total dose (person-rem/yr)	1.4	1.2	2.6
Annual latent fatal cancers[a]	5.6×10^{-4}	4.8×10^{-4}	1.0×10^{-3}
Average worker dose (mrem/yr)	4	4	4[b]
Annual latent fatal cancer risk	1.6×10^{-6}	1.6×10^{-6}	1.6×10^{-6}

[a] Values are based on a risk factor of 400 latent fatal cancers per million person-rem set by the National Research Council's Committee on the Biological Effects of Ionizing Radiations.
[b] Represents an average of the doses for both facilities.
Note: The radiological limit for a construction worker is 100 mrem/yr because they are categorized as members of the public (DOE 1993). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: ICRP 1991; NAS 1990; UC 1998g, 1998h.

J.4.2.5.2 Operation of Pit Conversion and MOX Facilities

Tables J–59 and J–60 present the incident-free radiological impacts of operation of the new pit conversion and MOX facilities at SRS.

Table J–59. Potential Radiological Impacts on the Public of Operation of New Pit Conversion and MOX Facilities at SRS

Impact	Pit Conversion	MOX[a]	Total[b]
Population within 80 km for year 2010			
Dose (person-rem)	1.6	0.18	1.8
Percent of natural background[c]	6.9×10^{-4}	7.8×10^{-5}	7.7×10^{-4}
10-year latent fatal cancers	8.0×10^{-3}	9.1×10^{-4}	8.9×10^{-3}
Maximally exposed individual			
Annual dose (mrem)	3.7×10^{-3}	3.7×10^{-3}	7.4×10^{-3}
Percent of natural background[c]	1.3×10^{-3}	1.3×10^{-3}	2.5×10^{-3}
10-year latent fatal cancer risk	1.9×10^{-8}	1.9×10^{-8}	3.7×10^{-8}
Average exposed individual within 80 km[d]			
Annual dose (mrem)	2.0×10^{-3}	2.3×10^{-4}	2.2×10^{-3}
10-year latent fatal cancer risk	1.0×10^{-8}	1.2×10^{-9}	1.1×10^{-8}

[a] Includes a dose component from liquid pathways because it is possible that liquid releases could reach these pathways at SRS.
[b] Totals are additive in all cases because the same groups or individuals would receive doses from both facilities.
[c] The annual natural background radiation level at SRS is 295 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive about 232,000 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of SRS in 2010 (about 790,000).
Source: Model results.

Table J–60. Potential Radiological Impacts on Involved Workers of Operation of New Pit Conversion and MOX Facilities at SRS

Impact	Pit Conversion	MOX	Total
Number of badged workers	383	331	714
Total dose (person-rem/yr)	192	22	214
10-year latent fatal cancers	0.77	0.088	0.86
Average worker dose (mrem/yr)	500	65	300[a]
10-year latent fatal cancer risk	2.0×10^{-3}	2.6×10^{-4}	1.2×10^{-3}

[a] Represents an average of the doses for both facilities.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: DOE 1999; UC 1998g, 1998h.

J.4.2.6 Immobilization and MOX Facilities

J.4.2.6.1 Construction of Immobilization and MOX Facilities

No radiological risk would be incurred by members of the public from the construction of new immobilization and MOX facilities at SRS. Construction worker exposures to radiation deriving from other activities, past or present, at the site would also be kept as low as is reasonably achievable. Construction workers would be monitored (badged) as appropriate. Summaries of radiological impacts of these activities are presented in Table J–61 for workers at risk.

Table J–61. Potential Radiological Impacts on Construction Workers of New Immobilization and MOX Facilities at SRS

Impact	Immobilization[a]	MOX	Total
Annual average number of workers	374	292	666
Total dose (person-rem/yr)	1.5	1.2	2.7
Annual latent fatal cancers[b]	6.0×10^{-4}	4.8×10^{-4}	1.1×10^{-3}
Average worker dose (mrem/yr)	4	4	4[c]
Annual latent fatal cancer risk	1.6×10^{-6}	1.6×10^{-6}	1.6×10^{-6}

[a] The values would be the same for immobilization in either ceramic or glass.
[b] Values are based on a risk factor of 400 latent fatal cancers per million person-rem set by the National Research Council's Committee on the Biological Effects of Ionizing Radiations.
[c] Represents an average of the doses for both facilities.
Note: The radiological limit for a construction worker is 100 mrem/yr because they are categorized as members of the public (DOE 1993). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: ICRP 1991; NAS 1990; UC 1998h, 1999c, 1999d.

J.4.2.6.2 Operation of Immobilization and MOX Facilities

Tables J–62 and J–63 present the incident-free radiological impacts of operation of the new immobilization and MOX facilities at SRS.

Table J–62. Potential Radiological Impacts on the Public of Operation of New Immobilization and MOX Facilities at SRS

	Immobilization (17 t)			
Impact	Ceramic	Glass	MOX[a]	Total[b]
Population within 80 km for year 2010				
Dose (person-rem)	2.8×10^{-3}	2.6×10^{-3}	0.18	0.18
Percent of natural background[c]	1.2×10^{-6}	1.1×10^{-6}	7.8×10^{-5}	7.9×10^{-5}
10-year latent fatal cancers	1.4×10^{-5}	1.3×10^{-5}	9.1×10^{-4}	9.2×10^{-4}
Maximally exposed individual				
Annual dose (mrem)	2.8×10^{-5}	2.6×10^{-5}	3.7×10^{-3}	3.7×10^{-3}
Percent of natural background[c]	9.5×10^{-6}	8.8×10^{-6}	1.3×10^{-3}	1.3×10^{-3}
10-year latent fatal cancer risk	1.4×10^{-10}	1.3×10^{-10}	1.9×10^{-8}	1.9×10^{-8}
Average exposed individual within 80 km[d]				
Annual dose (mrem)	3.6×10^{-6}	3.3×10^{-6}	2.3×10^{-4}	2.3×10^{-4}
10-year latent fatal cancer risk	1.8×10^{-11}	1.6×10^{-11}	1.2×10^{-9}	1.2×10^{-9}

[Text deleted.]
[a] Includes a dose component from liquid pathways because it is possible that liquid releases could reach these pathways at SRS.
[b] Totals represent the largest possible sums for each public category. Totals are additive in all cases because the same groups or individuals would receive doses from both facilities.
[c] The annual natural background radiation level at SRS is 295 mrem for the average individual; the population within 80 km (50 mi) in 2010 would receive about 232,000 person-rem.
[d] Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of the SRS facilities in 2010 (about 790,000).
Source: Model results.

Table J–63. Potential Radiological Impacts on Involved Workers of Operation of New Immobilization and MOX Facilities at SRS

Impact	Immobilization (17 t)[a]	MOX	Total
Number of badged workers	323	331	654
Total dose (person-rem/yr)	242	22	264
10-year latent fatal cancers	0.97	0.088	1.1
Average worker dose (mrem/yr)	750	65	404[b]
10-year latent fatal cancer risk	3.0×10^{-3}	2.6×10^{-4}	1.6×10^{-3}

[a] The values would be the same for immobilization in either ceramic or glass.
[b] Represents an average of the doses for both facilities.
Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: DOE 1999; UC 1998h, 1999c, 1999d.

J.4.2.7 Pit Conversion, Immobilization, and MOX Facilities

J.4.2.7.1 Construction of Pit Conversion, Immobilization, and MOX Facilities

No radiological risk would be incurred by members of the public from the construction of new pit conversion, immobilization, and MOX facilities at SRS. Construction worker exposures to radiation that derives from other activities at the site, past or present, would also be kept as low as is reasonably achievable. Construction workers would be monitored (badged) as appropriate. Summaries of radiological impacts of these activities are presented in Table J–64 for workers at risk.

Table J–64. Potential Radiological Impacts on Construction Workers of New Pit Conversion, Immobilization, and MOX Facilities at SRS

Impact	Pit Conversion	Immobilization[a]	MOX	Total
Annual average number of workers	341	374	292	1,007
Total dose (person-rem/yr)	1.4	1.5	1.2	4.1
Annual latent fatal cancers[b]	5.6×10^{-4}	6.0×10^{-4}	4.8×10^{-4}	1.6×10^{-3}
Average worker dose (mrem/yr)	4	4	4	4[c]
Annual latent fatal cancer risk	1.6×10^{-6}	1.6×10^{-6}	1.6×10^{-6}	1.6×10^{-6}

[a] The values would be the same for immobilization in either ceramic or glass.
[b] Values are based on a risk factor of 400 latent fatal cancers per million person-rem set by the National Research Council's Committee on the Biological Effects of Ionizing Radiations.
[c] Represents an average of the doses for all three facilities.
Note: The radiological limit for construction workers is 100 mrem/yr because they are categorized as members of the public (DOE 1993). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.
Source: ICRP 1991; NAS 1990; UC 1998g, 1998h, 1999c, 1999d.

J.4.2.7.2 Operation of Pit Conversion, Immobilization, and MOX Facilities

Tables J–65 and J–66 present all possible incident-free radiological impact scenarios of operation of all three new facilities at SRS.

Table J–65. Potential Radiological Impacts on the Public of Operation of New Pit Conversion, Immobilization, and MOX Facilities at SRS

Impact	Pit Conversion	Immobilization (17 t)		MOX[a]	Total[b]
		Ceramic	Glass		
Population within 80 km for year 2010					
Dose (person-rem)	1.6	2.8×10^{-3}	2.6×10^{-3}	0.18	1.8
Percent of natural background[c]	6.9×10^{-4}	1.2×10^{-6}	1.1×10^{-6}	7.8×10^{-5}	7.8×10^{-4}
10-year latent fatal cancers	8.0×10^{-3}	1.4×10^{-5}	1.3×10^{-5}	9.1×10^{-4}	9.0×10^{-3}
Maximally exposed individual					
Annual dose (mrem)	3.7×10^{-3}	2.8×10^{-5}	2.6×10^{-5}	3.7×10^{-3}	7.4×10^{-3}
Percent of natural background[c]	1.3×10^{-3}	9.5×10^{-6}	8.8×10^{-6}	1.3×10^{-3}	2.5×10^{-3}
10-year latent fatal cancer risk	1.9×10^{-8}	1.4×10^{-10}	1.3×10^{-10}	1.9×10^{-8}	3.7×10^{-8}
Average exposed individual within 80 km[d]					
Annual dose (mrem)	2.0×10^{-3}	3.6×10^{-6}	3.3×10^{-6}	2.3×10^{-4}	2.2×10^{-3}
10-year latent fatal cancer risk	1.0×10^{-8}	1.8×10^{-11}	1.6×10^{-11}	1.2×10^{-9}	1.1×10^{-8}

[Text deleted.]

a Includes a dose component from liquid pathways because it is possible that liquid releases could reach these pathways at SRS.

b Totals represent the largest possible sums for each public category. Totals are additive in all cases because the same groups or individuals would receive doses from all three facilities.

c The annual natural background radiation level at SRS is 295 mrem for the average individual; the population within 80 km (50 mi) in the year 2010 receives about 232,000 person-rem.

d Obtained by dividing the population dose by the number of people projected to live within 80 km (50 mi) of the SRS facilities in 2010 (about 790,000).

Source: Model results.

Table J–66. Potential Radiological Impacts on Involved Workers of Operation of New Pit Conversion, Immobilization, and MOX Facilities at SRS

Impact	Pit Conversion	Immobilization (17 t)[a]	MOX	Total
Number of badged workers	383	323	331	1,037
Total dose (person-rem/yr)	192	242	22	456
10-year latent fatal cancers	0.77	0.97	0.088	1.8
Average worker dose (mrem/yr)	500	750	65	440[b]
10-year latent fatal cancer risk	2.0×10^{-3}	3.0×10^{-3}	2.6×10^{-4}	1.8×10^{-3}

a The values would be the same for immobilization in either ceramic or glass.

b Represents an average of the doses for all three facilities.

Note: The radiological limit for an individual worker is 5,000 mrem/yr (DOE 1995). However, the maximum dose to a worker involved in operations would be kept below the DOE administrative control level of 2,000 mrem/yr (DOE 1994). An effective ALARA program would ensure that doses are reduced to levels that are as low as is reasonably achievable.

Source: DOE 1999; UC 1998g, 1998h, 1999c, 1999d.

J.5 LEAD ASSEMBLY FABRICATION

J.5.1 ANL–W

J.5.1.1 Assessment Data

This section presents applicable data and assumptions used in the assessment of lead assembly human health risks at ANL–W at INEEL. Appendix F.10 provides a summary of the methods and tools (e.g., the GENII computer code) used for the assessment.

J.5.1.1.1 Meteorological Data

The meteorological data used for the ANL–W dose assessments was in the form of a JFD file. A JFD file is a table listing the percentages of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The JFD file was based on measurements taken over a period of several years at a specific location and height. Average annual meteorological conditions, averaged over the measurement period, were used for normal operations. Table J–20 presents the JFD used in the dose assessments for ANL–W.

J.5.1.1.2 Population Data

The INEEL population distribution was based on the *1990 Census of Population and Housing Data* (DOC 1992). Projections were determined for the year 2005 for areas within 80 km (50 mi) of the proposed facility location. The site population in 2005 was assumed to be representative of the population over the operational period evaluated. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances out to an 80-km (50-mi) distance. The grid was centered at ANL–W, the location from which radionuclides are assumed to be released during incident-free operations. Table J–67 presents the population data used for the lead assembly dose assessments at ANL–W.

J.5.1.1.3 Agricultural Data

The 1987 Census of Agriculture was the source used to generate site-specific data for food production. Food production was spatially distributed on a circular grid similar to that used for the population distributions described previously. This food grid (or wheel) was generated by combining the fraction of a county in each segment (e.g., south, southwest, north-northeast) and the county production of the eight food categories analyzed by GENII—leafy vegetables, root vegetables, fruits, grains, beef, poultry, milk, and eggs. Each county's food production was assumed to be distributed uniformly over the given county's land area. These categorized food wheels were then used in the assessment of doses to the population from the ingestion pathway. The consumption rates used in the dose assessments were those for the MEI and average exposed individual. People living within the 80-km (50-mi) assessment area were assumed to consume only food grown in that area. ANL–W food production and consumption data used for the dose assessments in the SPD EIS were obtained from the *Health Risk Data for Storage and Disposition Final PEIS* (HNUS 1996).

J.5.1.1.4 Source Term Data

Estimated incident-free radiological releases associated with the MOX fuel lead assembly facility are presented in Table J–68. Stack height and release location are provided in the Oak Ridge National Laboratory (ORNL) *ANL-W MOX Fuel Lead Assemblies Data Report for the Surplus Plutonium Disposition Environmental Impact Statement* (O'Connor et al. 1998a).

Table J–67. Projected INEEL Population Surrounding ANL–W for Year 2005

Direction	Distance (mi)										Total
	0–1	1–2	2–3	3–4	4–5	5–10	10–20	20–30	30–40	40–50	
S	0	0	0	0	0	0	277	2,086	6,173	30,883	**39,419**
SSW	0	0	0	0	0	0	273	323	906	3,267	**4,769**
SW	0	0	0	0	0	0	246	247	224	334	**1,051**
WSW	0	0	0	0	0	0	0	238	177	181	**596**
W	0	0	0	0	0	0	0	179	224	528	**931**
WNW	0	0	0	0	0	0	35	474	824	467	**1,800**
NW	0	0	0	0	0	0	36	57	280	929	**1,302**
NNW	0	0	0	0	0	0	0	81	76	76	**233**
N	0	0	0	0	0	0	0	254	140	146	**540**
NNE	0	0	0	0	0	0	252	450	266	158	**1,126**
NE	0	0	0	0	0	0	252	443	515	98	**1,308**
ENE	0	0	0	0	0	0	253	706	1,411	5,196	**7,566**
E	0	0	0	0	0	0	367	1,405	18,570	32,506	**52,848**
ESE	0	0	0	0	0	103	509	4,197	90,875	756	**96,440**
SE	0	0	0	0	17	80	589	3,523	11,502	411	**16,122**
SSE	0	0	0	0	17	52	279	4,816	19,230	1,068	**25,462**
Total	**0**	**0**	**0**	**0**	**34**	**235**	**3,368**	**19,479**	**151,393**	**77,004**	**251,513**

Key: ANL–W, Argonne National Laboratory–West.
Source: DOC 1992.

Table J–68. Estimated Incident-Free Annual Radiological Releases From the MOX Lead Assembly Facility at ANL–W

Isotope	(μCi/yr)
Plutonium 236	–
Plutonium 238	0.85
Plutonium 239	23
Plutonium 240	5.3
Plutonium 241	58
Plutonium 242	9.3×10^{-4}
Americium 241	2.0
Uranium 234	1.3×10^{-3}
Uranium 235	5.4×10^{-5}
Uranium 238	3.1×10^{-3}

Source: O'Connor et al. 1998a.

J.5.1.1.5 Other Calculational Assumptions

To estimate radiological impacts of incident-free operation of the lead assembly facility at ANL–W, the following additional assumptions and factors were considered, in accordance with the guidelines established in NRC Regulatory Guide 1.109 (NRC 1977).

Ground surfaces were assumed to have no previous deposition of radionuclides for the purposes of modeling the incremental radiological impacts associated with surplus plutonium disposition activities.

However, doses associated with true instances of prior deposition are accounted for in the Affected Environment and Cumulative Impacts sections.

The annual external exposure time to the plume and to soil contamination was 0.7 year for the MEI (NRC 1977).

The annual external exposure time to the plume and to soil contamination was 0.5 year for the population (NRC 1977).

The annual inhalation exposure time to the plume was 1 year for the MEI and general population (NRC 1977).

The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of the adult human.

A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of contaminated animal products. Drinking water, aquatic food ingestion, and any other pathway that may involve liquid exposure were not examined because all releases are to the air.

Reported stack heights were used for atmospheric releases and were assumed to be the effective stack height. The resultant doses were conservative because use of the actual stack height negates plume rise.

The calculated doses are 50-year committed doses from 1 year of intake.

J.5.1.2 Human Health Impacts

Potential radiological impacts on the public and workers resulting from normal lead assembly operations are presented in Section 4.27.1.4. Potential impacts on postirradiation examination facility workers are presented in Section 4.27.6.2.

J.5.2 Hanford

J.5.2.1 Assessment Data

This section presents applicable data and assumptions used in the assessment of lead assembly human health risks at Hanford. Appendix F.10 provides a summary of the methods and tools (e.g., the GENII computer code) used for the assessment.

J.5.2.1.1 Meteorological Data

The meteorological data used for the Hanford dose assessments was in the form of a JFD file. A JFD file is a table listing the percentages of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The JFD file was based on measurements taken over a period of several years at a specific location and height. Average annual meteorological conditions, averaged over the measurement period, were used for normal operations. Table J–1 presents the JFD used in the dose assessments for Hanford.

J.5.2.1.2 Population Data

The Hanford population distribution was based on the *1990 Census of Population and Housing Data* (DOC 1992). Projections were determined for the year 2005 for areas within 80 km (50 mi) of the proposed facility location. The site population in 2005 was assumed to be representative of the population over the operational period evaluated. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances out to an 80-km (50-mi) distance. The grid was centered at FMEF in the 400 Area, the location from which radionuclides are assumed to be released during incident-free operations. Table J–69 presents the population data used for lead assembly dose assessments at Hanford.

Table J–69. Projected Hanford Population Surrounding FMEF for Year 2005

Direction	Distance (mi)										Total
	0–1	1–2	2–3	3–4	4–5	5–10	10–20	20–30	30–40	40–50	
S	0	0	0	0	0	3,886	40,763	1,039	7,050	19,641	**72,379**
SSW	0	0	0	0	2	1,380	2,513	399	2,888	3,828	**11,010**
SW	0	0	0	0	38	1,265	4,361	288	207	1,923	**8,082**
WSW	0	0	0	0	0	50	2,175	15,734	3,338	300	**21,597**
W	0	0	0	0	0	0	698	5,764	26,190	14,858	**47,510**
WNW	0	0	0	0	0	0	5	813	1,147	8,446	**10,411**
NW	0	0	0	0	0	0	0	592	377	163	**1,132**
NNW	0	0	0	0	0	0	0	1,034	1,317	1,362	**3,713**
N	0	0	0	0	0	0	0	1,224	3,458	2,520	**7,202**
NNE	0	0	0	0	0	16	425	5,074	1,388	23,720	**30,623**
NE	0	0	0	0	0	86	751	6,743	2,769	1,153	**11,502**
ENE	0	0	0	0	0	313	1,401	3,391	385	410	**5,900**
E	0	0	0	0	0	386	861	410	319	300	**2,276**
ESE	0	0	0	0	0	393	595	315	245	302	**1,850**
SE	0	0	0	0	0	381	1,191	1,604	366	1,364	**4,906**
SSE	0	0	0	0	0	6,366	79,333	30,715	565	979	**117,958**
Total	**0**	**0**	**0**	**0**	**40**	**14,522**	**135,072**	**75,139**	**52,009**	**81,269**	**358,051**

Key: FMEF, Fuels and Materials Examination Facility.
Source: DOC 1992.

J.5.2.1.3 Agricultural Data

The 1987 Census of Agriculture was the source used to generate site-specific data for food production. Food production was spatially distributed on a circular grid similar to that used for the population distributions described previously. This food grid (or wheel) was generated by combining the fraction of a county in each segment (e.g., south, southwest, north-northeast) and the county production of the eight food categories analyzed by GENII—leafy vegetables, root vegetables, fruits, grains, beef, poultry, milk, and eggs. Each county's food production was assumed to be distributed uniformly over the given county's land area. These categorized food wheels were then used in the assessment of doses to the population from the ingestion pathway. The consumption rates used in the dose assessments were those for the MEI and average exposed individual. People living within the 80-km (50-mi) assessment area were assumed to consume only food grown in that area. Hanford food production and consumption data used for the dose assessments in the SPD EIS were obtained from the *Health Risk Data for Storage and Disposition Final PEIS* (HNUS 1996).

J.5.2.1.4 Source Term Data

Estimated incident-free radiological releases associated with the MOX fuel lead assembly facility are presented in Table J–70. Stack height and release location are reported in the ORNL *Hanford MOX Fuel Lead Assemblies Data Report for the Surplus Plutonium Disposition Environmental Impact Statement* (O'Connor et al. 1998b).

Table J–70. Estimated Incident-Free Annual Radiological Releases From the MOX Lead Assembly Facility at Hanford

Isotope	(μCi/yr)
Plutonium 236	–
Plutonium 238	0.85
Plutonium 239	23
Plutonium 240	5.3
Plutonium 241	58
Plutonium 242	9.3×10^{-4}
Americium 241	2.0
Uranium 234	1.3×10^{-3}
Uranium 235	5.4×10^{-5}
Uranium 238	3.1×10^{-3}

Source: O'Connor et al. 1998b.

J.5.2.1.5 Other Calculational Assumptions

To estimate radiological impacts of incident-free operation of the lead assembly facility at Hanford, the following additional assumptions and factors were considered, in accordance with the guidelines established in NRC Regulatory Guide 1.109 (NRC 1977).

Ground surfaces were assumed to have no previous deposition of radionuclides for the purposes of modeling the incremental radiological impacts associated with surplus plutonium disposition activities. However, doses associated with true instances of prior deposition are accounted for in the Affected Environment and Cumulative Impacts sections.

The annual external exposure time to the plume and to soil contamination was 0.7 year for the MEI (NRC 1977).

The annual external exposure time to the plume and to soil contamination was 0.5 year for the population (NRC 1977).

The annual inhalation exposure time to the plume was 1 year for the MEI and general population (NRC 1977).

The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of the adult human.

A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of contaminated animal products. Drinking water, aquatic food ingestion, and any other pathway that may involve liquid exposure were not examined because all releases are to the air.

Reported stack heights were used for atmospheric releases and were assumed to be the effective stack height. The resultant doses were conservative because use of the actual stack height negates plume rise.

The calculated doses are 50-year committed doses from 1 year of intake.

J.5.2.2 Human Health Impacts

Potential radiological impacts on the public and workers resulting from normal lead assembly operations are presented in Section 4.27.2.4.

J.5.3 LLNL

J.5.3.1 Assessment Data

This section presents applicable data and assumptions used in the assessment of lead assembly human health risks at LLNL. Appendix F.10 provides a summary of the methods and tools (e.g., the GENII computer code) used for the assessment.

J.5.3.1.1 Meteorological Data

The meteorological data used for the LLNL dose assessments was in the form of a JFD file. A JFD file is a table listing the percentages of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The JFD file was based on measurements taken at a specific location and height. Annual meteorological conditions were used for normal operations. Table J–71 presents the JFD used in the dose assessments for LLNL.

J.5.3.1.2 Population Data

The LLNL population distribution was based on the *1990 Census of Population and Housing Data* (DOC 1992). Projections were determined for the year 2005 for areas within 80 km (50 mi) of the proposed facility location. The site population in 2005 was assumed to be representative of the population over the operational period evaluated. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances out to an 80-km (50-mi) distance. The grid was centered at Building 332, the location from which radionuclides are assumed to be released during incident-free operations. Table J–72 presents the population data that were used for lead assembly dose assessments at LLNL.

J.5.3.1.3 Agricultural Data

The 1992 Census of Agriculture (DOC 1992) was the source used to generate site-specific data for food production. Food production was spatially distributed on a circular grid similar to that used for the population distributions described previously. This food grid (or wheel) was generated by combining the fraction of a county in each segment (e.g., south, southwest, north-northeast) and the county production of the eight food categories analyzed by GENII—leafy vegetables, root vegetables, fruits, grains, beef, poultry, milk, and eggs. Each county's food production was assumed to be distributed uniformly over the given county's land area. These categorized food wheels were then used in the assessment of doses to the population from the ingestion pathway. The consumption rates used in the dose assessments were those for the MEI and average exposed individual. People living within the 80-km (50-mi) assessment area were assumed to consume only food grown in that area. LLNL food production and consumption data used for the dose assessments in the SPD EIS were obtained from the 1992 census data for LLNL (DOC 1992).

Table J–71. LLNL 1993 Joint Frequency Distributions at 10-m Height

Wind Speed (m/s)	Stability Class	Wind Blows Toward															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
0.89	A	0.45	0.41	0.4	0.33	0.27	0.17	0.14	0.11	0.13	0.34	0.62	1.14	1.53	0.78	0.57	0.45
	B	0.22	0.11	0.1	0.11	0.1	0.03	0.03	0.01	0.07	0.05	0.27	0.41	0.17	0.17	0.14	0.09
	C	0.13	0.09	0.15	0.03	0.02	0.01	0	0.03	0.08	0.14	0.16	0.22	0.16	0.09	0.08	0.07
	D	0.17	0.33	0.45	0.53	0.65	0.67	0.23	0.34	1.05	1.86	1.21	0.7	0.27	0.13	0.05	0.03
	E	0.18	0.33	0.86	0.99	1.01	1.13	0.39	0.48	1.07	1.7	0.74	0.41	0.25	0.06	0.09	0.03
	F	0.11	0.16	0.61	0.93	0.8	0.63	0.55	0.31	0.35	0.38	0.39	0.14	0.1	0.08	0.11	0.07
	G	0.62	0.74	1.06	1.64	1.97	1.78	1.53	0.97	0.73	0.75	0.49	0.48	0.34	0.27	0.35	0.37
2.86	A	0.3	0.37	0.24	0.18	0.03	0.02	0.02	0.01	0	0.02	0.26	0.81	0.89	0.31	0.21	0.16
	B	0.4	0.39	0.77	0.16	0	0.03	0.02	0.01	0.02	0.08	0.39	1.26	1.15	0.22	0.07	0.21
	C	0.07	0.59	1.21	0	0	0	0	0.01	0.02	0.09	0.7	1.28	1.17	0.23	0.01	0.03
	D	0.03	0.82	1.04	0.03	0	0	0.03	0.09	0.25	1.14	4.88	2.71	1.81	0.21	0.02	0
	E	0.07	0.13	0.27	0.07	0	0	0.05	0.06	0.63	1.91	0.93	0.16	0.03	0	0	0.02
	F	0.03	0.03	0.16	0.1	0.01	0.02	0.01	0.02	0.03	0.02	0.06	0.02	0.01	0.02	0.01	0.01
	G	0.01	0.05	0.07	0.06	0.05	0.02	0.03	0.02	0.05	0.03	0.06	0	0	0	0.01	0.01
4.71	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0.34	0.71	0.23	0.02	0	0.02	0	0.05	0.01	0.03	0.3	1.22	1.62	0.16	0.01	0
	D	0.08	0.72	0.56	0	0	0	0	0.06	0.09	0.61	3.64	1.51	2.04	0.11	0.01	0.02
	E	0	0.02	0	0	0	0	0	0	0	0.15	0.17	0.01	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	G	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
6.69	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	D	0.15	0.24	0.02	0	0	0	0	0	0.03	0.45	1.25	0.32	0.13	0.03	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	G	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
8.68	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	D	0.07	0.08	0	0	0	0	0	0	0.02	0.07	0.02	0	0.01	0	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	G	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Table J–71. LLNL 1993 Joint Frequency Distributions at 10-m Height (Continued)

Wind Speed (m/s)	Stability Class	Wind Blows Toward															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
10.5	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	D	0	0	0	0	0	0	0	0	0	0.01	0	0	0	0	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	G	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Key: LLNL, Lawrence Livermore National Laboratory.
Source: Gouveia 1997.

Table J–72. Projected LLNL Population Surrounding Building 332 for Year 2005

Direction	Distance (mi)										Total
	0–1	1–2	2–3	3–4	4–5	5–10	10–20	20–30	30–40	40–50	
S	5	14	6	8	10	84	178	157	15,286	56,124	**71,872**
SSW	5	15	13	8	10	47	1,080	301,887	190,271	27,874	**521,210**
SW	31	538	25	18	16	91	42,723	589,979	350,562	52,017	**1,036,000**
WSW	228	1,283	660	982	1,885	644	146,903	239,224	184,580	4,845	**581,234**
W	302	1,316	3,338	6,379	9,931	24,309	112,488	123,480	333,290	64,111	**678,944**
WNW	311	1,316	4,567	6,337	8,349	20,051	92,859	476,610	570,787	545,627	**1,726,814**
NW	272	1,316	1,770	2,274	212	677	78,366	170,569	454,881	135,688	**846,025**
NNW	109	1,423	2,850	2,109	53	404	8,150	275,850	117,234	154,923	**563,105**
N	5	49	1,094	324	39	367	4,555	139,309	1,444	230,332	**377,518**
NNE	5	15	25	35	45	283	13,831	24,535	7,317	5,523	**51,614**
NE	5	15	16	25	21	127	8,403	12,091	128,594	36,124	**185,421**
ENE	5	11	6	8	10	111	2,218	130,249	211,561	11,360	**355,539**
E	5	14	8	8	10	249	54,523	86,577	30,047	47,622	**219,063**
ESE	5	15	17	8	10	103	1,898	7,484	230,939	242,714	**483,193**
SE	5	15	10	8	10	91	512	902	18,290	23,344	**43,187**
SSE	5	12	6	8	10	85	314	83	26	1,063	**1,612**
Total	**1,303**	**7,367**	**14,411**	**18,539**	**20,621**	**47,723**	**569,001**	**2,578,986**	**2,845,109**	**1,639,291**	**7,742,351**

Key: LLNL, Lawrence Livermore National Laboratory.
Source: DOC 1992.

J.5.3.1.4 Source Term Data

Estimated incident-free radiological releases associated with the MOX fuel lead assembly facility are presented in Table J–73. Stack height and release location are provided in the ORNL *LLNL MOX Fuel Lead Assemblies Data Report for the Surplus Plutonium Disposition Environmental Impact Statement* (O'Connor et al. 1998c).

Table J–73. Estimated Incident-Free Annual Radiological Releases From the MOX Lead Assembly Facility at LLNL

Isotope	(μCi/yr)
Plutonium 236	–
Plutonium 238	0.85
Plutonium 239	23
Plutonium 240	5.3
Plutonium 241	58
Plutonium 242	9.3×10^{-4}
Americium 241	2.0
Uranium 234	1.3×10^{-3}
Uranium 235	5.4×10^{-5}
Uranium 238	3.1×10^{-3}

Source: O'Connor et al. 1998c.

J.5.3.1.5 Other Calculational Assumptions

To estimate radiological impacts of incident-free operation of the lead assembly facility at LLNL, the following additional assumptions and factors were considered, in accordance with the guidelines established in NRC Regulatory Guide 1.109 (NRC 1977).

Ground surfaces were assumed to have no previous deposition of radionuclides for the purposes of modeling the incremental radiological impacts associated with surplus plutonium disposition activities. However, doses associated with true instances of prior deposition are accounted for in the Affected Environment and Cumulative Impacts sections.

The annual external exposure time to the plume and to soil contamination was 0.7 year for the MEI (NRC 1977).

The annual external exposure time to the plume and to soil contamination was 0.5 year for the population (NRC 1977).

The annual inhalation exposure time to the plume was 1 year for the MEI and general population (NRC 1977).

The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of the adult human.

A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of contaminated animal products. Drinking water, aquatic food ingestion, and any other pathway that may involve liquid exposure were not examined because all releases are to the air.

Reported stack heights were used for atmospheric releases and were assumed to be the effective stack height. The resultant doses were conservative because use of the actual stack height negates plume rise.

The calculated doses are 50-year committed doses from 1 year of intake.

J.5.3.2 Human Health Impacts

Potential radiological impacts on the public and workers resulting from normal lead assembly operations are presented in Section 4.27.3.4.

J.5.4 LANL

J.5.4.1 Assessment Data

This section presents applicable data and assumptions used in the assessment of lead assembly human health risks at LANL. Appendix F.10 provides a summary of the methods and tools (e.g., the GENII computer code) used for the assessment.

J.5.4.1.1 Meteorological Data

The meteorological data used for the LANL dose assessments was in the form of a JFD file. A JFD file is a table listing the percentages of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The JFD file was based on measurements taken at a specific location and height. Annual meteorological conditions were used for normal operations. Table J–74 presents the JFD used in the dose assessments for LANL.

J.5.4.1.2 Population Data

The LANL population distribution was based on the *1990 Census of Population and Housing Data* (DOC 1992). Projections were determined for the year 2005 for areas within 80 km (50 mi) of the proposed facility location. The site population in 2005 was assumed to be representative of the population over the operational period evaluated. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances out to an 80-km (50-mi) distance. The grid was centered at Technical Area 55 (TA–55), the location from which radionuclides are assumed to be released during incident-free operations. Table J–75 presents the population data used for lead assembly dose assessments at LANL.

J.5.4.1.3 Agricultural Data

The 1992 Census of Agriculture was the source used to generate site-specific data for food production. Food production was spatially distributed on a circular grid similar to that used for the population distributions described previously. This food grid (or wheel) was generated by combining the fraction of a county in each segment (e.g., south, southwest, north-northeast) and the county production of the eight food categories analyzed by GENII—leafy vegetables, root vegetables, fruits, grains, beef, poultry, milk, and eggs. Each county's food production was assumed to be distributed uniformly over the given county's land area. These categorized food wheels were then used in the assessment of doses to the population from the ingestion pathway. The consumption rates used in the dose assessments were those for the MEI and average exposed individual. People living within the 80-m (50-mi) assessment area were assumed to consume only food grown in that area. LANL food production and consumption data used for the dose assessments in the SPD EIS were obtained from the *Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy Stored at the Rocky Flats Environmental Technology Site* (DOE 1998).

Table J–74. LANL 1993–1996 Joint Frequency Distributions at 11-m Height

Wind Speed (m/s)	Stability Class	Wind Blows Toward															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
0.78	A	0.12	0.26	0.5	0.84	0.74	0.54	0.45	0.32	0.18	0.11	0.08	0.05	0.06	0.06	0.07	0.07
	B	0.03	0.05	0.12	0.19	0.16	0.09	0.08	0.07	0.04	0.01	0.02	0.01	0.02	0.02	0.01	0.02
	C	0.05	0.09	0.14	0.2	0.16	0.09	0.09	0.09	0.07	0.04	0.03	0.03	0.02	0.03	0.02	0.03
	D	0.86	0.69	0.57	0.45	0.47	0.34	0.33	0.33	0.38	0.35	0.33	0.31	0.35	0.4	0.57	0.72
	E	0.59	0.45	0.33	0.23	0.22	0.15	0.13	0.13	0.17	0.24	0.32	0.28	0.29	0.4	0.51	0.62
	F	0.26	0.28	0.27	0.19	0.18	0.17	0.2	0.25	0.3	0.32	0.22	0.17	0.15	0.2	0.24	0.25
2.5	A	0.03	0.07	0.17	0.45	0.56	0.43	0.33	0.22	0.18	0.08	0.06	0.05	0.04	0.03	0.03	0.03
	B	0.02	0.05	0.2	0.39	0.42	0.31	0.27	0.22	0.16	0.1	0.06	0.05	0.05	0.04	0.03	0.02
	C	0.05	0.15	0.46	0.68	0.65	0.45	0.46	0.59	0.59	0.26	0.16	0.12	0.16	0.12	0.07	0.05
	D	0.95	1.09	0.94	0.72	0.56	0.34	0.47	1.3	2.12	1.89	1.93	0.95	1.08	0.81	0.56	0.63
	E	0.87	0.59	0.34	0.19	0.11	0.1	0.13	0.24	0.67	1.82	2.41	1.72	1.84	1.41	0.8	0.8
	F	0.09	0.07	0.05	0.03	0.01	0.01	0.05	0.1	0.25	0.33	0.11	0.36	0.39	0.39	0.12	0.07
4.5	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0.01	0.01	0.01	0.02	0.01	0
	C	0.02	0.04	0.07	0.04	0.02	0.01	0.01	0.03	0.15	0.09	0.11	0.19	0.31	0.19	0.09	0.02
	D	0.81	0.8	0.42	0.16	0.07	0.04	0.11	0.99	3.24	3.52	2.59	1.61	1.86	1.05	0.54	0.44
	E	0.21	0.2	0.08	0.01	0	0	0.01	0.07	0.32	1.74	1.08	1.32	1.31	0.32	0.23	0.22
	F	0	0.01	0	0	0	0	0	0	0.02	0.04	0	0.05	0.05	0.01	0.01	0
6.9	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0.01	0.01	0.01	0	0
	D	0.19	0.2	0.05	0	0	0	0.01	0.31	0.96	1.42	0.87	0.93	0.62	0.48	0.31	0.15
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
9.6	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	D	0.01	0.01	0	0	0	0	0	0.05	0.03	0.08	0.09	0.19	0.08	0.05	0.04	0.02
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
105	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	D	0	0	0	0	0	0	0	0.01	0	0	0.01	0.01	0	0	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Key: LANL, Los Alamos National Laboratory.
Source: LANL 1997.

Table J–75. Projected LANL Population Surrounding TA–55 for Year 2005

Direction	Distance (mi)										Total
	0–1	1–2	2–3	3–4	4–5	5–10	10–20	20–30	30–40	40–50	
S	0	0	25	26	44	221	701	1,606	1,125	2,962	**6,710**
SSW	0	0	26	20	56	21	1,373	4,464	4,949	43,596	**54,505**
SW	0	0	26	22	80	29	155	1,767	817	30,893	**33,789**
WSW	0	0	26	21	56	302	159	1,187	2,500	61	**4,312**
W	0	0	27	20	26	457	190	1,084	135	350	**2,289**
WNW	0	12	39	135	90	532	73	138	1,755	1,306	**4,080**
NW	0	152	1,287	2,379	1,500	720	102	195	248	274	**6,857**
NNW	0	427	844	224	126	421	169	211	174	220	**2,816**
N	500	585	264	107	137	560	609	688	659	289	**4,398**
NNE	0	480	61	57	56	463	958	919	658	143	**3,795**
NE	0	101	12	17	22	378	12,856	2,950	1,954	3,236	**21,526**
ENE	0	10	12	17	22	618	13,270	3,439	2,869	1,938	**22,195**
E	0	10	12	17	22	684	3,598	590	719	1,161	**6,813**
ESE	0	10	12	17	33	220	1,602	3,608	316	834	**6,652**
SE	0	0	0	0	4,488	952	6,143	76,455	4,503	742	**93,283**
SSE	0	0	0	117	85	224	5,021	10,633	2,091	483	**18,654**
Total	**500**	**1,787**	**2,673**	**3,196**	**6,843**	**6,802**	46,979	**109,934**	**25,472**	**88,488**	**292,674**

Key: LANL, Los Alamos National Laboratory; TA–55, Technical Area 55.
Source: DOC 1992.

J.5.4.1.4 Source Term Data

Estimated incident-free radiological releases associated with the MOX fuel lead assembly facility are presented in Table J–76. Stack height and release location are provided in the ORNL *LANL MOX Fuel Lead Assemblies Data Report for the Surplus Plutonium Disposition Environmental Impact Statement* (O'Connor et al. 1998d).

Table J–76. Estimated Incident-Free Annual Radiological Releases From the MOX Lead Assembly Facility at LANL

Isotope	(μCi/yr)
Plutonium 236	–
Plutonium 238	0.85
Plutonium 239	23
Plutonium 240	5.3
Plutonium 241	58
Plutonium 242	9.3×10^{-4}
Americium 241	2.0
Uranium 234	1.3×10^{-3}
Uranium 235	5.4×10^{-5}
Uranium 238	3.1×10^{-3}

Source: O'Connor et al. 1998d.

J.5.4.1.5 Other Calculational Assumptions

To estimate radiological impacts of incident-free operation of the lead assembly facility at LANL, the following additional assumptions and factors were considered, in accordance with the guidelines established in NRC Regulatory Guide 1.109 (NRC 1977).

Ground surfaces were assumed to have no previous deposition of radionuclides for the purposes of modeling the incremental radiological impacts associated with surplus plutonium disposition activities. However, doses associated with true instances of prior deposition are accounted for in the Affected Environment and Cumulative Impacts sections.

The annual external exposure time to the plume and to soil contamination was 0.7 year for the MEI (NRC 1977).

The annual external exposure time to the plume and to soil contamination was 0.5 year for the population (NRC 1977).

The annual inhalation exposure time to the plume was 1 year for the MEI and general population (NRC 1977).

The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of the adult human.

A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of contaminated animal products. Drinking water, aquatic food ingestion, and any other pathway that may involve liquid exposure were not examined because all releases are to the air.

Reported stack heights were used for atmospheric releases and were assumed to be the effective stack height. The resultant doses were conservative, because use of the actual stack height negates plume rise.

The calculated doses are 50-year committed doses from 1 year of intake.

J.5.4.2 Human Health Impacts

Potential radiological impacts on the public and workers resulting from normal lead assembly operations are presented in Section 4.27.4.4.

J.5.5 SRS

J.5.5.1 Assessment Data

This section presents applicable data and assumptions used in the assessment of lead assembly human health risks at SRS. Appendix F.10 provides a summary of the methods and tools (e.g., the GENII computer code) used for the assessment.

J.5.5.1.1 Meteorological Data

The meteorological data used for the SRS dose assessments was in the form of a JFD file. A JFD file is a table listing the percentages of time the wind blows in a certain direction, at a certain speed, and within a certain

stability class. The JFD file was based on measurements taken over a period of several years at a specific location (H-Area) and height. Average annual meteorological conditions, averaged over the measurement period, were used for normal operations. Table J–77 presents the JFD used in the dose assessments for SRS.

J.5.5.1.2 Population Data

The SRS population distribution was based on the *1990 Census of Population and Housing Data* (DOC 1992). Projections were determined for the year 2005 for areas within 80 km (50 mi) of the proposed facility location. The site population in 2005 was assumed to be representative of the population over the operational period evaluated. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances out to an 80-km (50-mi) distance. The grid was centered within H-Area, the location from which radionuclides are assumed to be released during incident-free operations. Table J–78 presents the population data used for the lead assembly dose assessments at SRS.

J.5.5.1.3 Agricultural Data

The 1987 Census of Agriculture was the source used to generate site-specific data for food production. Food production was spatially distributed on a circular grid similar to that used for the population distributions described previously. This food grid (or wheel) was generated by combining the fraction of a county in each segment (e.g., south, southwest, north-northeast) and the county production of the eight food categories analyzed by GENII—leafy vegetables, root vegetables, fruits, grains, beef, poultry, milk, and eggs. Each county's food production was assumed to be distributed uniformly over the given county's land area. These categorized food wheels were then used in the assessment of doses to the population from the ingestion pathway. The consumption rates used in the dose assessments were those for the MEI and average exposed individual. People living within the 80-km (50-mi) assessment area were assumed to consume only food grown in that area. SRS food production and consumption data used for the dose assessments in the SPD EIS were obtained from the *Health Risk Data for Storage and Disposition of Final PEIS* (HNUS 1996).

Table J–77. SRS 1987–1991 Joint Frequency Distributions at 61-m Height

Wind Speed (m/s)	Stability Class	Wind Blows Toward															
		S	SSW	SW	WSW	W	WNW	NW	NNW	N	NNE	NE	ENE	E	ESE	SE	SSE
2.0	A	0.37	0.41	0.37	0.42	0.4	0.37	0.4	0.36	0.36	0.35	0.45	0.39	0.45	0.43	0.37	0.41
	B	0.08	0.08	0.09	0.1	0.05	0.06	0.06	0.05	0.08	0.07	0.05	0.05	0.05	0.08	0.05	0.07
	C	0.03	0.06	0.09	0.07	0.06	0.05	0.06	0.05	0.07	0.05	0.06	0.05	0.08	0.05	0.05	0.05
	D	0.02	0.05	0.06	0.04	0.06	0.03	0.06	0.07	0.06	0.03	0.07	0.05	0.04	0.03	0.05	0.04
	E	0.01	0.02	0.04	0.01	0.01	0.03	0.03	0.03	0.02	0.02	0.01	0.01	0.02	0.01	0.02	0.02
	F	0	0.01	0.01	0.01	0.01	0.01	0	0.01	0.01	0.01	0.02	0.01	0.01	0.01	0.01	0.01
4.0	A	0.87	0.74	0.88	1	0.94	0.94	0.65	0.62	0.74	0.72	1	1.28	1.29	0.94	0.53	0.6
	B	0.27	0.41	0.58	0.62	0.43	0.34	0.24	0.22	0.32	0.33	0.48	0.67	0.56	0.37	0.25	0.21
	C	0.17	0.57	1.13	1.03	0.6	0.41	0.41	0.37	0.48	0.52	0.59	0.79	0.53	0.45	0.3	0.24
	D	0.1	0.44	1.07	0.89	0.55	0.5	0.71	0.69	0.92	0.91	0.8	0.81	0.72	0.57	0.43	0.27
	E	0.06	0.27	0.69	0.48	0.3	0.33	0.46	0.7	0.67	0.57	0.54	0.47	0.43	0.43	0.33	0.3
	F	0.02	0.05	0.09	0.04	0.02	0.08	0.09	0.09	0.11	0.08	0.12	0.09	0.03	0.05	0.05	0.07
6.0	A	0.57	0.26	0.16	0.19	0.15	0.07	0.07	0.09	0.14	0.14	0.21	0.24	0.27	0.24	0.14	0.24
	B	0.14	0.39	0.38	0.31	0.16	0.11	0.07	0.08	0.19	0.21	0.32	0.51	0.51	0.36	0.13	0.09
	C	0.12	0.54	1.3	0.74	0.35	0.19	0.22	0.25	0.47	0.46	0.56	0.69	0.64	0.56	0.21	0.12
	D	0.12	0.43	0.85	0.58	0.4	0.44	0.65	1.16	1.45	0.78	0.9	0.77	0.78	0.65	0.32	0.09
	E	0.07	0.53	0.69	0.71	0.6	0.45	0.65	1.01	1.18	0.94	0.91	0.89	0.48	0.4	0.19	0.14
	F	0.01	0.26	0.21	0.14	0.14	0.19	0.13	0.16	0.22	0.21	0.24	0.23	0.07	0.04	0.02	0.04
8.0	A	0.09	0.05	0.01	0.01	0.01	0	0.01	0.01	0.02	0.02	0.02	0.04	0.03	0.02	0.01	0.06
	B	0.01	0.08	0.03	0.01	0.01	0.01	0	0.01	0.05	0.04	0.05	0.1	0.17	0.21	0.06	0.01
	C	0.01	0.1	0.2	0.08	0.02	0.03	0.03	0.06	0.16	0.16	0.21	0.26	0.45	0.43	0.1	0.02
	D	0.01	0.05	0.1	0.02	0.01	0.01	0.05	0.18	0.22	0.15	0.1	0.09	0.03	0.05	0.03	0
	E	0	0.05	0.03	0.04	0.01	0.01	0	0.03	0.04	0.02	0.04	0.01	0.01	0	0	0
	F	0	0.03	0.02	0.02	0	0.01	0	0.01	0.02	0.01	0.02	0.01	0	0	0	0
12.0	A	0.01	0	0	0	0	0	0	0	0	0.01	0.01	0.01	0	0.01	0	0.01
	B	0	0.01	0	0	0	0	0	0	0	0	0.01	0.01	0.06	0.06	0.01	0
	C	0	0.01	0	0	0	0.01	0	0.03	0.04	0.04	0.05	0.06	0.16	0.17	0.02	0.01
	D	0	0.02	0.02	0	0	0	0	0.01	0.02	0.04	0	0	0.01	0	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
14.1	A	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	B	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	C	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	D	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	E	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
	F	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Source: Simpkins 1997.

Table J–78. Projected SRS Population Surrounding H-Area for Year 2005

Direction	Distance (mi)										Total
	0–1	1–2	2–3	3–4	4–5	5–10	10–20	20–30	30–40	40–50	
S	0	0	0	0	0	0	485	1,807	5,207	3,545	**11,044**
SSW	0	0	0	0	0	0	629	1,906	5,070	2,361	**9,966**
SW	0	0	0	0	0	25	895	7,586	1,939	2,953	**13,398**
WSW	0	0	0	0	0	71	2,428	4,529	3,330	8,327	**18,685**
W	0	0	0	0	0	683	4,586	54,394	22,338	13,086	**95,087**
WNW	0	0	0	0	0	1,384	7,849	172,996	76,767	6,917	**265,913**
NW	0	0	0	0	0	1,026	14,508	34,759	4,044	3,629	**57,966**
NNW	0	0	0	0	0	2,691	30,598	23,544	8,243	6,184	**71,260**
N	0	0	0	0	0	363	4,049	3,790	4,887	20,832	**33,921**
NNE	0	0	0	0	0	89	1,790	3,016	6,535	21,457	**32,887**
NE	0	0	0	0	0	15	3,754	3,684	6,147	9,896	**23,496**
ENE	0	0	0	0	0	9	3,723	6,246	6,956	43,139	**60,073**
E	0	0	0	0	0	113	7,647	3,844	6,830	4,084	**22,518**
ESE	0	0	0	0	0	3	1,329	2,551	3,551	5,933	**13,367**
SE	0	0	0	0	0	0	552	4,950	4,962	8,342	**18,806**
SSE	0	0	0	0	0	0	374	597	1,940	2,703	**5,614**
Total	**0**	**0**	**0**	**0**	**0**	**6,472**	**85,196**	**330,199**	**168,746**	**163,388**	**754,001**

Source: DOC 1992.

J.5.5.1.4 Source Term Data

Estimated incident-free radiological releases associated with the MOX fuel lead assembly facility are presented in Table J–79. Stack height and release location are provided in the ORNL *SRS MOX Fuel Lead Assemblies Data Report for the Surplus Plutonium Disposition Environmental Impact Statement* (O'Connor et al. 1998e).

Table J–79. Estimated Incident-Free Annual Radiological Releases From the MOX Lead Assembly Facility at SRS

Isotope	(μCi/yr)
Plutonium 236	–
Plutonium 238	0.85
Plutonium 239	23
Plutonium 240	5.3
Plutonium 241	58
Plutonium 242	9.3×10^{-4}
Americium 241	2.0
Uranium 234	1.3×10^{-3}
Uranium 235	5.4×10^{-5}
Uranium 238	3.1×10^{-3}

Source: O'Connor et al. 1998e.

J.5.5.1.5 Other Calculational Assumptions

To estimate radiological impacts of incident-free operation of the facilities at SRS, the following additional assumptions and factors were considered, in accordance with the guidelines established in NRC Regulatory Guide 1.109 (NRC 1977).

Ground surfaces were assumed to have no previous deposition of radionuclides for the purposes of modeling the incremental radiological impacts associated with surplus plutonium disposition activities. However, doses associated with true instances of prior deposition are accounted for in the Affected Environment and Cumulative Impacts sections.

The annual external exposure time to the plume and to soil contamination was 0.7 year for the MEI (NRC 1977).

The annual external exposure time to the plume and to soil contamination was 0.5 year for the population (NRC 1977).

The annual inhalation exposure time to the plume was 1 year for the MEI and general population (NRC 1977).

The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of the adult human.

A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of contaminated animal products. Drinking water, aquatic food ingestion, and any other pathway that may involve liquid exposure were not examined because all releases are to the air.

Reported stack heights were used for atmospheric releases and were assumed to be the effective stack height. The resultant doses were conservative because use of the actual stack height negates plume rise.

The calculated doses are 50-year committed doses from 1 year of intake.

J.5.5.2 Human Health Impacts

Potential radiological impacts on the public and workers resulting from normal lead assembly operations are presented in Section 4.27.5.4.

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Appendix K
Facility Accidents

K.1 IMPACT ASSESSMENT METHODS FOR FACILITY ACCIDENTS

K.1.1 Introduction

The potential for facility accidents and the magnitude of their consequences are important factors for making reasonable choices among the various surplus plutonium disposition alternatives analyzed in the *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS). Guidance on the implementation of 40 CFR 1502.22, as amended (EPA 1992), requires the evaluation of impacts that have a low frequency of occurrence but high consequences. Further, public comments received during the scoping process have clearly indicated the public's concern with facility safety and health risks and the need to address these concerns in the decisionmaking process.

For the No Action Alternative, potential accidents are defined in existing facility documentation, such as safety analysis reports (SARs), hazards assessment documents, National Environmental Policy Act (NEPA) documents, and probabilistic risk assessments (PRAs). The accidents include radiological and chemical accidents that have a low frequency of occurrence but high consequences, and a spectrum of other accidents that have a higher frequency of occurrence and lesser consequences. The data in these documents include accident scenarios, materials at risk, source terms (quantities of hazardous materials released to the environment), and consequences.

For each facility, a hazards analysis document identifying and estimating the effects of all major hazards that could affect the environment, workers, and the public would be issued in conjunction with the conceptual design package. Additional accident analyses for identified major hazards would be provided in a preliminary SAR issued during the period of definitive design (Title II) review. A final SAR would be prepared during the construction period and issued before testing began as final documented evidence that the new facility could be operated in a manner that did not pose any undue risk to the health and safety of workers and the public.

In determining the potential for facility accidents and the magnitude of their consequences, the SPD EIS considers two important concepts in the presentation of results: (1) risk and (2) uncertainties and conservatism.

K.1.1.1 Risk

One type of metric that can be obtained from the accident analysis results presented in the SPD EIS is accident risk. Risk is usually defined as the product of the consequences and estimated frequency of a given accident. Accident consequences may be presented in terms of dose (e.g., person-rem) or health effects (e.g., latent cancer fatalities [LCFs]). The accident frequency is the number of times the accident is expected to occur over a given period of time (e.g., per year). In general, the frequency of design basis and beyond-design-basis accidents is much lower than 1 per year, and therefore is approximately equal to the probability of the accident during 1 year. If an accident is expected to occur once every 1,000 years (i.e., a frequency of 1.0×10^{-3} per year) and the consequences of the accident is five LCFs, then the risk is $1.0\times10^{-3}\times5 = 5.0\times10^{-3}$ LCF per year.

A number of specific types of risk can be directly calculated from the Melcor Accident Consequence Code System (MACCS2) results reported in the SPD EIS (SNL 1997). One type of risk, average individual risk, is the product of the total consequences experienced by the population and the accident frequency, divided by the population.[1]
For example, if an accident has a frequency of 1.0×10^{-3} per year, the consequence thereof is 5 LCFs, and the

[1] Population data for each facility considered in the SPD EIS can be found in Appendix J.

population in which the fatalities are experienced is 100,000, then the average individual risk is $1.0\times10^{-3} \times 5/100{,}000 = 5.0\times10^{-8}$ LCF per year. This metric is meaningful only when the mean value for consequence is used because risk itself is not a random parameter, even though it involves underlying randomness. It is noteworthy that the value of the average individual risk depends on the size of the area for which the population is defined. In general, the larger the area considered, the smaller the average individual risk for a given accident. The choice of an 80-km (50-mi) radius is common practice.

The average individual risk is a measure of the risk that an average individual (in this case within 80 km [50 mi] of the accident) experiences from specified accidents at the facility. This risk can be compared with other average individual risks, such as the risk of dying from a motor vehicle accident (about 1 in 80), the risk of death from fires (about 1 in 500), or the risk of accidental poisoning (about 1 in 1,000). These comparisons are not meant to imply that risks of an LCF caused by U.S. Department of Energy (DOE) operations are trivial, but only to show how they compare with other, more common risks. Radiological risks to the general public from DOE operations are considered to be involuntary risks as opposed to voluntary risks, such as operating a motor vehicle.

It is also possible to calculate population risk, which is the product of the total consequences experienced by the population and accident frequency. For example, if an accident has a frequency of 1.0×10^{-3} per year and the consequences of the accident is 5 LCFs, then the population risk is $1.0\times10^{-3}\times5 = 5.0\times10^{-3}$ LCF per year. Population risk is a measure of the expected number of consequences experienced by the population as a whole over the course of a year.

It would be inappropriate, however, to simply take the LCFs given the dose at 1,000 m (3,281 ft) or the LCFs given the dose at the site boundary and multiply them by the corresponding accident frequencies in an attempt to obtain the maximum individual risk to the noninvolved worker or the maximally exposed individual (MEI) member of the public. The reasons for this are discussed in the following paragraphs.

The distribution of centerline consequences from which the reported doses are obtained is constructed by modeling the accidental release many times using different weather conditions (i.e., windspeed, wind direction, stability class, and rainfall) each time. For each weather condition, the centerline consequences at 1,000 m (3,281 ft) and at the site boundary are calculated, and those values contribute to their respective distributions. Thus, given the accidental release, there is a 95 percent chance that the centerline consequences at 1,000 m (3,281 ft) and at the site boundary will fall below the reported 95th percentile consequences, and the expected consequences would be equal to the reported mean consequences. It is noteworthy, however, that the actual locations of the centerline consequences vary with wind direction, so the reported consequences are not associated with a specific point at 1,000 m (3,281 ft) or the site boundary. It is known only that the centerline consequences, wherever they might be, are characterized by the reported values.

A problem arises when these consequences are used to characterize individual risk. Although there is always some location that is exposed to the centerline consequences, no location is associated with the risk obtained by multiplying the centerline consequences by the accident frequency, because the direction of the plume centerline changes for each set of weather conditions. As a result, the risk to an individual at the location of maximum risk is likely to be much lower than the risk calculated by multiplying the centerline consequences by the accident frequency. In fact, because there are 16 sectors, and because doses decrease with lateral movement away from the centerline even within a sector, risk values generated in this way would tend to overstate the risk by a factor of as much as 100, and possibly more. The values are bounding, but have a potentially misleading degree of conservatism. Ultimately, MACCS2 is capable of calculating individual consequences at the point of maximum consequence (as reported in the SPD EIS), but it is not configured to calculate individual risk at the point of maximum risk.

K.1.1.2 Uncertainties and Conservatism

The analyses of accidents are based on calculations relevant to hypothetical sequences of events and models of their effects. The models provide estimates of the frequencies, source terms, pathways for dispersion, exposures, and the effects on human health and the environment that are as realistic as possible within the scope of the analysis. In many cases, a paucity of experience with the accidents postulated leads to uncertainty in the calculation of their consequences and frequencies. This fact has prompted the use of models or input values that yield conservative estimates of consequence and frequency. All alternatives have been evaluated using uniform methods and data, allowing for a fair comparison of all alternatives.

Although average individual and population risks can be calculated from the information in the SPD EIS, the equations for such calculations involve accident frequency, a parameter whose calculation is subject to considerable uncertainty. The uncertainty in estimates of the frequency of highly unlikely events can be several orders of magnitude. This is the reason accident frequencies are reported in the SPD EIS qualitatively, in terms of broad frequency bins, as opposed to numerically. Similarly, any metric that includes frequency as a factor will have at least as much, and generally more, uncertainty associated with it. Therefore, the consequence metrics have been preserved as the primary accident analysis results, and accident frequencies identified qualitatively, to provide a perspective on risk that does not imply an unjustified level of precision.

K.1.2 Safety Design Process

The proposed surplus plutonium disposition facilities would be designed to comply with current Federal, State, and local laws, DOE orders, and industrial codes and standards. This would result in a plant that is highly resistant to the effects of natural phenomena, including earthquake, flood, tornado, and high wind, as well as credible events as appropriate to the site, such as fire, explosions, and man-made threats.

The design process for the proposed facilities would comply with the requirements for safety analysis and evaluation in DOE Orders 430.1 and 5480.23. These orders require that the safety assessment be an integral part of the design process to ensure compliance with all DOE construction and operation safety criteria by the time the facilities are constructed and in operation.

The safety analysis process begins early in conceptual design with the identification of hazards that could produce unintended adverse safety consequences to workers or the public. As the design develops, failure modes and effects analyses (FMEAs) are performed to identify events capable of releasing hazardous material. The kinds of events considered include equipment failures, spills, human errors, fires, explosions, criticality, earthquakes, electrical storms, tornadoes, floods, and aircraft crashes. These postulated events become focal points for design changes or improvements to prevent unacceptable accidents. The analyses continue as the design progresses, the object being to assess the need for safety equipment and the performance of such equipment. Eventually, the safety analyses are formally documented in a SAR and, if appropriate, a PRA. The PRA documents the estimated frequency and consequences of a complete spectrum of accidents and helps to identify where design improvements could make meaningful safety improvements.

The first SAR, completed at the conclusion of conceptual design, includes identification of hazards and some limited assessment of a few enveloping design basis accidents. It includes deterministic safety analysis and FMEA of major systems. A comprehensive preliminary SAR, completed by the end of the preliminary design, provides a broad assessment of the range of design basis accident scenarios and the performance of equipment provided in the facility specifically for accident consequence mitigation. A limited PRA may be included in that analysis.

The SAR continues to be developed during detailed design. The safety review of the report and any supporting PRA are completed and safety issues resolved before the initiation of facility construction. Also, a final SAR is produced that includes documentation of safety-related design changes made during construction and the impact of those changes on the safety assessment. It also includes the results of any safety-related research and development that was performed to support the safety assessment of the facility. Approval of the final SAR is required before the facility is allowed to commence operation.

K.1.3 DOE Facility Accident Identification and Quantification

K.1.3.1 Background

Identification of accident scenarios for the proposed facilities is fairly straightforward. The proposed facilities are simple, and their processes have been used in other facilities for other purposes. From an accident identification and quantification perspective, therefore, these processes are well known and understood. Very few of the proposed activities would differ from activities at other facilities.

New facilities would likely be designed, constructed, and operated to provide an even lower accident risk than other facilities that have used these types of processes. The new facilities would benefit from lessons learned in the operation of similar processes. They would be designed to surpass existing plutonium facilities in the ability to reduce the frequency of accidents and to mitigate the consequences thereof.

A large experience base exists for the design of the proposed facilities and processes. Because the principal hazard to workers and the public from plutonium is the inhalation of very small particles, the safety management approach that has evolved is centered on control of those particles. The control approach is to perform all operations that could release airborne plutonium particles in a glovebox. The glovebox protects workers from inhalation of the particles and provides a convenient means for the collection of any particle that becomes airborne on filters. Air from the gloveboxes, operating areas, and buildings is exhausted through multiple stages of high-efficiency particulate air (HEPA) filters and monitored for radioactivity prior to release from the building. These exhaust systems are designed for effective performance even under the severe conditions of design basis accidents, such as major fires involving an entire process line.

While the new processes and facilities would be designed to reduce the risks of a wide range of possible accidents to a level deemed acceptable, some such risks would remain. As with all engineered structures—e.g., houses, bridges, dams—there is some level of earthquake or high wind the structure could not survive. While new plutonium facilities must be designed to very high standards—for instance, they must survive, with little plutonium release, a 1-in-10,000-year earthquake—an accident more severe than the design basis can always be postulated. Current DOE standards require that new facilities be designed to prevent to the extent possible, and then withstand, control, and mitigate, all credible process-related accidents. For safety analysis purposes, credible accidents are generally defined as accidents with frequencies greater than 1 in 1 million per year, including such natural-phenomena-induced accidents as earthquakes, high winds, and flooding. The accidents considered in the design, construction, and operation of these facilities are generally called design basis accidents.

In addition to the accident risks from the design basis accidents, the new facilities would face risks from beyond-design-basis accidents. For most plutonium facilities, the design basis includes all types of process-related accidents that have occurred in past operations: major spills, leaks, transfer errors, process-related fires, explosions, and nuclear criticalities. Certain natural-phenomena-initiated accidents also meet the DOE design basis criteria. While extremely unlikely, all new plutonium facilities, as essentially all manmade structures, could collapse under the influence of an earthquake. For most new plutonium facilities, the worst possible accident is a beyond-design-basis earthquake that results in partial or total collapse of the structure, spills, possibly fires, and loss of confinement of the plutonium powder. Also conceivable are such external events

as the crash of a large aircraft onto the structure with an ensuing fuel-fed fire. At most locations away from major airports, however, the likelihood is less than 1 in 10 million per year. For some locations, such as Pantex, the frequency is higher, so aircraft crash–initiated accidents are a basic consideration.

The accident analysis reported in the SPD EIS is less detailed than a formal PRA or facility safety analysis because it addresses bounding accidents (accidents with low frequency of occurrence and high consequence) and a representative spectrum of possible operational accidents (accidents with high frequency of occurrence and low consequence). The technical approach for the selection of accidents is consistent with the DOE Office of NEPA Oversight's *Recommendations for the Preparation of Environmental Assessments and Environmental Impact Statements* (DOE 1993), which recommends consideration of two major categories of accidents: design basis accidents and beyond-design-basis accidents.[2]

K.1.3.2 Identification of Accident Scenarios and Frequencies

A range of design basis and beyond-design-basis accident scenarios have been identified for each of the surplus plutonium disposition technologies (UC 1998a–h, 1999a–d). For each technology, the wide range of process-related accidents possible during construction and operation of the facility have been evaluated to ensure that their consequences are low or the frequency of occurrence, extremely low.

All of the analyzed accidents would involve a release of small, respirable plutonium particles or direct gamma and neutron radiation, and to a lesser extent, fission products from a nuclear criticality. Analyses of each proposed operation for accidents involving hazardous chemicals are reflected in the data reports supporting the SPD EIS. However, as the quantities of hazardous chemicals to be handled are small relative to those of many industrial facilities, no major chemical accidents were identified. The general categories of process-related accidents considered include:

- Drops or spills of materials within and outside the gloveboxes
- Fires involving process equipment or materials, and room or building fires
- Explosions initiated by the process equipment or materials or by conditions or events external to the process
- Nuclear criticalities

The analyses considered synergistic effects and determined that the only significant source of such effects would be a seismic event (i.e., a design basis seismic event or a seismically induced total collapse). The synergy would be due to the common-cause initiator (i.e., seismic ground motion). This was accounted for by summing population doses and LCFs for alternatives in which facilities would be located at the same site. MEI doses were not summed because an individual would only receive a summed dose if he or she were located along the line connecting the release points from two facilities and the wind were blowing along the same line at the time of the accident.

For each of these accident categories, a conservative preliminary assessment of consequence was made, and where consequences were significant, one or more bounding accident scenarios were postulated. The building confinement and fire suppression systems would be adequate to reduce the risks of most spills and minor fires. The systems would be designed to prevent, to the extent practicable, larger fires and explosions. Great efforts have always been made to prevent nuclear criticalities, which have the potential to kill workers in their immediate

[2] Some of the data reports supporting the SPD EIS use the terms "evaluation basis" and "beyond-evaluation-basis" to denote the two major categories of accidents. For clarity, the SPD EIS uses the terms "design basis" and "beyond-design-basis" throughout.

vicinity. In all cases, standard practice is expected to keep the frequency of accidental nuclear criticalities as low as possible.

The proposed surplus plutonium disposition facilities would be expected to meet or exceed the requirements of DOE Order 420.1, *Facility Safety,* and *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities* (DOE-STD-1020-94) (DOE 1994a), or the requirements of 10 CFR 70, *Domestic Licensing of Special Nuclear Material,* if the proposed facility were to be licensed by the U.S. Nuclear Regulatory Commission (NRC). Because the DOE and, if applicable, NRC design criteria require that new plutonium-processing buildings be of very robust, reinforced-concrete construction, very few events outside the building would have sufficient energy to threaten the building confinement. The principal concern would be the crash of a large commercial or military aircraft into the facility. Such an event, however, is highly unlikely. Only those crashes with a frequency greater than 10^{-7} per year are addressed in the SPD EIS.

Design basis and beyond-design-basis natural-phenomena-initiated accidents are also considered. Because of the robust nature of construction of new plutonium facilities, the only design basis natural-phenomena-initiated accidents with the potential to impact the facility interior are seismic events. Similarly, seismic events also bound the consequences and risks posed by beyond-design-basis natural phenomena.

The suite of generic accidents in the *Storage and Disposition PEIS* (DOE 1996a) was considered in the analysis of accidents for the SPD EIS. However, the more detailed design information in the surplus plutonium disposition data reports was the primary basis for the identification of accidents because it most accurately represents the expected facility configuration. The fire on the loading dock and the oxyacetylene explosion in a process cell were unsupported by this information, so were not included in the SPD EIS.

Accident frequencies are generally grouped into the bins of "anticipated," "unlikely," and "extremely unlikely," with estimated frequencies of greater than 10^{-2}, 10^{-2} to 10^{-4}, and 10^{-4} to 10^{-6} per year, respectively. The accidents evaluated represent a spectrum of accident frequencies and consequences ranging from low-frequency/high-consequence to high-frequency/low-consequence events. However, given the preliminary nature of the designs under consideration, it was not possible to assess quantitatively the frequency of occurrence of all the events addressed. The evaluation does not indicate the total risk of operating the facility, but does provide information on high-risk events that could be used to develop an accident risk ranking of the various alternatives.

K.1.3.3 Identification of Material at Risk

For each accident scenario, the material at risk—generally plutonium—was identified. Plutonium to be disposed of has a wide range of chemical and isotopic forms. The sources of plutonium vary among the various candidate facilities, and for specific facilities among various alternatives. Table K–1 presents the isotopic compositions that were used in the development of accident consequences in the SPD EIS. The vulnerability of material generally depends on the form of that material, the degree and robustness of containment, and the energetics of the potential accident scenario (UC 1998a:table 6-6; 1998c:tables 9-2 and A-7; 1998d:table B-1). For example, plutonium stored in strong, tight storage containers is not generally vulnerable to simple drops or spills, but may be vulnerable in a total collapse earthquake scenario.

Table K–1. Isotopic Composition of Plutonium Used in Accident Analysis (wt %)

Isotope	Pit Disassembly and MOX	Immobilization: Plutonium Conversion	Immobilization: First Stage, Hybrid Case	Immobilization: First Stage, 50-t Case
Plutonium 238	3.00×10^{-2}	0.0	0.0	2.0×10^{-2}
Plutonium 239	92.2	86.9	86.9	91.0
Plutonium 240	6.46	11.1	11.1	8.2
Plutonium 241	5.00×10^{-2}	1.5	1.5	5.80×10^{-1}
Plutonium 242	1.00×10^{-1}	5.0×10^{-1}	5.0×10^{-1}	2.50×10^{-1}
Americium 241	9.00×10^{-1}	1.0	1.0	9.4×10^{-1}

On an industrial scale, the quantities of hazardous chemicals are generally small. The occupational risks are generally limited to material handling and are managed under the required industrial hygiene program. No substantial hazardous chemical releases are expected.

K.1.3.4 Identification of Material Potentially Released to the Environment

The amount and particle size distribution of material aerosolized in an accident generally depends on the form of that material, the degree and robustness of containment, and the energetics of the potential accident scenario. Once the material is aerosolized, it must still travel through building confinement and filtration systems or bypass the systems before being released to the environment.

A standard DOE formula was used to estimate the source term for each accident at each of the proposed surplus plutonium facilities:

$$\text{Source Term} = \text{MAR} \times \text{DR} \times \text{ARF} \times \text{RF} \times \text{LPF}$$

where:

MAR = material at risk (curies or grams)
DR = damage ratio
ARF = airborne release fraction
RF = respirable fraction[3]
LPF = leak path factor

The value of each of these factors depends on the details of the specific accident scenario postulated. ARF and RF were estimated according to reference material in *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE-HDBK-3010-94) (DOE 1994b). Conservative HEPA filter efficiencies of 0.999 and 0.99 were assumed, based on two stages of filtration, for a total LPF of 1.0×10^{-5}; however, actual efficiencies would likely be 0.999 and 0.998 or better. [Text deleted.]

No accident scenarios were identified that would result in a substantial release of plutonium or other radionuclides via liquid pathways.

[3] Respirable fractions are not applied in the assessment of doses based on noninhalation pathways, such as criticality.

K.1.4 Evaluation of Consequences of Accidents

K.1.4.1 Potential Receptors

For each potential accident, information is provided on accident consequences and frequencies to three types of receptors: (1) a noninvolved worker, (2) the maximally exposed member of the public, and (3) the offsite population. The first receptor, a noninvolved worker, is a hypothetical individual working on the site but not involved in the proposed activity. The worker is assumed to be downwind at a point 1,000 m (3,281 ft) from the accident. Although other distances closer to the accident could have been assumed, the calculations break down at distances of about 200 m (656 ft) or less due to limitations in modeling the effects of building wake and local terrain on dispersion of the released radioactive substances. A worker closer than 1,000 m (3,281 ft) to the accident would generally receive a higher dose; a worker farther away, a lower dose. At some sites where the distance from the accident to the nearest site boundary is less than 1,000 m (3,281 ft), the worker is assumed to be at the site boundary. The second receptor, a maximally exposed member of the public, is a hypothetical individual assumed to be downwind at the site boundary. Exposures received by this individual are intended to represent the highest doses to a member of the public. The third receptor, the offsite population, is all members of the public within 80 km (50 mi) of the accident location.

Consequences to workers directly involved in the processes under consideration are addressed generically, without attempt at a scenario-specific quantification of consequences. This approach to in-facility consequences was selected for two reasons. First, the uncertainties involved in quantifying accident consequences become overwhelming for most radiological accidents due to the high sensitivity of dose values to assumptions about the details of the release and the location and behavior of the impacted worker. Also, the dominant accident risks to the worker of facility operations are from standard industrial accidents, as opposed to bounding radiological accidents. The accident fatality risk for DOE has been reported as 2.7×10^{-5} per person per year (DOE 1999a). According to historical data on standard industrial accidents, the national average fatality risk from manufacturing operations is 3.5×10^{-5} per person per year (DOL 1997).

Consequences for potential receptors as a result of plume passage were determined without regard for emergency response measures, and thus are more conservative than would be expected if evacuation and sheltering were explicitly modeled. Instead, it is assumed that potential receptors are fully exposed in fixed positions for the duration of plume passage, thereby maximizing their exposure to the plume. As discussed in Appendix K.1.4.2, a conservative estimate of total risk was obtained by assuming that all released radionuclides contributed to the inhalation dose rather than being removed from the plume by surface deposition, which is a less significant contributor to overall risk and is controllable through interdiction.

K.1.4.2 Modeling of Dispersion of Releases to the Environment

The MACCS2 computer code (version 1.12) was used to estimate the consequences of accidents for the proposed facilities. A detailed description of the MACCS2 model is available in NUREG/CR-4691 (NRC 1990). Originally developed to model the radiological consequences of nuclear reactor accidents, this code has been used for the analysis of accidents for many EISs and other safety documentation, and is considered applicable to the analysis of accidents associated with the disposition of plutonium.

MACCS2 models the offsite consequences of an accident that releases a plume of radioactive materials into the atmosphere, specifically, the degree of dispersion versus distance as a function of historical wind direction, speed, and atmospheric conditions. Were such an accidental release to occur, the radioactive gases and aerosols in the plume would be transported by the prevailing wind and dispersed in the atmosphere, and the population would be exposed to radiation. MACCS2 generates the distribution of downwind doses at specified distances, as well as the distribution of population doses out to 80 km (50 mi).

As implemented, the MACCS2 model evaluates doses due to inhalation of aerosols, such as respirable plutonium, as well as exposure to the passing plume. This represents the major portion of the dose that a noninvolved worker or member of the public would receive as a result of a plutonium disposition facility accident. The longer-term effects of plutonium deposited on the ground and surface waters after the accident, including the resuspension and inhalation of plutonium and the ingestion of contaminated crops, were not modeled for the SPD EIS. These pathways have been studied and been found not to contribute as significantly to dosage as inhalation, and they are controllable through interdiction. Instead, the deposition velocity of the radioactive material was set to zero, so that material that might otherwise be deposited on surfaces remained airborne and available for inhalation. This adds a conservatism to inhalation doses that can become considerable at large distances (as much as two orders of magnitude at the 80-km [50-mi] limit). Thus, the method used in the SPD EIS is conservative compared with dose results that would be obtained if deposition and resuspension were taken into account.

Longer-term effects of fission products released in a nuclear criticality accident have been extensively studied. The principal concern is ingestion of iodine 131 via milk that becomes contaminated due to the ingestion of contaminated grains by milk cows. This pathway can be controlled if necessary. In terms of the effects of an accidental criticality, doses from this pathway are small.

The potential for tritium contamination of the Ogallala aquifer as a consequence of an accident at Pantex involving tritium was identified as a specific concern during the development of the SPD EIS. The assessment of consequences of accidental tritium releases in the SPD EIS is consistent with the method used in the *Final Programmatic Environmental Impact Statement for Tritium Supply and Recycling* (DOE 1995a). Unlike plutonium, oxidized tritium (i.e., water vapor) is not significantly deposited on the ground for subsequent percolation into the local groundwater except under conditions of rain or dew. Pantex has a rather arid climate, so the chance of these weather conditions at the time of an accident is slight. Moreover, even if it were to happen as indicated in Section 4.6.1.2 of the *Final Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapon Components* (DOE 1996b), actual movement of contaminated groundwater off the site would require about 10 to 20 years. In fact, current test data show that it could take as long as 50 or more years for a contaminant plume to move off the site. The half-life of tritium is 12 years; therefore, any hypothetical contamination deposited on the ground surface and carried into the groundwater regime would be reduced by a factor of roughly 2 to 16 by the time it moved off the site. Because of these considerations, health consequences of contamination of the Ogallala aquifer were not considered to be a significant contributor to health risks from a tritium release accident.

The region around the facility is divided by a polar-coordinate grid centered on the facility itself. The user specifies the number of radial divisions and their endpoint distances. The angular divisions used to define the spatial grid correspond to the 16 directions of the compass.

MACCS2 was applied in a probabilistic manner using a weather bin–sampling technique. Centerline doses, as a function of distance, were calculated for each of 1,460 meteorological sequence samples, resulting in a distribution of doses reflecting variations in weather conditions at the time of the postulated accidental release. The code outputs the conditional probability of exceeding a dose as a function of distance. The mean and 95th percentile consequences are reported in the SPD EIS. Doses higher than the 95th percentile values would be expected only 5 percent of the time.

MACCS2 cannot be used to calculate directly the distribution of maximum doses (resulting from meteorological variations) around irregular contours, such as a site boundary. As a result, analyses that use MACCS2 to calculate site boundary doses usually default to calculating doses at the distance corresponding to the shortest distance to the site boundary. In effect, the site boundary is treated as if it were circular, with a radius equal to the shortest distance from the facility to the actual site boundary. While this approximation is conservative with respect to dose (with the possible exception of doses from elevated plumes), it eliminates the use of some

site-specific information, namely the site boundary location (other than the nearest point), wind direction, and any correlation between wind direction and other meteorological parameters. Because the primary purpose of the SPD EIS is to aid in decisions about facility locations, and because differences in dose values among the various options are largely a function of site-specific variations, a different approach was taken to more accurately characterize the potential for maximum doses at the site boundary.

For the SPD EIS, MACCS2 was used to generate intermediate results that could be further processed to obtain the distribution of doses around the site boundary, accounting for variations in site boundary distance as a function of direction. The specific instrument was the Type B result option of MACCS2, which renders the distribution of doses at a specified radial distance within a specified compass sector, given a release. Type B results were requested for the site boundary distance for each of the 16 compass sectors over which the meteorological data is defined. This resulted in 16 separate dose distributions; one for each specific location around the site boundary. The distribution of maximum doses around the site boundary was constructed by first summing the values of the Type B distributions for each dose value. The resulting distribution was then truncated for low dose values to the point where the remainder of the distribution was normalized. This produced the distribution of maximum doses around the site boundary, which is the distribution from which the mean and 95th percentile doses are reported.

Radiological consequences may vary somewhat as a result of variations in the duration of release. For longer releases, there is a greater chance of plume meander (i.e., variations in wind direction over the duration of release). MACCS2 models plume meander by increasing the lateral dispersion coefficient of the plume for longer release durations, thus lowering the dose. For perspective, doses from an homogenous, 1-hr release would be 30 percent lower than those of a 10-min release as a result of plume meander; doses from a 2-hr release, 46 percent lower. The other effect of longer release durations is involvement of a greater variety of meteorological conditions in a given release, which reduces the variance of the resulting dose distributions. This would tend to lower high-percentile doses, raise low-percentile doses, and have no effect on the mean dose.

For the SPD EIS accident analysis, a duration of 10 min was assumed for all releases. This is consistent with the accident phenomenology expected for all scenarios, with the possible exception of fire. Depending on the circumstances, the time between fire ignition and extinction may be considerably longer, particularly for the larger, beyond-design-basis fires. However, even in a fire of long duration, it is possible to release substantial fractions of the total radiological source term in fairly short periods, as the fire consumes areas of high MAR concentrations. The assumption of a 10-min release duration for fire is intended to generically account for this circumstance.

K.1.4.3 Modeling of Consequences of Releases to the Environment

The mean and 95th percentile consequences of accidental radiological releases, given variations in meteorological conditions at the time of the accident, are calculated as radiological doses in terms of rem. The mean consequences, or the expected consequences of the accident, are an appropriate statistic for use in risk estimates. The 95th percentile consequences represent bounding consequences of the accident; that is, if the accident were to occur and release the stated source term, there would be a 95 percent probability of lower than the stated consequences. This statistic is thus useful for characterizing the bounding consequence potential of the proposed activity under the stated accident condition. The consequences are also expressed as the additional potential or likelihood of death from cancer for the noninvolved worker and the maximally exposed member of the public, and the expected number of incremental LCFs among the exposed population.

The probability coefficients for determining the likelihood of fatal cancer, given a dose, are taken from the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991). For low doses or low dose rates, respective probability coefficients of 4.0×10^{-4} and 5.0×10^{-4} fatal cancers per rem are applied

for workers and the general public.[4] For high doses received at a high rate, respective probability coefficients of 8.0×10^{-4} and 1.0×10^{-3} fatal cancers per rem are applied for noninvolved workers and the public. These higher probability coefficients apply where doses are above 20 rem and dose rates above 10 rem/hr.

K.1.5 Accident Scenarios for Surplus Plutonium Disposition Facilities

Bounding design basis and beyond-design-basis accident scenarios have been developed from accident scenarios presented in each of the surplus plutonium disposition data reports (UC 1998a–h, 1999a–d). These scenarios are discussed in detail, along with specific assumptions for each facility and site, in these documents.

K.1.5.1 Accident Scenario Consistency

In preparing the accident analysis for the SPD EIS, the primary objective was to ensure consistency between the data reports so that results of the analyses for the proposed surplus plutonium disposition alternatives could be compared on as equal a footing as possible. In spite of efforts by all parties, some inconsistencies exist between the data reports. This does not imply technical inaccuracy in any analysis; it merely reflects the uncertainties and reliance on convention that are inherent in accident analyses in general. In order to provide a consistent analytical basis, information in the data reports has been modified or augmented as described below.

Aircraft Crash. It was decided early in the process of developing accident scenarios that aircraft crash scenarios would not be provided in the data reports, but would be developed, as appropriate, directly for the SPD EIS.

Frequencies of an aircraft crash into each facility for each alternative were developed in accordance with DOE-STD-3014 (DOE 1996c). The frequency of crashes involving aircraft capable of penetrating the subject facility (assumed to be all aircraft except those in general aviation) would be below 1.0×10^{-7} per year for all facilities except those at Pantex. For facilities at Pantex, the frequency of impact would be 1.7×10^{-6} per year.

Of the variety of impact conditions accounted for in the above frequency values (e.g., impact angle, direction, lateral distance from building center, speed) only a fraction would have the potential to produce consequences comparable to those reported in the SPD EIS, while other impacts (grazing impacts, impacts into office areas, etc.) would not result in significant radiological impacts. [Text deleted.] Aircraft crashes at Pantex with the potential for significant consequences could occur more frequently than 1.0×10^{-7} per year, so these scenarios were analyzed further.

For the facilities at Pantex, the potential for an aircraft crash into vaults containing large quantities of plutonium powder was examined in relation to the potential for a crash into the facility as a whole. For the pit conversion and mixed oxide (MOX) facilities, the footprint of the vault would be considerably less than one-tenth that of the facility as a whole, indicating that vault impact frequencies would be on the order of, and perhaps less than, one-tenth the facility impact frequencies. Moreover, fewer types of aircraft would have the potential to penetrate the vault due to the robustness of the reinforced-concrete vault structures and their location in the basements of the facilities. Inside the vault, the storage containers would provide additional protection against the release of material. The protection provided by the vault structure and the storage containers can be regarded as conducive to a further reduction in the frequency of aircraft crashes into vault areas.

In response to public concern over the risk of an aircraft crash at Pantex, and consistent with a Memorandum of Understanding between the DOE Amarillo Area Office and the Federal Aviation Administration (FAA), an

[4] Probability coefficients for the likelihood of nonfatal cancer are 8.0×10^{-5} for adult workers and 1.0×10^{-4} for the public. The probability coefficients for severe hereditary effects are 8.0×10^{-5} for adult workers and 1.3×10^{-4} for the public.

Overflight Working Group was established. This working group provided a number of recommendations for reducing the risk of an aircraft crash into any facility at Pantex. DOE supplemented the Memorandum of Understanding with an Interagency Agreement with the FAA. These actions resulted in the following recommendations:

- Modifying the vectoring of approaching aircraft to preclude extended flying over plant boundaries and reducing the number of aircraft turning on final approach over the plant
- Modifying holding patterns so that they are away from the plant
- Developing a new global positioning satellite (GPS), nonprecision approach to runway 22
- Replacing the backcourse localizer approach to runway 22 with an offset localizer approach
- Upgrading the lighting system for the approach to runway 4
- Establishing a hotline between the FAA and DOE
- Establishing new very high frequency omnidirection radio tactical (VORTAC) air navigation device locations
- Installing a GPS ground differential station, and commissioning a new GPS precision approach to runway 22

As of this date, all the recommendations except the last two have been implemented. The recommendation to install a precision approach is on hold until the FAA develops the standards for the augmentation system. While these changes cannot be quantitatively reflected in the frequency of aircraft crash as calculated by DOE-STD-3014, the improvements have been acknowledged as representing a reduction in the exposure of Pantex to aircraft, which translates to a reduction in the aircraft crash frequency at that site.

As a result of these considerations, it was qualitatively estimated that the overall scenario frequency of an aircraft crash into a plutonium powder vault associated with either the pit conversion or MOX facility was below the threshold frequency of 1.0×10^{-7} per year. Additionally, it was qualitatively estimated that in light of these considerations, the overall frequency of aircraft impact into the pit conversion or MOX facility at Pantex was below 1×10^{-6} per year, or "beyond extremely unlikely." The development of consequences of an aircraft crash was therefore refocused on the MAR that could be in process areas at the time of the crash. To develop representative consequences, it was assumed that the aircraft impact would involve the process area containing the largest amount of material in the most dispersable form. For the MOX facility, the impact was assumed to involve the unloading vessel and hopper storage, powder-blending process, and MOX powder storage areas. These processes would contain the bulk of process plutonium in powder form. The total quantity of plutonium in powder form would be 1.8×10^{5} g (6.3×10^{3} oz) (UC 1998d:table B-13), assuming that one-third of the plutonium in MOX powder storage was in powder form, one-third in green pellet form, and one-third in the form of sintered pellets. However, given the potentially high-energy densities associated with an aircraft crash, it was assumed that the green pellets would be equally vulnerable to release as powder, for a total effective powder quantity of 3.5×10^{5} g (1.2×10^{4} oz). For the pit conversion facility, the impact was assumed to involve the bisector, blending, canning, nondestructive analysis, and temporary storage areas, for a total of 6.0×10^{4} g (2.1×10^{3} oz) (UC 1998a:table 7-3) of plutonium in powder form.

The initial effect of the impact would be to disperse the material in a manner consistent with DOE-HDBK-3010-94 values for debris impact in powder. For this phenomenon, DOE-HDBK-3010-94

recommends bounding ARF and RF values of 1.0×10^{-2} and 0.2 (DOE 1994a:4-10), respectively, resulting in an initial source term of 117 g (4.1 oz) for the pit conversion facility and 690 g (24 oz) for the MOX facility. An aircraft crash could also induce a fire capable of entraining additional material in a lofted plume. The ARF and RF values for thermal stress, 6.0×10^{-3} and 1.0×10^{-2} (DOE 1994a:4-7), respectively, would result in a 3 percent increase in the source term. This additional source term should not contribute significantly to the noninvolved worker dose or the MEI dose, given the trajectory of the plume. However, it would contribute to the population dose. For simplicity, the source term was included in the ground-level release, yielding a total plutonium release of 124 g (4.4 oz) for the pit conversion facility and 710 g (25 oz) for the MOX facility.

The same source terms would result from postulated aircraft crashes into the pit conversion and MOX facilities regardless of their location. As discussed above, inclusion of the consequence analysis for Pantex, but not for other sites such as SRS, was solely due to differences in accident frequency.

Criticality. All of the data reports provide technically defensible information on criticality, but the analytical assumptions vary among the reports. To assess the significance of the variations, MACCS2 runs were performed for each criticality source term. The resulting doses varied by a factor of about 15 for all criticalities except the natural phenomena hazard (NPH) vault criticality in the immobilization data report. Doses from this criticality were roughly 100 times larger than any other doses and were dominated by aerosolized plutonium from the vault.

For the SPD EIS, it was decided to discard the NPH vault criticality on the grounds that it is, at most, an improbable event that is conditional on the occurrence of a beyond-design-basis earthquake and does not represent the potential consequences of an isolated criticality. Beyond-design-basis earthquakes have been addressed via a total collapse scenario in all data reports, and the additional assumption of a criticality occurring in addition to the total collapse does not significantly increase doses beyond those resulting from the collapse itself.

Of the remaining criticalities, the criticality in the rotary splitter tumbler in the glass immobilization data report produced the highest doses, dominated by fission products as opposed to plutonium. The source term for this criticality is based on a fission yield from 1.0×10^{19} fissions in an oxide powder.

For the SPD EIS, it was decided to use this source term for criticality for all facilities, because all facilities would handle oxide powder in quantities sufficient for criticality. For the aqueous plutonium-polishing process at the MOX facility, a solution criticality of 10^{19} fissions was also postulated, which bounds the powder criticality due to the greater release potential of fission products from solution. The estimated frequency of extremely unlikely (i.e., 10^{-6} to 10^{-4} per year) reported in the immobilization data report was also used because it is the bounding estimate.

The criticality source term provided in the immobilization data report neglects some very short-lived isotopes that would be expected in a criticality, namely bromine 85, iodine 136, krypton 89 and 90, and xenon 137. Since the half-lives of these isotopes are all less than 4 min, they do not have a significant direct impact on radiological consequences. However, the daughters of some of the isotopes are themselves radioactive; in particular, krypton 89 decays to rubidium 89, which has a half-life of 15 min. The significance of the daughters for overall consequences has been assessed for Pantex, which is considered bounding because Pantex has the highest windspeeds and tends to carry the daughters the farthest for a given level of decay. As expected, the increase in dose is greatest for the noninvolved worker; approximately 25 percent higher for both the mean and 95th percentile. The dose increase decreases to 3 and 13 percent, respectively, for the mean and 95th percentile doses to the population within 80 km (50 mi). Dose increases at other sites are expected to be lower than corresponding increases at Pantex. Because these increases are small considering the great uncertainty inherent in the estimate of the total number of fissions, the source term in the immobilization data report remains a conservative estimate of the potential release from a criticality accident, and no modification of the source term has been made.

Design Basis Earthquake. Each data report presents an analysis of the design basis earthquake. The immobilization and MOX data reports provide source terms for that earthquake, while the pit conversion data reports indicate no release as a result of a design basis earthquake because the facility would be designed to withstand the event.

For the SPD EIS, a nonzero source term for pit conversion was generated by applying a building ventilation LPF of 1.0×10^{-5}, accounting for a HEPA filtered release, to the beyond-design-basis earthquake source term. It is recognized that this is a conservative procedure, in that the beyond-design-basis earthquake would release more material into the air within the building than a design basis earthquake. The combined ARF×RF for powder under beyond-design-basis earthquake conditions has been assessed as three times that for design basis earthquake conditions, and the total amount of vulnerable material may be somewhat greater. (For perspective, it resulted in a ratio of design basis earthquake to beyond-design-basis earthquake source term values that is somewhat higher than the corresponding ratio for MOX fuel fabrication, but lower than for plutonium conversion and immobilization.)

Beyond-Design-Basis Earthquake. All of the proposed operations would be in either existing or new facilities that would be expected to meet or exceed the requirements of DOE O 420.1 (DOE 1995b) and DOE-STD-1020-94 for reducing the risks associated with natural phenomena hazards. The proposed facilities would be characterized as Performance Category 3 facilities. Such facilities would have to be designed or evaluated for a design basis earthquake with a mean annual exceedance probability of 5×10^{-4}, corresponding to a return period of 2,000 years. For sites such as Lawrence Livermore National Laboratory (LLNL), which are near tectonic plate boundaries, the requirements would include a mean annual seismic hazard exceedance probability of 1.0×10^{-3}, or a return period of 1,000 years.

The numerical seismic design requirements detailed in DOE-STD-1020-94 are structured such that there is assurance that specific performance goals are met. For plutonium facilities (Performance Category 3), the performance goal is that occupant safety, continued operation, and hazard confinement would be ensured for earthquakes with an annual probability exceeding approximately 1×10^{-4}. There is sufficient conservatism in the design of buildings and the structures, systems, and components important to safety that these goals should be met given that they are designed against earthquakes with an estimated mean annual probability of 5×10^{-4}.

[Text deleted.]

By contrast, nonnuclear structures at these sites and the surrounding community would be constructed to the standards of the Uniform Building Code for that region. These peak acceleration values are 50 to 82 percent of the peak acceleration design requirements for plutonium facilities in the same area and correspond approximately to DOE Performance Category 1 facilities with 500-year return intervals. During major earthquakes, structures built to these Uniform Building Code requirements would be expected to suffer significantly more damage than reinforced-concrete structures designed for plutonium operations.

At sites far from tectonic plate boundaries, deterministic techniques such as those used by NRC in evaluating safe-shutdown earthquakes for the siting of nuclear reactors have also been used to determine the maximum seismic ground motion requirements for facility designs. These techniques involve estimating the ground acceleration at the proposed facility either by assuming the largest historical earthquake within the tectonic province or by assessing the maximum earthquake potential of the appropriate tectonic structure or capable fault closest to the facility. For NRC-licensed reactors, this technique resulted in safe-shutdown earthquakes with estimated return periods in the 1,000- to 100,000-year range (DOE 1994a:C-17).

All the existing facilities under consideration in the SPD EIS have had seismic evaluations demonstrating that they meet the seismic evaluation requirements for the design basis earthquake. Some facilities, such as

Building 332 at LLNL under consideration for preparation of the lead test assemblies, have had extensive evaluations of the ability of the structures, systems, and components important to safety to survive a range of seismic loadings. Evaluations reported in the *Final Environmental Impact Statement and Environmental Impact Report for Continued Operation of Lawrence Livermore National Laboratory and Sandia National Laboratories, Livermore* (DOE 1992) indicate that Building 332 would survive a postulated 0.8g earthquake and retain those features essential for the safe containment of radioactive materials. The estimated return interval for this level of ground accelerations is about 10,000 years. The facility was also examined for damage due to a 0.9g earthquake and found to be survivable (DOE 1992:app. D.5.2.1), albeit with some potential for loss of confinement due to equipment damage in safety systems (DOE 1992:table I-14).

The magnitude of potential earthquakes with return periods greater than 10,000 years is highly uncertain. For purposes of the SPD EIS, it was assumed that at all the candidate sites, earthquakes with return periods in the 100,000- to 10-million-year range might result in sufficient ground motion to cause major damage to even a modern, well-engineered and well-constructed plutonium facility. Therefore, in the absence of convincing evidence otherwise, a total collapse of the plutonium facilities was assumed to be scientifically credible and within the rule of reason for return intervals in this range.

Each data report presents an analysis of total collapse. The immobilization and MOX data reports are fairly consistent in their use of damage estimates and release fractions. They assume that material in storage containers in vault storage would be adequately protected from the scenario energetics, for a damage ratio of zero in the vault. They also assume powder ARF and RF values of 1.0×10^{-3} and 0.3 (UC 1998c:tables 8-14 and 8-15; 1998d:169), respectively. The pit conversion data reports assume a damage ratio of 50 percent for material held in storage containers, applies cumulative ARF and RF values of 2.7×10^{-3} to powder subject to seismic vibration, free-fall spill, and turbulent air currents; and also presents a resuspension source term (UC 1998a:79–81).

For the SPD EIS, the pit conversion source term was modified by adjusting the damage ratio in the vault from 0.5 to 0 based on the corresponding analyses in the immobilization and MOX data reports, and adjusting the ARF and RF values for powder to 1.0×10^{-3} and 0.3, respectively. The assumption of vault survival in the beyond-design-basis earthquake was based on the fact that the vaults would be designed with significantly more robustness than the balance of the proposed facilities. The requirements for the additional robustness of the vault derive from the desire for increased protection of vault contents against external events such as aircraft crash or proliferation concerns, as well as increased earthquake survivability. It is expected that the vaults would survive the most likely seismic events of sufficient magnitude to collapse the processing areas of the proposed facilities. While there may be even more intense seismic events capable of compromising the protection afforded by the vaults, such events are expected to be beyond extremely unlikely.

The value of 2.7×10^{-3}, used in the pit conversion data report, is based on seismic-induced collapse of large structures into loose bulk powder; this assumption is considered unnecessarily conservative given the expectation of containered storage for the majority of the powder inventory at any given time. The resuspension source term was kept (and was not applied to either immobilization or MOX). Although worth noting, this difference between the data reports is not considered particularly significant, for the resuspension source term constitutes only 30 percent of the total.

The frequency for all beyond-design-basis earthquakes for all facilities is reported in the SPD EIS as extremely unlikely to beyond extremely unlikely (the pit conversion facility data report estimated a frequency of less than 1×10^{-6} per year.) They are reported as such because the uncertainties inherent in associating damage levels with earthquake frequencies become overwhelming below frequencies of about 1.0×10^{-5} per year.

Filtration Efficiency. The immobilization and MOX data reports use a building filtration efficiency of 1.0×10^{-5} for particulate releases (UC 1998c:8-3; 1998d:tables B-18–B-20). The pit conversion data report uses a building

filtration efficiency of 2.0×10^{-6} (UC 1998a:73). For consistency, the pit conversion source terms have been adjusted to reflect an LPF of 1.0×10^{-5}. This is reasonable because it is expected that the ventilation efficiencies of all HEPA-filtered buildings would be essentially the same.

Beyond-Design-Basis Fire. The MOX data report presents an analysis of a beyond-design-basis fire whose basis in terms of scenario definition was from the *Data Report for Plutonium Conversion Facility* (Smith, Wilkey, and Siebe 1996), which was produced for the *Storage and Disposition PEIS* (DOE 1996a). Neither the pit conversion nor the immobilization data reports contain analyses of a beyond-design-basis fire.

For the SPD EIS, beyond-design-basis fires were developed for pit conversion and immobilization by replacing the building filtration LPF with an LPF of 1.4 percent, in accordance with the beyond-design-basis scenario definition presented in the *Data Report for Plutonium Conversion Facility* (Smith, Wilkey, and Siebe 1996) and adapted for the MOX fuel fabrication analysis. (For perspective, it resulted in a ratio of design basis fire to beyond-design-basis fire source term values that are within a factor of 2 of the corresponding ratio for MOX fuel fabrication.)

It is understood that the LPF of 1.4 percent is based on a facility-specific analysis of the Plutonium Finishing Building (PF–4) in Technical Area 55 at LANL, and that an analysis of other facilities using the same phenomenological assumptions might yield somewhat different results. However, for the purpose of this analysis, and considering the degree of similarity expected between facilities as a result of required plutonium-handling practices, this value was used generically in the assessment of beyond-design-basis fire.

K.1.5.2 Facility Accident Scenarios

K.1.5.2.1 Pit Conversion Facility

A wide range of potential accident scenarios were considered for the pit conversion facility. These scenarios are considered in detail in the pit conversion facility data reports (UC 1998a, 1998c, 1998e, 1998f). The analysis assumes that the pit conversion facility is located in a new or upgraded existing building designed to withstand design basis natural phenomena hazards such as earthquakes, winds, tornadoes, and floods such that no unfiltered releases would be expected. Also, no site-specific accidents conducive to releases are identified. Therefore, the potential accident scenarios apply to all four candidate sites.

Analysis of the proposed process operations for the pit conversion facility identified the following broad categories of accidents: aircraft crash, criticality, design basis earthquake, beyond-design-basis earthquake, explosion, fire, leaks or spills, and tritium release. Basic characteristics of each of these postulated accidents are described below. Additional discussion of scenario development based on consistency concerns can be found in Appendix K.1.5.1.

Aircraft Crash. A crash of a large, heavy commercial or military aircraft directly into a reinforced-concrete facility could damage the structure sufficient to breach confinement and disperse material into the environment. A subsequent fuel-fed fire could provide energy to further damage structures and equipment, aerosolize material, and drive materials into the environment. Source terms are highly speculative but would be expected to exceed those from the beyond-design-basis earthquake. At all sites except Pantex, the frequency of such a crash is below 10^{-7} per year.

Criticality. Engineered and administrative controls should be available to ensure that the double-contingency principles are in place for all portions of the process. It is assumed that human error results in multiple failures leading to an inadvertent nuclear criticality. The estimated frequency of this accident is in the range of 10^{-4} to 10^{-6} per year. A bounding source term resulting from 10^{19} fissions is assumed.

Design Basis Earthquake. The principal design basis natural phenomena event that could release material to the environment is the design basis earthquake. While the major safety systems, including building confinement and the building HEPA filtration system should continue to function, the vibratory motion would be expected to resuspend loose plutonium powder within gloveboxes and cause some minor spills. These would be picked up by the ventilation system and filtered by the HEPA filters before release from the building. Although highly uncertain, the source term should be much lower than that postulated for the beyond-design-basis earthquake. Based on an LPF of 1.0×10^{-5} for two HEPA filters, a stack release of 3.9×10^{-4} g (1.4×10^{-5} oz) is postulated. The estimated frequency of this accident is in the range of 10^{-4} to 10^{-2} per year.

Beyond-Design-Basis Earthquake. The postulated beyond-design-basis earthquake is assumed to be of sufficient magnitude to cause total collapse of the process equipment, building walls, roof, and floors, and loss of the containment function of the building. The material in the building is assumed to be driven airborne by the seismic vibrations, free-fall during the collapse, and impact. Molten metal in furnaces is also assumed to burn in the aftermath of the collapse. An instantaneous plus-resuspension ground-level release of 39 g (1.4 oz) of respirable plutonium is estimated for the process area. While the release of an additional 2,529 g (89 oz) from the vault would be possible, it would be unlikely given the expected packaging of materials in the vault. The estimated frequency of this accident is in the range of 10^{-5} to 10^{-7} per year.

Explosion. The bounding explosion is a deflagration of a hydrogen gas mixture inside the hydride oxidation (HYDOX) furnace. The deflagration is assumed to result from multiple equipment failures and operator errors that lead to a buildup of hydrogen and a flow of oxygen into the inert-atmosphere glovebox used in the HYDOX process. Also assumed is an MAR of 4.5 kg (9.9 lb) of plutonium powder, and given the venting of pressurized gas through the powder, bounding ARF and RF of 0.1 and 0.7, respectively. The explosive energy would be sufficient to damage glovebox windows but insufficient to threaten the building HEPA filter system. Based on an LPF of 1.0×10^{-5} for two HEPA filters, a stack release of 3.2×10^{-3} g (1.1×10^{-4} oz) is postulated. The estimated frequency of this accident is in the range of 10^{-2} to 10^{-4} per year.

Fire. According to the several safety analyses of the plutonium facility at LANL, the bounding fire within the pit conversion facility is a fire involving all of the gloves in a glovebox used for blending plutonium powder. A flammable cleaning liquid is assumed to be brought into the glovebox, in violation of procedure, then to spill and ignite. The gloves are assumed to be stowed outside the glovebox but to be ignited by the fire and completely consumed. An MAR of 2 g (0.07 oz) of plutonium dust is assumed for each of 12 gloves, with all of the 24 g (0.85 oz) assumed to be aerosolized. The sprinkler system is assumed to function and protect the room and remainder of the building. Also assumed are an ARF of 0.05 and an RF of 1.0, resulting in a 1.2-g (0.04-oz) release to the building ventilation system. Based on an LPF of 1.0×10^{-5} for two HEPA filters, a stack release of 1.2×10^{-5} g (4.2×10^{-7} oz) is postulated. The estimated frequency of this accident is in the range of 10^{-2} to 10^{-4} per year.

Leaks or Spills of Nuclear Material. The most catastrophic leak or spill postulated would result from a forklift or other large vehicle running over a package of nuclear material and breaching the storage container. If a 4-kg (8.8-lb) package of plutonium oxide were breached, a total airborne release of 0.44 g (0.016 oz) to the room would occur, and after HEPA filtration of the facility exhaust, a total release of 4.4×10^{-6}. This accident has an estimated frequency in the range of 10^{-4} to 10^{-6} per year.

Tritium Release. A major glovebox fire is assumed to heat multiple parts contaminated with up to 20 g (0.71 oz) of tritium and convert all of it into tritiated water vapor. Very conservatively, the ARF, RF, and LPF are all assumed to be 1.0, resulting in a release of 20 g (0.71 oz) (1.9×10^{-5} Ci) through the stack to the atmosphere. The estimated frequency of this accident is in the range of 10^{-4} to 10^{-6} per year.

K.1.5.2.2 Immobilization Facility

A wide range of potential accident scenarios are reflected in the immobilization facility data reports (UC 1999a–d). The analysis assumes that the immobilization facility is located in a new or upgraded existing building designed to withstand design basis natural phenomena hazards such as earthquakes, winds, tornadoes, and floods such that no unfiltered releases would be expected. Also, no site-specific accidents conducive to releases are identified. Therefore, the potential accident scenarios apply to all four candidate sites. Additional discussion of scenario development based on consistency concerns can be found in Appendix K.1.5.1.

Analysis of the proposed process operations identified specific scenarios for the conversion process, each of the immobilization options (ceramic and glass), and the canister-handling portion of the process. Design basis and beyond-design-basis earthquakes were identified for the overall facility. Identified as accidents specific to the plutonium conversion processes were a criticality, an explosion in HYDOX furnace, a calcining furnace–glovebox fire, and a hydrogen explosion in the plutonium conversion room. For the ceramic immobilization option, moreover, a sintering furnace–glovebox fire was identified; for the glass immobilization option, a melter eruption and a melter spill. All of the scenarios identified with the canister-handling phase were negligible compared with the conversion and immobilization scenarios.

Plutonium Conversion Operations

Criticality. Review of the possibility of accidents attributable to plutonium conversion operations indicated that the principal processes of concern include the halide wash operations, the HYDOX furnace, and the sorting/unpacking glovebox. Engineered and administrative controls should be available to ensure that the double-contingency principles are in place for all portions of the process. It is assumed that human error could result in multiple failures leading to an inadvertent nuclear criticality. The estimated frequency of this accident is in the range of 10^{-4} to 10^{-6} per year. A bounding source term resulting from 10^{19} fissions is assumed.

Explosion in HYDOX Furnace. The bounding explosion is a deflagration of a hydrogen gas mixture inside the HYDOX furnace. The deflagration is assumed to result from multiple equipment failures and operator errors that lead to a buildup of hydrogen and a flow of oxygen into the inert-atmosphere glovebox used in the HYDOX process. Also assumed is an MAR of 4.8 kg (11 lb) of plutonium powder, and given the venting pressurized gas through the powder, bounding ARF and RF of 0.1 and 0.7, respectively. The explosive energy would be sufficient to damage glovebox windows but insufficient to threaten the building HEPA filter system. Based on an LPF of 1.0×10^{-5} for two HEPA filters, a stack release of 3.4×10^{-3} g (1.2×10^{-4} oz) is postulated. The estimated frequency of this accident is approximately 10^{-3} per year or in the unlikely range.

Hydrogen Explosion in Plutonium Conversion Room. A supply pipe leak in the plutonium conversion room could result in a hydrogen explosion. Conversion of plutonium metal is accomplished using the HYDOX process, which entails the introduction of hydrogen gas. Were the hydrogen supply piping to leak into the operating/maintenance room, the gas could be ignited by an electrical short or operating mechanical equipment, causing an explosion. Depending on the volume of the leak, the structural integrity of the glovebox glove ports could fail and disperse the plutonium oxide. It is assumed that the building ventilation does not fail, and that the two HEPA filters provide filtration prior to discharge of the powder to the stack. An entire day's inventory of 25 kg (55 lb) of plutonium oxide powder is assumed present in the plutonium conversion gloveboxes. Based on an ARF of 5×10^{-3}, an RF of 0.3, and an LPF of 1.0×10^{-5} for two HEPA filters, a stack release of 3.8×10^{-4} g (1.3×10^{-5} oz) of plutonium is postulated. The estimated frequency of this accident is approximately 10^{-3} per year or in the unlikely range.

Furnace-Initiated Glovebox Fire (Calcining Furnace). It is assumed that a fault in the calcining furnace results in the ignition of any combustibles (e.g., bags) left inside the glovebox. The fire would be self-limiting, but would cause suspension of the radioactive material. It is also assumed that the glovebox (including the window) maintains its structural integrity, but that the internal glovebox HEPA filter fails. All of the loose

surface contamination within the glovebox, assumed to be 10 percent of the daily inventory (4.5 kg [9.9 lb] of plutonium) of the calcining furnace, is assumed to be involved. Based on an ARF of 6×10^{-3}, an RF of 0.01, and an LPF of 1.0×10^{-5} for two HEPA filters, a stack release of 2.7×10^{-7} g (9.5×10^{-9} oz) of plutonium is postulated. The estimated frequency of this accident is in the range of 10^{-4} to 10^{-6} per year.

CERAMIC IMMOBILIZATION OPTION

Criticality. Review of the possibility of accidents attributable to the ceramic immobilization operations indicated that the principal operation of concern is the rotary splitter tumbler. Engineered and administrative controls should be available to ensure that the double-contingency principles are in place for all portions of the process. It is assumed that human error results in multiple failures leading to an inadvertent nuclear criticality. The estimated frequency of this accident is in the range of 10^{-4} to 10^{-6} per year. A bounding source term resulting from 10^{19} fissions is assumed.

Design Basis Earthquake. The principal design basis natural phenomena event that could release material to the environment is the design basis earthquake. While the major safety systems, including building confinement and the building HEPA filtration system should continue to function, the vibratory motion would be expected to suspend loose plutonium powder within gloveboxes and cause some minor spills. These would be picked up by the ventilation system and filtered by the HEPA filters before release from the building. Most material storage containers are assumed to be engineered to withstand design basis earthquakes without failing. For plutonium conversion, it is assumed that at the time of the event the entire day's inventory (25 kg [55 lb] of plutonium) is present in the form of oxide powder. For the ceramic immobilization portion, this includes the oxide inventories from the rotary splitter, oxide grinding, blend and granulate feed storage, drying and storage, pressing, inspection, and load trays and weigh areas. Although the source term is highly uncertain, an assessment of the MAR, ARF, and RF for each of the process areas indicated a potential for the release of 38 g (1.3 oz) of plutonium to the still-functioning building ventilation system and 3.8×10^{-4} g (1.3×10^{-5} oz) from the stack. The nominal frequency estimate for a design basis earthquake affecting new DOE plutonium facilities is 5×10^{-4} per year, or in the unlikely range.

Beyond-Design-Basis Earthquake. The postulated beyond-design-basis earthquake is assumed to be of sufficient magnitude to cause total collapse of the process equipment, building walls, roof, and floors, and loss of the containment function of the building. The material in the building is assumed to be driven airborne by the seismic vibrations, free-fall during the collapse, and impact. Material in storage containers in vaults would be adequately protected from the scenario energetics. Although the source term is highly uncertain, an assessment of the MAR, ARF, and RF for each of the process areas indicated a potential for the release of 19 g (0.67 oz) of plutonium at ground level. The estimated frequency of this accident is in the range of 10^{-5} to 10^{-7} per year.

Furnace-Initiated Glovebox Fire (Sintering Furnace). It is assumed that the sintering gas supplied to the furnace gloveboxes is a safe gas mixture—hydrogen and argon. Human errors are at issue—either a vendor/supplier that causes a supply of air or noninerting gas to be supplied to the furnace glovebox, or a piping error at the facility itself, in which oxygen is inadvertently substituted for the inert gas. Any combustibles (e.g., bags) left inside the glovebox could ignite, causing a glovebox fire. It is assumed that the fire is self-limiting, but causes suspension of the radioactive material. It is also assumed that the glovebox (including the window) maintains its structural integrity, but that the internal glovebox HEPA filter fails. All of the loose surface contamination within the glovebox, assumed to be 10 percent of the daily inventory (25 kg [55 lb] of plutonium) of the calcining furnace, is assumed to be involved. Based on an ARF of 6×10^{-3}, an RF of 0.01, and an LPF of 1.0×10^{-5} for two HEPA filters, a stack release of 1.5×10^{-6} g (5.3×10^{-8} oz) of plutonium is postulated. The estimated frequency of this accident is in the range of 10^{-4} to 10^{-6} per year.

GLASS IMMOBILIZATION OPTION

Design Basis Earthquake. The principal design basis natural phenomena event that could release material to the environment is the design basis earthquake. While the major safety systems, including building confinement and the building HEPA filtration system should continue to function, the vibratory motion would be expected to suspend loose plutonium powder within gloveboxes and cause some minor spills. These would be picked up by the ventilation system and filtered by the HEPA filters before release from the building. Most material storage containers are assumed to be engineered to withstand design basis earthquakes without failing. For plutonium conversion, it is assumed that at the time of the event the entire day's inventory (25 kg [55 lb] of plutonium) is present in the form of oxide powder. For the glass immobilization portion, this includes oxide inventories from the rotary splitter, oxide grinding, blend melter, and feed storage. Although the source term is highly uncertain, an assessment of the MAR, ARF, and RF for each of the process areas indicated a potential for the release of 33 g (1.2 oz) of plutonium to the still-functioning building ventilation system and 3.3×10^{-4} g (1.2×10^{-5} oz) from the stack. The nominal frequency estimate for a design basis earthquake affecting new DOE plutonium facilities is 5×10^{-4} per year, or in the unlikely range.

Beyond-Design-Basis Earthquake. The postulated beyond-design-basis earthquake is assumed to be of sufficient magnitude to cause total collapse of the process equipment, building walls, roof, and floors, and loss of the containment function of the building. The material in the building is assumed to be driven airborne by the seismic vibrations, free-fall during the collapse, and impact. Material in storage containers in vaults storage would be adequately protected from the scenario energetics. Although the source term is highly uncertain, an assessment of the MAR, ARF, and RF for each of the process areas indicated a potential for the release of 17 g (0.60 oz) of plutonium released at ground level. The estimated frequency of this accident is in the range of 10^{-5} to 10^{-7} per year.

Melter Eruption. A melter eruption could result from the buildup of impurities in, or addition of impurities to, the glass frit or melt. Impurities range from water, which could cause a steam eruption, to chemical contaminants, which could react at elevated temperatures and produce a highly exothermic reaction (eruption or deflagration). The resulting sudden pressure increase could eject the fissile material bearing melt liquid into the processing glovebox structure. However the energy release would likely be insufficient to challenge the glovebox structure. It is assumed that the entire contents of the melter, about 1.4 kg (3.1 lb) of plutonium, are ejected into the glovebox. Based on an ARF of 4×10^{-4}, an RF of 1, and an LPF of 1.0×10^{-5} for two HEPAs, a stack release of 1.4×10^{-6} g (4.9×10^{-8} oz) of plutonium is postulated. The estimated frequency of this accident is approximately 2.5×10^{-3} per year, or in the unlikely range.

Melter Spill. A melter spill into the glovebox could occur due to improper alignment of the product glass cans during pouring operations. The melter glovebox enclosure and the off-gas exhaust ventilation system would confine radioactive material released in the spill. The glovebox structure and its associated filtered exhaust ventilation system would not be impacted by this event. It is assumed that the entire contents of the melter, about 1.4 kg (3.1 lb) of plutonium, are spilled into the glovebox. On the basis of an ARF of 2.4×10^{-5}, a RF of 1, and an LPF of 1.0×10^{-5} for two HEPAs, a stack release of 3.3×10^{-7} g (1.2×10^{-8} oz) of plutonium is postulated. The estimated frequency of this accident is approximately 3×10^{-4} per year, or in the unlikely range.

Can-In-Canister Operations

Can-Handling Accident (Before Shipment to Vitrification Facility). A can-handling accident would involve a can containing either ceramic pellets or a vitrified glass log of plutonium material. Studies supporting the Defense Waste Processing Facility (DWPF) SAR (UC 1999a–d) indicate that the source term resulting from dropping or tipping a log of vitrified waste, even without credit for the steel canister, would be negligible. Both surplus plutonium immobilization technologies (ceramic and glass) result in a form with a durability that is comparable to that of the DWPF vitrified waste form. Consequently, no postulated can-handling event would result in a radioactive release to the environment.

Melter Spill (Melt Pour at Vitrification Facility). Analysis of a spill of melt material was included in studies performed in support of the DWPF SAR. According to that analysis, the source term resulting from the dropping or tipping a log of vitrified waste, even without credit for the steel canister, would be negligible. Both surplus plutonium immobilization technologies (ceramic and glass) result in a form with a durability that is comparable to the DWPF vitrified waste form. Consequently, it is postulated that no melter spill event results in a radioactive release to the environment.

Canister-Handling Accident (After Melt Pour at DWPF). Analysis of events involving the handling and storage of vitrified waste canisters was included in studies performed in support of the DWPF SAR. Results of that analysis indicate that the source term resulting from the dropping or tipping of a log of vitrified waste, even without credit for the steel canister, would be negligible. Both surplus plutonium immobilization technologies (ceramic and glass) result in a form with a durability that is comparable to the DWPF vitrified waste form. Consequently, it is postulated that no canister-handling event results in a radioactive release to the environment.

K.1.5.2.3 MOX Facility Accident Scenarios

A wide range of potential accident scenarios were considered in the analysis reflected in the MOX facility data reports (UC 1998b, 1998d, 1998g, 1998h). The analysis assumes that the MOX facility is located in a new or upgraded existing building designed to withstand design basis natural phenomena hazards such as earthquakes, winds, tornadoes, and floods such that no unfiltered releases would be expected. The MOX facility includes an aqueous plutonium-polishing process by which impurities, in particular gallium, are removed from the plutonium feed for MOX fuel fabrication. Bounding accidents for this process were developed separately from the accidents reflected in the MOX facility data reports and are documented in a stand-alone, process-specific data report (ORNL 1998).

Analysis of the proposed process operations for the MOX facility identified the following broad categories of accidents: aircraft crash (Pantex only), criticality, design basis earthquake, beyond-design-basis earthquake, explosion in sintering furnace, fire, and beyond-design-basis fire. Basic characteristics of each of these postulated accidents are described below. Additional discussion of scenario development based on consistency concerns can be found in Appendix K.1.5.1.

Aircraft Crash. A crash of a large, heavy commercial or military aircraft directly into a reinforced-concrete facility could damage the structure sufficiently to breach confinement and disperse material into the environment. A subsequent fuel-fed fire could provide energy to further damage structures and equipment, aerosolize material, and drive materials into the environment. Source terms are highly speculative but would be expected to exceed those from the beyond-design-basis earthquake. At all sites except Pantex, the frequency of such a crash is below 10^{-7} per year.

Criticality. Review of the possibility of accidents for the MOX facility indicated no undue criticality risk associated with the proposed operations. Engineered and administrative controls should be available to ensure that the double-contingency principles are in place for all portions of the process. It is assumed that human error could result in multiple failures leading to an inadvertent nuclear criticality. The estimated frequency of this accident is in the range of 10^{-4} to 10^{-6} per year. A bounding source term resulting from 10^{19} fissions in solution is assumed.

Design Basis Earthquake. The principal design basis natural phenomena event that could release material to the environment is the design basis earthquake. While the major safety systems, including building confinement and the building HEPA filtration system should continue to function, the vibratory motion would be expected to resuspend loose plutonium powder within gloveboxes and cause some minor spills. These would be picked up by the ventilation system and filtered by the HEPA filters before to release from the building. Material storage

containers including cans, hoppers, and bulk storage vessels are assumed to be engineered to withstand design basis earthquakes without failing. Although the source term is highly uncertain, an assessment of the MAR, ARF, and RF for each of the process areas indicated a potential for the release of 4 g (0.14 oz) of plutonium (in the form of MOX powder) to the still-functioning building ventilation system and 4.0×10^{-5} g (3.5×10^{-7} oz) from the stack. The nominal frequency estimate for a design basis earthquake for new DOE plutonium facilities is 5×10^{-4} per year, or in the unlikely range.

Beyond-Design-Basis Earthquake. The postulated beyond-design-basis earthquake is assumed to be of sufficient magnitude to cause total collapse of the process equipment, building walls, roof, and floors, and loss of the containment function of the building. The material in the building is assumed to be driven airborne by the seismic vibrations, free-fall during the collapse, and impact. Although the source term is highly uncertain, an assessment of the MAR, ARF, and RF for each of the process areas indicated a potential for the release of 124 g (4.4 oz) of plutonium (in the form of MOX powder) at ground level. The estimated frequency of this accident is in the range of 10^{-5} to 10^{-7} per year.

Explosion in Sintering Furnace. The several furnaces proposed for the MOX fuel fabrication process all use nonexplosive mixtures of 6 percent hydrogen and 94 percent argon. Given the physical controls on the piping for nonexplosive and explosive gas mixtures, operating procedures, and other engineered safety controls, accidental use of an explosive gas is extremely unlikely, though not impossible. A bounding explosion or deflagration is postulated to occur in one of the three sintering furnaces in the MOX facility building. Multiple equipment failures and operator errors would be required to lead to a buildup of hydrogen and an inflow of oxygen into the inert furnace atmosphere. As much as 5.6 kg (12.3 lb) of plutonium in the form of MOX powder would be at risk, and a bounding ARF of 0.01 and RF of 1.0 is assumed. Based on an LPF of 1.0×10^{-5} for two HEPA filters, a stack release of 5.6×10^{-4} g (2.0×10^{-5} oz) of plutonium (in the form of MOX powder) is postulated. It is estimated that the frequency of this accident is in the range of 10^{-4} to 10^{-6} per year.

Ion Exchange Column Exotherm. A thermal excursion within an ion exchange column is postulated to result from offnormal operations, degraded resin, or a glovebox fire. It is also assumed that the column venting/pressure relief valve fails to vent the overpressure, causing the column to rupture violently. The overpressure releases plutonium nitrate solution as an aerosol within the affected glovebox, which in turn is processed through the ventilation system. If the overpressure also breaches the glovebox, a fraction of the aerosol is released within the room as well. The combined ARF and RF values for this scenario are 9.0×10^{-3} for burning resin and 6.0×10^{-3} for liquid behaving as a flashing spray on depressurization. Additionally, 10 percent of the resin is assumed to burn, yielding a combined ARF and RF value of 9.0×10^{-3} for loaded plutonium. The LPF for the ventilation system is 1.0×10^{-5}.

With regard to probability, process controls are used to ensure that nitrated anion exchange resins are maintained in a wet condition, that the maximum nitric acid concentration and the operating temperature are limited to safe values, and that the time for absorption of plutonium in the resin is minimized. With these controls in place, the frequency of this accident is estimated to be in the unlikely range.

Fire. It is assumed that the liquid organic solvent containing the maximum plutonium concentration leaks as a spray into the glovebox, builds to a flammable concentration, and is contacted by an ignition source. The combined ARF and RF value for this scenario is 1.0×10^{-2} for quiescent burning to self-extinguishment. The LPF for the ventilation system is 1.0×10^{-5}. Scenario frequency is assessed as unlikely.

Spill. Leakage of liquids from process equipment must be considered as an anticipated event. However, with multiple containment barriers, a release from the process room would be extremely unlikely. A bounding scenario involved a liquid spill of concentrated aqueous plutonium solution, with 50 l (13.2 gal) accumulating before the

leak is stopped. The ARF and RF values used for this scenario are 2.0×10^{-4} and 0.5, respectively. The LPF for the building ventilation system is 1.0×10^{-5}.

Beyond-Design-Basis Fire. The MOX facility would be built and operated such that there would be insufficient combustible materials to support a large fire. To bound the possible consequences of a major fire, a large quantity of combustible materials are assumed to be introduced into the process area near the blending area, which contains a fairly large amount of plutonium. A major fire is assumed to occur that causes the building ventilation and filtration systems to fail, possibly due to clogged HEPA filters. A total of 11 kg (24 lb) of plutonium in the form of MOX powder is assumed at risk. Based on an ARF of 6×10^{-3}, a RF of 0.01, and an LPF of 1.4×10^{-2} for two damaged, clogged HEPA filters, a stack release of 9.4×10^{-3} g (3.3×10^{-4} oz) of plutonium (in the form of MOX powder) is postulated. It is estimated that the frequency of this accident is less than 10^{-6} per year.

K.1.5.2.4 Lead Assembly Accident Scenarios

Design basis and beyond-design-basis accident scenarios have been developed for the fabrication of MOX fuel lead assemblies. These scenarios are discussed in detail, with specific assumptions for each facility and site, in the site data reports (O'Connor et al. 1998a–e). In spite of efforts by all parties, however, some inconsistencies exist between the data reports. This does not imply technical inaccuracy in any analysis; it merely reflects the uncertainties and reliance on convention inherent in accident analyses in general. In preparing the accident analysis for the SPD EIS, therefore, information in the data reports was modified or augmented to ensure the consistency, as appropriate, that is necessary for a reliable comparison of lead assembly fabrication accidents and the other accidents analyzed herein. Modifications were made to ensure that, to the extent practical, differences in analytical results were based on actual differences in facility conditions, as opposed to arbitrary differences in analytical methods or assumptions. One change, reflected in Table K–2, involved the assumption for all accidents of an isotopic composition of plutonium identical to that assumed in the analyses of pit disassembly and conversion and MOX fuel fabrication.

Table K–2. Isotopic Composition of Plutonium Used in Lead Assembly Accident Analysis

Isotope	Weight Percent
Plutonium 238	3.0×10^{-2}
Plutonium 239	92.2
Plutonium 240	6.46
Plutonium 241	5.0×10^{-2}
Plutonium 242	1.0×10^{-1}
Americium 241	9.0×10^{-1}

Criticality. Criticalities could be postulated in several areas (e.g., powder storage, the gloveboxes involved in mixing, the furnace, the fuel rod storage area). The estimated frequencies associated with these events would vary depending on the controls in place, the number of operator movements, and the amount of fissile material present. A generic approach was taken with respect to the selection of the specifics of this event, rather than selection of a criticality scenario associated with a specific operation in the lead assembly fabrication.

The criticality source term stipulated in the data reports was modified to make it identical to the corresponding source term used in the assessment of criticality in the pit conversion, immobilization, and MOX facilities. That source term is based on a fission yield from 1.0×10^{19} fissions in an oxide powder. The discussion provided in Appendix K.1.5 on criticality is also applicable here.

Design Basis Earthquake. An earthquake appropriate with the facility's design basis was selected. For this event, major portions of the process line gloveboxes are assumed to be breached, making the contents available for release. The storage vault and receiving area are assumed to have suitable storage containers for plutonium oxide that would survive the earthquake (storage containers with double containment). In-process material in gloveboxes is, however, more vulnerable, as are powder storage areas that may exist. Of particular concerns are the dispersable powders at the powder-blending stations. Finished pellets and fuel rods are thought to be generally nondispersable, even though they could escape the gloveboxes. In this earthquake, some non-seismically qualified process equipment could fail, and some process material spill. It is also conservatively assumed that glovebox filtration would fail.

The lead assembly data reports use ARF and RF values of 1.0×10^{-2} and 0.2, respectively, for plutonium oxide in cans involved in a design basis earthquake. These values are based on DOE-HDBK-3010-94 recommendations for the suspension of bulk powder by debris impact and air turbulence from falling objects. For consistency with the design basis accident analyses for the other facilities, these values were changed to 1.0×10^{-3} and 0.1, values based on DOE-HDBK-3010-94 recommendations for the suspension of bulk powder due to vibration of substrate from shock-impact to powder confinement (e.g., gloveboxes, cans) due to external energy (e.g., seismic vibrations). Such values are appropriate for earthquakes in which structural integrity is largely maintained and there is not a significant amount of debris or falling objects.

Beyond-Design-Basis Earthquake. For this analysis an event much more severe in consequences than would be expected from the design basis earthquake was examined. For some existing DOE facilities, the estimated seismic frequencies of beyond-design-basis events can be greater than 1.0×10^{-6} per year. The design basis for every building in the complex varies considerably depending on site specifics, including the type of construction used in the building. A damage assessment of the facility is further complicated by the fact that seismic considerations could also be incorporated in the glovebox design of the facility. In reality, such a catastrophic event may or may not demolish the building and the gloveboxes. However, for the purposes of illustrating a high-consequence accident, total demolition of the building is assumed. In this event, no credit is taken for the building, filters, or gloveboxes.

In the data report, an estimated frequency of 1.0×10^{-6} per year is cited as appropriate. To acknowledge the high degree of uncertainty in assessing a frequency of this scenario, a range of extremely unlikely to beyond extremely unlikely has been assigned to this event.

The source term for the beyond-design-basis earthquake includes a contribution from the plutonium storage vault, the assumed DR being 5 percent. The values used for the ARF, RF and vault DR—1.0×10^{-3}, 0.3, and 0, respectively—derive from adjustments consistent with the analysis of the corresponding scenario in the MOX facility data report. This results in a reduction of the source term for this accident by a factor of 2, to 11 g (0.39 oz) plutonium.

Extensive analyses have been performed on the seismic hazard at LLNL and the response of the plutonium facility, Building 332, to that hazard. According to the geology and seismology studies characterizing the nature and magnitude of the seismic threat, there is no physiographic basis for postulating earthquake magnitudes and ground accelerations higher than Richter magnitude 6.9 and 1.1g, respectively. Building 332, Increment III, has been evaluated for resistance to earthquakes and ground accelerations of these magnitudes and found to be adequate. Events of significantly higher magnitude and ground acceleration would be required to collapse Increment III. The frequency of these larger events would most likely be extremely low (1.0×10^{-6} per year or less), as the physiography of the dominant fault systems is such that they are thought incapable of producing the required magnitudes of ground accelerations (Coats 1998). Results of a number of reviews of Increment III indicate that the actual ground motion needed to cause collapse of the structure is above 1.5g. Based on the current LLNL hazard curve and various estimates of the fragility curves for collapse of Increment III, the

frequency of collapse is estimated at 1.0×10^{-7} per year or less (Murray 1998). The frequency of a total collapse of Building 332 at LLNL is thus considered sufficiently low that additional examination is unnecessary.

Explosion. An explosion event was postulated in the sintering furnace in the lead assembly fabrication facility. A nonexplosive mixture of 6 percent hydrogen and 94 percent argon is used in the furnace. Multiple equipment and operator errors would have to occur to enable the buildup of an explosive mixture of hydrogen and air in the box. It is assumed that green pellets are subjected to the direct force of the shock waves resulting from such an explosion. It is further assumed that the gloveboxes involved in powder blending are damaged indirectly by the explosion. It is not expected that the shock wave impacting this area would be severe enough to significantly damage all of the storage inventory because interim storage containers would provide some mitigation.

Fire. A moderate-size room fire is assumed. Combustible material such as hydraulic fluid, alcohol, or contaminated combustibles is assumed to be present in the room. Adjoining facilities such as offices conceivably add to the risk of fires in the building. The gloveboxes are assumed to fail in the fire. The MOX powder in interim storage is assumed to be at risk and subjected to the thermal stress of the fire, given failure of the gloveboxes. Because of the limited combustible material and mitigation features such as fire protection systems and a firefighting unit, the event is assumed to be terminated. This fire is not severe enough to jeopardize the overall confinement characteristics of the building.

The source term for the design basis fire analyzed in the lead assembly data reports is dominated by the explosive release of high pressure from two plutonium oxide cans as they are heated to the point of failure. The ARF and RF values for this phenomenon are 0.1 and 0.7, respectively, and reflect burst pressures on the order of 25 to 500 psig. The potential for this kind of release is highly uncertain, and a valid design basis fire may be defined without including it, as is the case with the data reports for the other facilities. Therefore, for greater consistency between the design basis fire for the lead assembly and those for the other facilities, it is assumed that the two plutonium oxide cans are already open and vulnerable to the same phenomena as the rest of the analyzed powder. This results in a reduction of the data report source term by a factor of 38.

It is noteworthy that the lead assembly data report assumes a room fire, and the other data reports, a process fire. This is not considered inconsistent: the lead assembly processes are expected to be closer to one another other than the MOX processes, so the potential for propagation of fire may be somewhat greater.

Beyond-Design-Basis Fire. Fuel-manufacturing operations do not involve the use of significant amounts of combustible material. For the purpose of analysis, the lead assembly data reports define a beyond-design-basis fire that results in building collapse, the breach of material in the plutonium storage vault, and a lofted plume. These assumptions, however, are inconsistent with the beyond-design-basis fires analyzed for the other facilities. The beyond-design-basis fire has therefore been modified to reflect a room fire or building fire that clogs the building HEPA filters, resulting in a ground-level, unfiltered release. The assumed LPF is 1.4×10^{-2} (Smith, Wilkey, and Siebe 1996), consistent with the other analyses. Additionally, it is assumed that the fire does not involve the vault or that the storage canisters in the vault provide adequate protection for the duration of the fire.

K.2 FACILITY ACCIDENT IMPACTS AT HANFORD

The potential source terms and consequences of postulated bounding facility accidents for each facility option at Hanford are presented in Tables K–3 through K–9. Accident scenarios and source terms were developed from data reports prepared for each technology. Consequences were estimated using the MACCS2 computer code and local population and meteorology data. The consequences are presented for mean and 95th percentile meteorological conditions.

Meteorological data are based on 10-m (33-ft) weather readings at Hanford for the 1996 calendar year.[5] In accordance with the MACCS2 format requirements, the data set consists of 8,760 consecutive hourly readings of windspeed, wind direction, Pasquill-Gifford stability class, and accumulated rainfall.

Population estimates for Hanford are for the year 2010, are based on the *Census of Population and Housing, 1990* (DOC 1992), and are identical to the estimates used for the analysis of normal operations in the SPD EIS. Population values are formatted into 16 sectors centered around the 16 standard compass directions, which are further subdivided into 10 radial distance intervals out to 80 km (50 mi).

[5] The choice of calendar year was based primarily on data quality. For some combinations of site and calendar year, the data set contains significant gaps, making that data undesirable for use in dispersion modeling. As a result, not all sites were analyzed using meteorological data for the same calendar year.

Table K–3. Accident Impacts of Pit Conversion Facility in FMEF at Hanford

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Latent Cancer Fatalities[b]**
Fire	1.2×10^{-5}	Unlikely	Mean	2.8×10^{-6}	1.1×10^{-9}	5.2×10^{-7}	2.6×10^{-10}	8.7×10^{-4}	4.3×10^{-7}
			95th percentile	1.1×10^{-5}	4.3×10^{-9}	1.6×10^{-6}	8.1×10^{-10}	5.3×10^{-3}	2.6×10^{-6}
Explosion	3.2×10^{-3}	Unlikely	Mean	7.3×10^{-4}	2.9×10^{-7}	1.4×10^{-4}	6.8×10^{-8}	2.3×10^{-1}	1.1×10^{-4}
			95th percentile	2.8×10^{-3}	1.1×10^{-6}	4.2×10^{-4}	2.1×10^{-7}	1.4	6.8×10^{-4}
Leaks/spills of nuclear material	4.4×10^{-6}	Extremely unlikely	Mean	1.0×10^{-6}	4.1×10^{-10}	1.9×10^{-7}	9.6×10^{-11}	3.2×10^{-4}	1.6×10^{-7}
			95th percentile	3.9×10^{-6}	1.6×10^{-9}	5.9×10^{-7}	3.0×10^{-10}	1.9×10^{-3}	9.5×10^{-7}
Tritium release	2.0×10^{1}	Extremely unlikely	Mean	1.2×10^{-1}	4.7×10^{-5}	2.2×10^{-2}	1.1×10^{-5}	3.7×10^{1}	1.8×10^{-2}
			95th percentile	4.5×10^{-1}	1.8×10^{-4}	6.8×10^{-2}	3.4×10^{-5}	2.2×10^{2}	1.1×10^{-1}
Criticality	1.0×10^{19} Fissions	Extremely unlikely	Mean	1.1×10^{-2}	4.4×10^{-6}	1.2×10^{-3}	6.0×10^{-7}	8.5×10^{-1}	4.3×10^{-4}
			95th percentile	3.3×10^{-2}	1.3×10^{-5}	3.4×10^{-3}	1.7×10^{-6}	5.4	2.7×10^{-3}
Design basis earthquake	3.9×10^{-4}	Unlikely	Mean	9.0×10^{-5}	3.6×10^{-8}	1.7×10^{-5}	8.4×10^{-9}	2.8×10^{-2}	1.4×10^{-5}
			95th percentile	3.5×10^{-4}	1.4×10^{-7}	5.2×10^{-5}	2.6×10^{-8}	1.7×10^{-1}	8.4×10^{-5}
Beyond-design-basis fire	1.7×10^{-2}	Beyond extremely unlikely	Mean	2.9×10^{-2}	1.1×10^{-5}	1.1×10^{-3}	5.6×10^{-7}	1.5	7.7×10^{-4}
			95th percentile	1.1×10^{-1}	4.3×10^{-5}	4.1×10^{-3}	2.0×10^{-6}	9.9	4.9×10^{-3}
Beyond-design-basis earthquake	3.9×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	6.6×10^{1}	2.6×10^{-2}	2.6	1.3×10^{-3}	3.6×10^{3}	1.8
			95th percentile	2.5×10^{2}	9.9×10^{-2}	9.4	4.7×10^{-3}	2.3×10^{4}	11

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: FMEF, Fuels and Materials Examination Facility.

Note: Calculated using the source terms in the pit conversion data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1998a.

Table K–4. Accident Impacts of Ceramic Immobilization Facility in FMEF and HLWVF at Hanford (Hybrid Case)

Accident	Source Term (g)	Frequency (per year)	Meteorology	Impacts on Noninvolved Worker: Dose (rem)	Impacts on Noninvolved Worker: Probability of Cancer Fatality[a]	Impacts at Site Boundary: Dose (rem)	Impacts at Site Boundary: Probability of Cancer Fatality[a]	Impacts of Population Within 80 km: Dose (person-rem)	Impacts of Population Within 80 km: Latent Cancer Fatalities[b]
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	1.1×10^{-2}	4.4×10^{-6}	1.2×10^{-3}	6.0×10^{-7}	8.5×10^{-1}	4.3×10^{-4}
			95th percentile	3.3×10^{-2}	1.3×10^{-5}	3.4×10^{-3}	1.7×10^{-6}	5.4	2.7×10^{-3}
Explosion in HYDOX furnace	3.4×10^{-3}	Unlikely	Mean	1.0×10^{-3}	4.0×10^{-7}	1.9×10^{-4}	9.4×10^{-8}	3.1×10^{-1}	1.6×10^{-4}
			95th percentile	3.8×10^{-3}	1.5×10^{-6}	5.8×10^{-4}	2.9×10^{-7}	1.9	9.4×10^{-4}
Glovebox fire (calcining furnace)	2.7×10^{-7}	Extremely unlikely	Mean	8.0×10^{-8}	3.2×10^{-11}	1.5×10^{-8}	7.4×10^{-12}	2.5×10^{-5}	1.2×10^{-8}
			95th percentile	3.0×10^{-7}	1.2×10^{-10}	4.6×10^{-8}	2.3×10^{-11}	1.5×10^{-4}	7.4×10^{-8}
Hydrogen explosion	3.8×10^{-4}	Unlikely	Mean	1.1×10^{-4}	4.4×10^{-8}	2.1×10^{-5}	1.0×10^{-8}	3.4×10^{-2}	1.7×10^{-5}
			95th percentile	4.2×10^{-4}	1.7×10^{-7}	6.4×10^{-5}	3.2×10^{-8}	2.1×10^{-1}	1.0×10^{-4}
Glovebox fire (sintering furnace)	1.5×10^{-6}	Extremely unlikely	Mean	4.4×10^{-7}	1.8×10^{-10}	8.3×10^{-8}	4.1×10^{-11}	1.4×10^{-4}	6.9×10^{-8}
			95th percentile	1.7×10^{-6}	6.8×10^{-10}	2.6×10^{-7}	1.3×10^{-10}	8.3×10^{-4}	4.1×10^{-7}
Design basis earthquake	3.8×10^{-4}	Unlikely	Mean	1.1×10^{-4}	4.5×10^{-8}	2.1×10^{-5}	1.0×10^{-8}	3.5×10^{-2}	1.7×10^{-5}
			95th percentile	4.3×10^{-4}	1.7×10^{-7}	6.4×10^{-5}	3.2×10^{-8}	2.1×10^{-1}	1.0×10^{-4}
Beyond-design-basis fire	2.1×10^{-3}	Beyond extremely unlikely	Mean	4.5×10^{-3}	1.8×10^{-6}	1.8×10^{-4}	8.9×10^{-8}	2.4×10^{-1}	1.2×10^{-4}
			95th percentile	1.7×10^{-2}	6.8×10^{-6}	6.5×10^{-4}	3.2×10^{-7}	1.6	7.8×10^{-4}
Beyond-design-basis earthquake	1.9×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	4.1×10^{1}	1.6×10^{-2}	1.6	8.1×10^{-4}	2.2×10^{3}	1.1
			95th percentile	1.5×10^{2}	1.6×10^{-2}	5.8	2.9×10^{-3}	1.4×10^{4}	7.1

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: FMEF, Fuels and Materials Examination Facility; HLWVF, high-level-waste vitrification facility, HYDOX, hydride oxidation.

Note: Calculated using the source terms in the immobilization data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1999a.

Table K–5. Accident Impacts of Glass Immobilization Facility in FMEF and HLWVF at Hanford (Hybrid Case)

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Latent Cancer Fatalities[b]**
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	1.1×10^{-2}	4.4×10^{-6}	1.2×10^{-3}	6.0×10^{-7}	8.5×10^{-1}	4.3×10^{-4}
			95th percentile	3.3×10^{-2}	1.3×10^{-5}	3.4×10^{-3}	1.7×10^{-6}	5.4	2.7×10^{-3}
Explosion in HYDOX furnace	3.4×10^{-3}	Unlikely	Mean	1.0×10^{-3}	4.0×10^{-7}	1.9×10^{-4}	9.4×10^{-8}	3.1×10^{-1}	1.6×10^{-4}
			95th percentile	3.8×10^{-3}	1.5×10^{-6}	5.8×10^{-4}	2.9×10^{-7}	1.9	9.4×10^{-4}
Glovebox fire (calcining furnace)	2.7×10^{-7}	Extremely unlikely	Mean	8.0×10^{-8}	3.2×10^{-11}	1.5×10^{-8}	7.4×10^{-12}	2.5×10^{-5}	1.2×10^{-8}
			95th percentile	3.0×10^{-7}	1.2×10^{-10}	4.6×10^{-8}	2.3×10^{-11}	1.5×10^{-4}	7.4×10^{-8}
Hydrogen explosion	3.8×10^{-4}	Unlikely	Mean	1.1×10^{-4}	4.4×10^{-8}	2.1×10^{-5}	1.0×10^{-8}	3.4×10^{-2}	1.7×10^{-5}
			95th percentile	4.2×10^{-4}	1.7×10^{-7}	6.4×10^{-5}	3.2×10^{-8}	2.1×10^{-1}	1.0×10^{-4}
Melter eruption	1.4×10^{-6}	Unlikely	Mean	4.1×10^{-7}	1.6×10^{-10}	7.6×10^{-8}	3.8×10^{-11}	1.3×10^{-4}	6.4×10^{-8}
			95th percentile	1.6×10^{-6}	6.3×10^{-10}	2.4×10^{-7}	1.2×10^{-10}	7.7×10^{-4}	3.8×10^{-7}
Melter spill	3.3×10^{-7}	Unlikely	Mean	9.6×10^{-8}	3.9×10^{-11}	1.8×10^{-8}	9.0×10^{-12}	3.0×10^{-5}	1.5×10^{-8}
			95th percentile	3.7×10^{-7}	1.5×10^{-10}	5.6×10^{-8}	2.8×10^{-11}	1.8×10^{-4}	9.0×10^{-8}
Design basis earthquake	3.3×10^{-4}	Unlikely	Mean	9.7×10^{-5}	3.9×10^{-8}	1.8×10^{-5}	9.1×10^{-9}	3.0×10^{-2}	1.5×10^{-5}
			95th percentile	3.7×10^{-4}	1.5×10^{-7}	5.6×10^{-5}	2.8×10^{-8}	1.8×10^{-1}	9.1×10^{-5}
Beyond-design-basis fire	3.8×10^{-4}	Beyond extremely unlikely	Mean	8.1×10^{-4}	3.3×10^{-7}	3.2×10^{-5}	1.6×10^{-8}	4.4×10^{-2}	2.2×10^{-5}
			95th percentile	3.1×10^{-3}	1.2×10^{-6}	1.2×10^{-4}	5.8×10^{-8}	2.8×10^{-1}	1.4×10^{-4}
Beyond-design-basis earthquake	1.7×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	3.6×10^{1}	1.4×10^{-2}	1.4	7.1×10^{-4}	1.9×10^{3}	9.7×10^{-1}
			95th percentile	1.4×10^{2}	5.4×10^{-2}	5.1	2.6×10^{-3}	1.2×10^{4}	6.2

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: FMEF, Fuels and Materials Examination Facility; HLWVF, high-level-waste vitrification facility; HYDOX, hydride oxidation.

Note: Calculated using the source terms in the immobilization data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1999b.

Table K–6. Accident Impacts of Ceramic Immobilization Facility in FMEF and HLWVF at Hanford (50-t Case)

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Latent Cancer Fatalities[b]**
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	1.1×10^{-2}	4.4×10^{-6}	1.2×10^{-3}	6.0×10^{-7}	8.5×10^{-1}	4.3×10^{-4}
			95th percentile	3.3×10^{-2}	1.3×10^{-5}	3.4×10^{-3}	1.7×10^{-6}	5.4	2.7×10^{-3}
Explosion in HYDOX furnace	3.4×10^{-3}	Unlikely	Mean	1.0×10^{-3}	4.0×10^{-7}	1.9×10^{-4}	9.4×10^{-8}	3.1×10^{-1}	1.6×10^{-4}
			95th percentile	3.8×10^{-3}	1.5×10^{-6}	5.8×10^{-4}	2.9×10^{-7}	1.9	9.4×10^{-4}
Glovebox fire (calcining furnace)	2.7×10^{-7}	Extremely unlikely	Mean	8.0×10^{-8}	3.2×10^{-11}	1.5×10^{-8}	7.4×10^{-12}	2.5×10^{-5}	1.2×10^{-8}
			95th percentile	3.0×10^{-7}	1.2×10^{-10}	4.6×10^{-8}	2.3×10^{-11}	1.5×10^{-4}	7.4×10^{-8}
Hydrogen explosion	3.8×10^{-4}	Unlikely	Mean	1.1×10^{-4}	4.4×10^{-8}	2.1×10^{-5}	1.0×10^{-8}	3.4×10^{-2}	1.7×10^{-5}
			95th percentile	4.2×10^{-4}	1.7×10^{-7}	6.4×10^{-5}	3.2×10^{-8}	2.1×10^{-1}	1.0×10^{-4}
Glovebox fire (sintering furnace)	1.5×10^{-6}	Extremely unlikely	Mean	4.4×10^{-7}	1.8×10^{-10}	8.3×10^{-8}	4.1×10^{-11}	1.4×10^{-4}	6.9×10^{-8}
			95th percentile	1.7×10^{-6}	6.8×10^{-10}	2.6×10^{-7}	1.3×10^{-10}	8.3×10^{-4}	4.1×10^{-7}
Design basis earthquake	3.8×10^{-4}	Unlikely	Mean	1.0×10^{-4}	4.1×10^{-8}	1.9×10^{-5}	9.6×10^{-9}	3.2×10^{-2}	1.6×10^{-5}
			95th percentile	3.9×10^{-4}	1.6×10^{-7}	5.9×10^{-5}	3.0×10^{-8}	1.9×10^{-1}	9.6×10^{-5}
Beyond-design-basis fire	2.1×10^{-3}	Beyond extremely unlikely	Mean	4.5×10^{-3}	1.8×10^{-6}	1.8×10^{-4}	8.9×10^{-8}	2.4×10^{-1}	1.2×10^{-4}
			95th percentile	1.7×10^{-2}	6.8×10^{-6}	6.5×10^{-4}	3.2×10^{-7}	1.6	7.8×10^{-4}
Beyond-design-basis earthquake	1.9×10^{1}	Unlikely to beyond extremely unlikely	Mean	3.8×10^{1}	1.5×10^{-2}	1.5	7.4×10^{-4}	2.0×10^{3}	1.0
			95th percentile	1.4×10^{2}	5.7×10^{-2}	5.4	2.7×10^{-3}	1.3×10^{4}	6.5

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: FMEF, Fuels and Materials Examination Facility; HLWVF, high-level-waste vitrification facility, HYDOX, hydride oxidation.

Note: Calculated using the source terms in the immobilization data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1999a.

Table K–7. Accident Impacts of Glass Immobilization Facility in FMEF and HLWVF at Hanford (50-t Case)

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality**[a]	**Dose (rem)**	**Probability of Cancer Fatality**[a]	**Dose (person-rem)**	**Latent Cancer Fatalities**[b]
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	1.1×10^{-2}	4.4×10^{-6}	1.2×10^{-3}	6.0×10^{-7}	8.5×10^{-1}	4.3×10^{-4}
			95th percentile	3.3×10^{-2}	1.3×10^{-5}	3.4×10^{-3}	1.7×10^{-6}	5.4	2.7×10^{-3}
Explosion in HYDOX furnace	3.4×10^{-3}	Unlikely	Mean	1.0×10^{-3}	4.0×10^{-7}	1.9×10^{-4}	9.4×10^{-8}	3.1×10^{-1}	1.6×10^{-4}
			95th percentile	3.8×10^{-3}	1.5×10^{-6}	5.8×10^{-4}	2.9×10^{-7}	1.9	9.4×10^{-4}
Glovebox fire (calcining furnace)	2.7×10^{-7}	Extremely unlikely	Mean	8.0×10^{-8}	3.2×10^{-11}	1.5×10^{-8}	7.4×10^{-12}	2.5×10^{-5}	1.2×10^{-8}
			95th percentile	3.0×10^{-7}	1.2×10^{-10}	4.6×10^{-8}	2.3×10^{-11}	1.5×10^{-4}	7.4×10^{-8}
Hydrogen explosion	3.8×10^{-4}	Unlikely	Mean	1.1×10^{-4}	4.4×10^{-8}	2.1×10^{-5}	1.0×10^{-8}	3.4×10^{-2}	1.7×10^{-5}
			95th percentile	4.2×10^{-4}	1.7×10^{-7}	6.4×10^{-5}	3.2×10^{-8}	2.1×10^{-1}	1.0×10^{-4}
Melter eruption	1.4×10^{-6}	Unlikely	Mean	4.1×10^{-7}	1.6×10^{-10}	7.6×10^{-8}	3.8×10^{-11}	1.3×10^{-4}	6.4×10^{-8}
			95th percentile	1.6×10^{-6}	6.3×10^{-10}	2.4×10^{-7}	1.2×10^{-10}	7.7×10^{-4}	3.8×10^{-7}
Melter spill	3.3×10^{-7}	Unlikely	Mean	9.6×10^{-8}	3.9×10^{-11}	1.8×10^{-8}	9.0×10^{-12}	3.0×10^{-5}	1.5×10^{-8}
			95th percentile	3.7×10^{-7}	1.5×10^{-10}	5.6×10^{-8}	2.8×10^{-11}	1.8×10^{-4}	9.0×10^{-8}
Design basis earthquake	3.3×10^{-4}	Unlikely	Mean	9.0×10^{-5}	3.6×10^{-8}	1.7×10^{-5}	8.4×10^{-9}	2.8×10^{-2}	1.4×10^{-5}
			95th percentile	3.5×10^{-4}	1.4×10^{-7}	5.2×10^{-5}	2.6×10^{-8}	1.7×10^{-1}	8.4×10^{-5}
Beyond-design-basis fire	3.8×10^{-4}	Beyond extremely unlikely	Mean	8.1×10^{-4}	3.3×10^{-7}	3.2×10^{-5}	1.6×10^{-8}	4.4×10^{-2}	2.2×10^{-5}
			95th percentile	3.1×10^{-3}	1.2×10^{-6}	1.2×10^{-4}	5.8×10^{-8}	2.8×10^{-1}	1.4×10^{-4}
Beyond-design-basis earthquake	1.7×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	3.3×10^{1}	1.3×10^{-2}	1.3	6.6×10^{-4}	1.8×10^{3}	9.0×10^{-1}
			95th percentile	1.3×10^{2}	5.0×10^{-2}	4.8	2.4×10^{-3}	1.2×10^{4}	5.8

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: FMEF, Fuels and Materials Examination Facility; HLWVF, high-level-waste vitrification facility; HYDOX, hydride oxidation.

Note: Calculated using the source terms in the immobilization data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1999b.

Table K–8. Accident Impacts of MOX Facility in FMEF at Hanford

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Latent Cancer Fatalities[b]**
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	5.1×10^{-2}	2.0×10^{-5}	6.5×10^{-3}	3.3×10^{-6}	6.2	3.1×10^{-3}
			95th percentile	1.5×10^{-1}	6.0×10^{-5}	1.9×10^{-2}	9.4×10^{-6}	3.9×10^{1}	1.9×10^{-2}
Explosion in sintering furnace	5.5×10^{-4}	Extremely unlikely	Mean	1.3×10^{-4}	5.1×10^{-8}	2.4×10^{-5}	1.2×10^{-8}	4.0×10^{-2}	2.0×10^{-5}
			95th percentile	4.9×10^{-4}	2.0×10^{-7}	7.4×10^{-5}	3.7×10^{-8}	2.4×10^{-1}	1.2×10^{-4}
Ion exchange exotherm	2.4×10^{-5}	Unlikely	Mean	5.6×10^{-6}	2.2×10^{-9}	1.0×10^{-6}	5.2×10^{-10}	1.7×10^{-3}	8.7×10^{-7}
			95th percentile	2.1×10^{-5}	8.6×10^{-9}	3.2×10^{-6}	1.6×10^{-9}	1.1×10^{-2}	5.2×10^{-6}
Fire	4.0×10^{-6}	Unlikely	Mean	9.3×10^{-7}	3.7×10^{-10}	1.7×10^{-7}	8.7×10^{-11}	2.9×10^{-4}	1.4×10^{-7}
			95th percentile	3.6×10^{-6}	1.4×10^{-9}	5.4×10^{-7}	2.7×10^{-10}	1.8×10^{-3}	8.7×10^{-7}
Spill	5.0×10^{-6}	Extremely unlikely	Mean	1.2×10^{-6}	4.7×10^{-10}	2.2×10^{-7}	1.1×10^{-10}	3.6×10^{-4}	1.8×10^{-7}
			95th percentile	4.5×10^{-6}	1.8×10^{-9}	6.7×10^{-7}	3.4×10^{-10}	2.2×10^{-3}	1.1×10^{-6}
Design basis earthquake	7.9×10^{-5}	Unlikely	Mean	1.8×10^{-5}	7.3×10^{-9}	3.4×10^{-6}	1.7×10^{-9}	5.7×10^{-3}	2.8×10^{-6}
			95th percentile	7.0×10^{-5}	2.8×10^{-8}	1.1×10^{-5}	5.3×10^{-9}	3.4×10^{-2}	1.7×10^{-5}
Beyond-design-basis fire	6.0×10^{-2}	Beyond extremely unlikely	Mean	1.0×10^{-1}	4.1×10^{-5}	4.0×10^{-3}	2.0×10^{-6}	5.5	2.8×10^{-3}
			95th percentile	3.8×10^{-1}	1.5×10^{-4}	1.5×10^{-2}	7.3×10^{-6}	3.5×10^{1}	1.8×10^{-2}
Beyond-design-basis earthquake	9.5×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	1.6×10^{2}	6.5×10^{-2}	6.4	3.2×10^{-3}	8.7×10^{3}	4.4
			95th percentile	6.1×10^{2}	2.4×10^{-1}	2.3×10^{1}	1.2×10^{-2}	5.6×10^{4}	2.8×10^{1}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: FMEF, Fuels and Materials Examination Facility.

Note: Calculated using the source terms in the MOX data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1998b.

Table K–9. Accident Impacts of New MOX Facility at Hanford

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Latent Cancer Fatalities[b]**
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	1.8×10^{-1}	7.2×10^{-5}	9.9×10^{-3}	4.9×10^{-6}	8.2	4.1×10^{-3}
			95th percentile	6.1×10^{-1}	2.5×10^{-4}	3.5×10^{-2}	1.7×10^{-5}	5.5×10^{1}	2.8×10^{-2}
Explosion in sintering furnace	5.5×10^{-4}	Extremely unlikely	Mean	8.0×10^{-4}	3.2×10^{-7}	3.5×10^{-5}	1.8×10^{-8}	5.0×10^{-2}	2.5×10^{-5}
			95th percentile	2.9×10^{-3}	1.2×10^{-6}	1.1×10^{-4}	5.7×10^{-8}	3.2×10^{-1}	1.6×10^{-4}
Ion exchange exotherm	2.4×10^{-5}	Unlikely	Mean	3.5×10^{-5}	1.4×10^{-8}	1.5×10^{-6}	7.7×10^{-10}	2.2×10^{-3}	1.1×10^{-6}
			95th percentile	1.3×10^{-4}	5.1×10^{-8}	5.0×10^{-6}	2.5×10^{-9}	1.4×10^{-2}	7.0×10^{-6}
Fire	4.0×10^{-6}	Unlikely	Mean	5.8×10^{-6}	2.3×10^{-9}	2.6×10^{-7}	1.3×10^{-10}	3.6×10^{-4}	1.8×10^{-7}
			95th percentile	2.1×10^{-5}	8.4×10^{-9}	8.3×10^{-7}	4.2×10^{-10}	2.3×10^{-3}	1.2×10^{-6}
Spill	5.0×10^{-6}	Extremely unlikely	Mean	7.3×10^{-6}	2.9×10^{-9}	3.2×10^{-7}	1.6×10^{-10}	4.5×10^{-4}	2.3×10^{-7}
			95th percentile	2.6×10^{-5}	1.1×10^{-8}	1.0×10^{-6}	5.2×10^{-10}	2.9×10^{-3}	1.5×10^{-6}
Design basis earthquake	7.9×10^{-5}	Unlikely	Mean	1.1×10^{-4}	4.6×10^{-8}	5.0×10^{-6}	2.5×10^{-9}	7.1×10^{-3}	3.6×10^{-6}
			95th percentile	4.1×10^{-4}	1.7×10^{-7}	1.6×10^{-5}	8.2×10^{-9}	4.6×10^{-2}	2.3×10^{-5}
Beyond-design-basis fire	6.0×10^{-2}	Beyond extremely unlikely	Mean	1.0×10^{-1}	4.1×10^{-5}	4.0×10^{-3}	2.0×10^{-6}	5.5	2.8×10^{-3}
			95th percentile	3.8×10^{-1}	1.5×10^{-4}	1.5×10^{-2}	7.3×10^{-6}	3.5×10^{1}	1.8×10^{-2}
Beyond-design-basis earthquake	9.5×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	1.6×10^{2}	6.5×10^{-2}	6.4	3.2×10^{-3}	8.7×10^{3}	4.4
			95th percentile	6.1×10^{2}	2.4×10^{-1}	2.3×10^{1}	1.2×10^{-2}	5.6×10^{4}	2.8×10^{1}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Note: Calculated using the source terms in the MOX data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1998b.

K.3 FACILITY ACCIDENT IMPACTS AT INEEL

The potential source terms and consequences of postulated bounding facility accidents for each facility option for INEEL are presented in Tables K–10 and K–11. Accident scenarios and source terms were developed from data reports prepared for each technology. Consequences were estimated using the MACCS2 computer code and local population and meteorology data. The consequences are presented for mean and 95th percentile meteorological conditions.

Meteorological data are based on 10-m (33-ft) weather readings at INEEL for the 1993 calendar year.[6] In accordance with MACCS2 format requirements, the data set consists of 8,760 consecutive hourly readings of windspeed, wind direction, Pasquill-Gifford stability class, and accumulated rainfall.

Population estimates for INEEL are for the year 2010, are based on the *Census of Population and Housing, 1990* (DOC 1992), and are identical to the estimates used for the analysis of normal operations in the SPD EIS. Population values are formatted into 16 sectors centered around the 16 standard compass directions, which are further subdivided into 10 radial distance intervals out to 80 km (50 mi).

[6] The choice of calendar year was based primarily on data quality. For some combinations of site and calendar year, the data set contains significant gaps, making that data undesirable for use in dispersion modeling. As a result, not all sites were analyzed using meteorological data for the same calendar year.

Table K–10. Accident Impacts of Pit Conversion Facility in FPF at INEEL

Accident	Source Term (g)	Frequency (per year)	Meteorology	Impacts on Noninvolved Worker: Dose (rem)	Impacts on Noninvolved Worker: Probability of Cancer Fatality[a]	Impacts at Site Boundary: Dose (rem)	Impacts at Site Boundary: Probability of Cancer Fatality[a]	Impacts on Population Within 80 km: Dose (person-rem)	Impacts on Population Within 80 km: Latent Cancer Fatalities[b]
Fire	1.2×10^{-5}	Unlikely	Mean	2.5×10^{-6}	1.0×10^{-9}	3.0×10^{-7}	1.5×10^{-10}	5.6×10^{-5}	2.8×10^{-8}
			95th percentile	6.4×10^{-6}	2.5×10^{-9}	1.1×10^{-6}	5.3×10^{-10}	2.1×10^{-4}	1.0×10^{-7}
Explosion	3.2×10^{-3}	Unlikely	Mean	6.5×10^{-4}	2.6×10^{-7}	7.8×10^{-5}	3.9×10^{-8}	1.5×10^{-2}	7.4×10^{-6}
			95th percentile	1.7×10^{-3}	6.7×10^{-7}	2.8×10^{-4}	1.4×10^{-7}	5.5×10^{-2}	2.7×10^{-5}
Leaks/spills of nuclear material	4.4×10^{-6}	Extremely unlikely	Mean	9.1×10^{-7}	3.6×10^{-10}	1.1×10^{-7}	5.4×10^{-11}	2.1×10^{-5}	1.0×10^{-8}
			95th percentile	2.3×10^{-6}	9.3×10^{-10}	3.9×10^{-7}	1.9×10^{-10}	7.7×10^{-5}	3.8×10^{-8}
Tritium release	2.0×10^{1}	Extremely unlikely	Mean	1.0×10^{-1}	4.2×10^{-5}	1.2×10^{-2}	6.2×10^{-6}	2.4	1.2×10^{-3}
			95th percentile	2.7×10^{-1}	1.1×10^{-4}	4.5×10^{-2}	2.2×10^{-5}	8.8	4.4×10^{-3}
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	1.1×10^{-2}	4.4×10^{-6}	4.8×10^{-4}	2.4×10^{-7}	2.2×10^{-2}	1.1×10^{-5}
			95th percentile	3.3×10^{-2}	1.3×10^{-5}	1.6×10^{-3}	7.9×10^{-7}	8.5×10^{-2}	4.2×10^{-5}
Design basis earthquake	3.9×10^{-4}	Unlikely	Mean	8.0×10^{-5}	3.2×10^{-8}	9.5×10^{-6}	4.8×10^{-9}	1.8×10^{-3}	9.1×10^{-7}
			95th percentile	2.1×10^{-4}	8.2×10^{-8}	3.4×10^{-5}	1.7×10^{-8}	6.8×10^{-3}	3.4×10^{-6}
Beyond-design-basis fire	1.7×10^{-2}	Beyond extremely unlikely	Mean	3.0×10^{-2}	1.2×10^{-5}	8.1×10^{-4}	4.1×10^{-7}	9.6×10^{-2}	4.8×10^{-5}
			95th percentile	1.1×10^{-1}	4.5×10^{-5}	2.9×10^{-3}	1.5×10^{-6}	3.6×10^{-1}	1.8×10^{-4}
Beyond-design-basis earthquake	3.9×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	7.0×10^{1}	2.8×10^{-2}	1.9	9.3×10^{-4}	2.2×10^{2}	1.1×10^{-1}
			95th percentile	2.6×10^{2}	1.0×10^{-1}	6.7	3.3×10^{-3}	8.4×10^{2}	4.2×10^{-1}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 mi] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: FPF, Fuel Processing Facility.

Note: Calculated using the source terms in the pit conversion data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1998f.

Table K–11. Accident Impacts of New MOX Facility at INEEL

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Latent Cancer Fatalities[b]**
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	1.9×10^{-1}	7.4×10^{-5}	4.3×10^{-3}	2.1×10^{-6}	2.7×10^{-1}	1.4×10^{-4}
			95th percentile	7.5×10^{-1}	3.0×10^{-4}	1.6×10^{-2}	8.2×10^{-6}	1.0	5.2×10^{-4}
Explosion in sintering furnace	5.5×10^{-4}	Extremely unlikely	Mean	8.3×10^{-4}	3.3×10^{-7}	2.2×10^{-5}	1.1×10^{-8}	3.1×10^{-3}	1.5×10^{-6}
			95th percentile	3.6×10^{-3}	1.4×10^{-6}	8.4×10^{-5}	4.2×10^{-8}	1.2×10^{-2}	5.8×10^{-6}
Ion exchange exotherm	2.4×10^{-5}	Unlikely	Mean	3.6×10^{-5}	1.4×10^{-8}	9.5×10^{-7}	4.8×10^{-10}	1.3×10^{-4}	6.7×10^{-8}
			95th percentile	1.6×10^{-4}	6.3×10^{-8}	3.7×10^{-6}	1.8×10^{-9}	5.1×10^{-4}	2.5×10^{-7}
Fire	4.0×10^{-6}	Unlikely	Mean	6.0×10^{-6}	2.4×10^{-9}	1.6×10^{-7}	7.9×10^{-11}	2.2×10^{-5}	1.1×10^{-8}
			95th percentile	2.6×10^{-5}	1.0×10^{-8}	6.1×10^{-7}	3.1×10^{-10}	8.5×10^{-5}	4.2×10^{-8}
Spill	5.0×10^{-6}	Extremely unlikely	Mean	7.5×10^{-6}	3.0×10^{-9}	2.0×10^{-7}	9.9×10^{-11}	2.8×10^{-5}	1.4×10^{-8}
			95th percentile	3.3×10^{-5}	1.3×10^{-8}	7.7×10^{-7}	3.8×10^{-10}	1.1×10^{-4}	5.3×10^{-8}
Design basis earthquake	7.9×10^{-5}	Unlikely	Mean	1.2×10^{-4}	4.7×10^{-8}	3.1×10^{-6}	1.6×10^{-9}	4.4×10^{-4}	2.2×10^{-7}
			95th percentile	5.1×10^{-4}	2.1×10^{-7}	1.2×10^{-5}	6.0×10^{-9}	1.7×10^{-3}	8.3×10^{-7}
Beyond-design-basis fire	6.0×10^{-2}	Beyond extremely unlikely	Mean	1.1×10^{-1}	4.3×10^{-5}	2.9×10^{-3}	1.4×10^{-6}	3.4×10^{-1}	1.7×10^{-4}
			95th percentile	4.1×10^{-1}	1.6×10^{-4}	1.0×10^{-2}	5.2×10^{-6}	1.3	6.5×10^{-4}
Beyond-design-basis earthquake	9.5×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	1.7×10^{2}	6.8×10^{-2}	4.6	2.3×10^{-3}	5.4×10^{2}	2.7×10^{-1}
			95th percentile	6.5×10^{2}	2.6×10^{-1}	1.6×10^{1}	8.2×10^{-3}	2.1×10^{3}	1.0
			95th percentile						

a Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

b Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Note: Calculated using the source terms in the MOX data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1998g.

K.4 FACILITY ACCIDENT IMPACTS AT PANTEX

The potential source terms and consequences of postulated bounding facility accidents for each facility option for Pantex are presented in Tables K–12 and K–13. Accident scenarios and source terms were developed from data reports prepared for each technology. Consequences were estimated using the MACCS2 computer code and local population and meteorology data. The consequences are presented for mean and 95th percentile meteorological conditions.

Meteorological data are based on 10-m (33-ft) weather readings from the Pantex Tower for the 1996 calendar year.[7] In accordance with MACCS2 format requirements, the data set consists of 8,760 consecutive hourly readings of windspeed, wind direction, Pasquill-Gifford stability class, and accumulated rainfall.

Population estimates for Pantex are for the year 2010, are based on the *Census of Population and Housing, 1990* (DOC 1992), and are identical to the estimates used for the analysis of normal operations in the SPD EIS. Population values are formatted into 16 sectors centered around the 16 standard compass directions, which are further subdivided into 10 radial distance intervals out to 80 km (50 mi).

[7] The choice of calendar year was based primarily on data quality. For some combinations of site and calendar year, the data set contains significant gaps, making that data undesirable for use in dispersion modeling. As a result, not all sites were analyzed using meteorological data for the same calendar year.

Table K–12. Accident Impacts of New Pit Conversion Facility at Pantex

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	Source Term (g)	Frequency (per year)	Meteorology	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]
Fire	1.2×10^{-5}	Unlikely	Mean	2.3×10^{-6}	9.1×10^{-10}	7.6×10^{-7}	3.8×10^{-10}	1.8×10^{-4}	9.1×10^{-8}
			95th percentile	5.2×10^{-6}	2.1×10^{-9}	2.1×10^{-6}	1.0×10^{-9}	8.6×10^{-4}	4.3×10^{-7}
Explosion	3.2×10^{-3}	Unlikely	Mean	6.0×10^{-4}	2.4×10^{-7}	2.0×10^{-4}	9.9×10^{-8}	4.8×10^{-2}	2.4×10^{-5}
			95th percentile	1.4×10^{-3}	5.4×10^{-7}	5.4×10^{-4}	2.7×10^{-7}	2.2×10^{-1}	1.1×10^{-4}
Leaks/spills of nuclear material	4.4×10^{-6}	Extremely unlikely	Mean	8.4×10^{-7}	3.3×10^{-10}	2.8×10^{-7}	1.4×10^{-10}	6.7×10^{-5}	3.3×10^{-8}
			95th percentile	1.9×10^{-6}	7.6×10^{-10}	7.6×10^{-7}	3.8×10^{-10}	3.1×10^{-4}	1.6×10^{-7}
Tritium release	2.0×10^{1}	Extremely unlikely	Mean	9.6×10^{-2}	3.8×10^{-5}	3.2×10^{-2}	1.6×10^{-5}	7.7	3.8×10^{-3}
			95th percentile	2.2×10^{-1}	8.7×10^{-5}	8.7×10^{-2}	4.4×10^{-5}	3.6×10^{1}	1.8×10^{-2}
Criticality	1.0×10^{19} Fissions	Extremely unlikely	Mean	6.1×10^{-3}	2.5×10^{-6}	2.7×10^{-3}	1.3×10^{-6}	2.7×10^{-1}	1.4×10^{-4}
			95th percentile	1.5×10^{-2}	6.0×10^{-6}	6.0×10^{-3}	3.0×10^{-6}	1.6	7.9×10^{-4}
Design basis earthquake	3.9×10^{-4}	Unlikely	Mean	7.4×10^{-5}	2.9×10^{-8}	2.4×10^{-5}	1.2×10^{-8}	5.9×10^{-3}	2.9×10^{-6}
			95th percentile	1.7×10^{-4}	6.7×10^{-8}	6.7×10^{-5}	3.3×10^{-8}	2.8×10^{-2}	1.4×10^{-5}
Beyond-design-basis fire	1.7×10^{-2}	Beyond extremely unlikely	Mean	9.6×10^{-3}	3.8×10^{-6}	1.5×10^{-3}	7.5×10^{-7}	2.8×10^{-1}	1.4×10^{-4}
			95th percentile	2.8×10^{-2}	1.1×10^{-5}	4.4×10^{-3}	2.2×10^{-6}	1.3	6.3×10^{-4}
Beyond-design-basis earthquake	3.9×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	2.2×10^{1}	8.8×10^{-3}	3.5	1.7×10^{-3}	6.4×10^{2}	3.2×10^{-1}
			95th percentile	6.4×10^{1}	2.6×10^{-2}	1.0×10^{1}	5.1×10^{-3}	3.0×10^{3}	1.5
Aircraft crash	1.2×10^{2}	Beyond extremely unlikely	Mean	6.8×10^{1}	2.7×10^{-2}	1.1×10^{1}	5.4×10^{-3}	2.0×10^{3}	1.0
			95th percentile	2.0×10^{2}	7.9×10^{-2}	3.1×10^{1}	1.6×10^{-2}	9.2×10^{3}	4.5

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Note: Calculated using the source terms in the pit conversion data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1998e.

Table K–13. Accident Impacts of New MOX Facility at Pantex

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Latent Cancer Fatalities[b]**
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	7.5×10^{-2}	3.0×10^{-5}	1.9×10^{-2}	9.3×10^{-6}	1.9	9.4×10^{-4}
			95th percentile	2.4×10^{-1}	9.5×10^{-5}	4.7×10^{-2}	2.3×10^{-5}	1.1×10^{1}	5.4×10^{-3}
Explosion in sintering furnace	5.5×10^{-4}	Extremely unlikely	Mean	2.8×10^{-4}	1.1×10^{-7}	4.8×10^{-5}	2.4×10^{-8}	9.1×10^{-3}	4.5×10^{-6}
			95th percentile	8.9×10^{-4}	3.5×10^{-7}	1.3×10^{-4}	6.6×10^{-8}	4.2×10^{-2}	2.1×10^{-5}
Ion exchange exotherm	2.4×10^{-5}	Unlikely	Mean	1.2×10^{-5}	5.0×10^{-9}	2.1×10^{-6}	1.0×10^{-9}	4.0×10^{-4}	2.0×10^{-7}
			95th percentile	3.9×10^{-5}	1.5×10^{-8}	5.8×10^{-6}	2.9×10^{-9}	1.8×10^{-3}	9.0×10^{-7}
Fire	4.0×10^{-6}	Unlikely	Mean	2.1×10^{-6}	8.3×10^{-10}	3.5×10^{-7}	1.7×10^{-10}	6.6×10^{-5}	3.3×10^{-8}
			95th percentile	6.4×10^{-6}	2.6×10^{-9}	9.6×10^{-7}	4.8×10^{-10}	3.0×10^{-4}	1.5×10^{-7}
Spill	5.0×10^{-6}	Extremely unlikely	Mean	2.6×10^{-6}	1.0×10^{-9}	4.4×10^{-7}	2.2×10^{-10}	8.3×10^{-5}	4.1×10^{-8}
			95th percentile	8.1×10^{-6}	3.2×10^{-9}	1.2×10^{-6}	6.0×10^{-10}	3.8×10^{-4}	1.9×10^{-7}
Design basis earthquake	7.9×10^{-5}	Unlikely	Mean	4.1×10^{-5}	1.6×10^{-8}	6.8×10^{-6}	3.4×10^{-9}	1.3×10^{-3}	6.5×10^{-7}
			95th percentile	1.3×10^{-4}	5.1×10^{-8}	1.9×10^{-5}	9.4×10^{-9}	5.9×10^{-3}	3.0×10^{-6}
Beyond-design-basis fire	6.0×10^{-2}	Beyond extremely unlikely	Mean	3.4×10^{-2}	1.4×10^{-5}	5.4×10^{-3}	2.7×10^{-6}	1.0	5.0×10^{-4}
			95th percentile	9.9×10^{-2}	4.0×10^{-5}	1.6×10^{-2}	7.8×10^{-6}	4.6	2.3×10^{-3}
Beyond-design-basis earthquake	9.5×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	5.4×10^{1}	2.2×10^{-2}	8.5	4.3×10^{-3}	1.6×10^{3}	7.9×10^{-1}
			95th percentile	1.6×10^{2}	6.3×10^{-2}	2.5×10^{1}	1.2×10^{-2}	7.3×10^{3}	3.6
Aircraft crash	7.1×10^{2}	Beyond extremely unlikely	Mean	4.0×10^{2}	1.6×10^{-1}	6.3×10^{1}	3.2×10^{-2}	1.2×10^{4}	5.9
			95th percentile	1.2×10^{3}	4.7×10^{-1}	1.9×10^{2}	9.3×10^{-2}	5.4×10^{4}	2.7×10^{1}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Note: Calculated using the source terms in the MOX data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1998h.

K.5 FACILITY ACCIDENT IMPACTS AT SRS

The potential source terms and consequences of postulated bounding facility accidents for each facility option for SRS are presented in Tables K–14 through K–19. Accident scenarios and source terms were developed from data reports prepared for each technology. Consequences were estimated using the MACCS2 computer code and local population and meteorology data. The consequences are presented for both mean and 95th percentile meteorological conditions.

Meteorological data are based on 10-m (33-ft) weather readings at SRS, are identical to the data used in *F-Canyon Plutonium Solutions Environmental Impact Statement*, and included in Sample Problem D of the MACCS2 User's Guide (Chanin and Young 1997:4-4). In accordance with MACCS2 format requirements, the data set consists of 8,760 consecutive hourly readings of windspeed, wind direction, Pasquill-Gifford stability class, and accumulated rainfall.

Population estimates for SRS are for the year 2010, are based on the *Census of Population and Housing, 1990* (DOC 1992), and are identical to the estimates used for the analysis of normal operations in the SPD EIS. Population values are formatted into 16 sectors centered around the 16 standard compass directions, which are further subdivided into 10 radial distance intervals out to 80 km (50 mi).

[Tables deleted.]

Table K–14. Accident Impacts of New Pit Conversion Facility at SRS

Accident	Source Term (g)	Frequency (per year)	Meteorology	Impacts on Noninvolved Worker: Dose (rem)	Impacts on Noninvolved Worker: Probability of Cancer Fatality[a]	Impacts at Site Boundary: Dose (rem)	Impacts at Site Boundary: Proability of Cancer Fatality[a]	Impacts on Population Within 80 km: Dose (person-rem)	Impacts on Population Within 80 km: Latent Cancer Fatalities[b]
Fire	1.2×10^{-5}	Unlikely	Mean	2.6×10^{-6}	1.1×10^{-9}	2.1×10^{-7}	1.0×10^{-10}	5.4×10^{-4}	2.7×10^{-7}
			95th percentile	6.2×10^{-6}	2.5×10^{-9}	6.7×10^{-7}	3.3×10^{-10}	2.4×10^{-3}	1.2×10^{-6}
Explosion	3.2×10^{-3}	Unlikely	Mean	6.9×10^{-4}	2.8×10^{-7}	5.4×10^{-5}	2.7×10^{-8}	1.4×10^{-1}	7.0×10^{-5}
			95th percentile	1.6×10^{-3}	6.5×10^{-7}	1.8×10^{-4}	8.8×10^{-8}	6.2×10^{-1}	3.1×10^{-4}
Leaks/spills of nuclear material	4.4×10^{-6}	Extremely unlikely	Mean	9.6×10^{-7}	3.9×10^{-10}	7.5×10^{-8}	3.8×10^{-11}	2.0×10^{-4}	9.8×10^{-8}
			95th percentile	2.3×10^{-6}	9.1×10^{-10}	2.5×10^{-7}	1.2×10^{-10}	8.7×10^{-4}	4.3×10^{-7}
Tritium release	2.0×10^{1}	Extremely unlikely	Mean	1.1×10^{-1}	4.4×10^{-5}	8.6×10^{-3}	4.3×10^{-6}	2.3×10^{1}	1.1×10^{-2}
			95th percentile	2.6×10^{-1}	1.0×10^{-4}	2.8×10^{-2}	1.4×10^{-5}	1.0×10^{2}	5.0×10^{-2}
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	7.9×10^{-3}	3.2×10^{-6}	5.8×10^{-4}	2.9×10^{-7}	4.2×10^{-1}	2.1×10^{-4}
			95th percentile	1.7×10^{-2}	6.7×10^{-6}	1.8×10^{-3}	9.2×10^{-7}	1.8	9.0×10^{-4}
Design basis earthquake	3.9×10^{-4}	Unlikely	Mean	8.5×10^{-5}	3.4×10^{-8}	6.6×10^{-6}	3.3×10^{-9}	1.7×10^{-2}	8.6×10^{-6}
			95th percentile	2.0×10^{-4}	8.0×10^{-8}	2.2×10^{-5}	1.1×10^{-8}	7.7×10^{-2}	3.8×10^{-5}
Beyond-design-basis fire	1.7×10^{-2}	Beyond extremely unlikely	Mean	1.1×10^{-2}	4.4×10^{-6}	4.8×10^{-4}	2.4×10^{-7}	8.8×10^{-1}	4.4×10^{-4}
			95th percentile	4.0×10^{-2}	1.6×10^{-5}	1.6×10^{-3}	7.8×10^{-7}	3.7	1.9×10^{-3}
Beyond-design-basis earthquake	3.9×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	2.5×10^{1}	1.0×10^{-2}	1.1	5.5×10^{-4}	2.0×10^{3}	1.0
			95th percnetile	9.2×10^{1}	3.7×10^{-2}	3.6	1.8×10^{-3}	8.5×10^{3}	4.3

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] (or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Note: Calculated using the source terms in the pit conversion data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1998c.

Table K–15. Accident Impacts of Ceramic Immobilization Facility in New Construction and DWPF at SRS (Hybrid Case)

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Latent Cancer Fatalities[b]**
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	5.3×10^{-3}	2.1×10^{-6}	4.6×10^{-4}	2.3×10^{-7}	3.5×10^{-1}	1.8×10^{-4}
			95th percentile	1.0×10^{-2}	4.2×10^{-6}	1.6×10^{-3}	7.8×10^{-7}	1.5	7.5×10^{-4}
Explosion in HYDOX furnace	3.4×10^{-3}	Unlikely	Mean	3.9×10^{-4}	1.6×10^{-7}	5.3×10^{-5}	2.7×10^{-8}	1.6×10^{-1}	7.8×10^{-5}
			95th percentile	8.6×10^{-4}	3.4×10^{-7}	1.6×10^{-4}	8.1×10^{-8}	7.1×10^{-1}	3.5×10^{-4}
Glovebox fire (calcining furnace)	2.7×10^{-7}	Extremely unlikely	Mean	3.1×10^{-8}	1.2×10^{-11}	4.2×10^{-9}	2.1×10^{-12}	1.2×10^{-5}	6.2×10^{-9}
			95th percentile	6.8×10^{-8}	2.7×10^{-11}	1.3×10^{-8}	6.5×10^{-12}	5.6×10^{-5}	2.8×10^{-8}
Hydrogen explosion	3.8×10^{-4}	Unlikely	Mean	4.3×10^{-5}	1.7×10^{-8}	5.9×10^{-6}	2.9×10^{-9}	1.7×10^{-2}	8.6×10^{-6}
			95th percentile	9.5×10^{-5}	3.8×10^{-8}	1.8×10^{-5}	9.0×10^{-9}	7.8×10^{-2}	3.8×10^{-5}
Glovebox fire (sintering furnace)	1.5×10^{-6}	Extremely unlikely	Mean	1.7×10^{-7}	6.9×10^{-11}	2.4×10^{-8}	1.2×10^{-11}	6.9×10^{-5}	3.4×10^{-8}
			95th percentile	3.8×10^{-7}	1.5×10^{-10}	7.2×10^{-8}	3.6×10^{-11}	3.1×10^{-4}	1.5×10^{-7}
Design basis earthquake	3.8×10^{-4}	Unlikely	Mean	4.4×10^{-5}	1.7×10^{-8}	5.9×10^{-6}	3.0×10^{-9}	1.7×10^{-2}	8.7×10^{-6}
			95th percentile	9.6×10^{-5}	3.8×10^{-8}	1.8×10^{-5}	9.1×10^{-9}	7.9×10^{-2}	3.9×10^{-5}
Beyond-design-basis fire	2.1×10^{-3}	Beyond extremely unlikely	Mean	1.7×10^{-3}	6.9×10^{-7}	7.6×10^{-5}	3.8×10^{-8}	1.4×10^{-1}	7.0×10^{-5}
			95th percentile	6.3×10^{-3}	2.5×10^{-6}	2.5×10^{-4}	1.2×10^{-7}	5.8×10^{-1}	2.9×10^{-4}
Beyond-design-basis earthquake	1.9×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	1.6×10^{1}	6.3×10^{-3}	6.8×10^{-1}	3.4×10^{-4}	1.3×10^{3}	6.3×10^{-1}
			95th percentile	5.7×10^{1}	2.3×10^{-2}	2.2	1.1×10^{-3}	5.3×10^{3}	2.7

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: DWPF, Defense Waste Processing Facility; HYDOX, hydride oxidation.

Note: Calculated using the source terms in the immobilization data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1999c.

Table K–16. Accident Impacts of Glass Immobilization Facility in New Construction and DWPF at SRS (Hybrid Case)

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	Source Term (g)	Frequency (per year)	Meteorology	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	5.3×10^{-3}	2.1×10^{-6}	4.6×10^{-4}	2.3×10^{-7}	3.5×10^{-1}	1.8×10^{-4}
			95th percentile	1.0×10^{-2}	4.2×10^{-6}	1.6×10^{-3}	7.8×10^{-7}	1.5	7.5×10^{-4}
Explosion in HYDOX furnace	3.4×10^{-3}	Unlikely	Mean	3.9×10^{-4}	1.6×10^{-7}	5.3×10^{-5}	2.7×10^{-8}	1.6×10^{-1}	7.8×10^{-5}
			95th percentile	8.6×10^{-4}	3.4×10^{-7}	1.6×10^{-4}	8.1×10^{-8}	7.1×10^{-1}	3.5×10^{-4}
Glovebox fire (calcining furnace)	2.7×10^{-7}	Extremely unlikely	Mean	3.1×10^{-8}	1.2×10^{-11}	4.2×10^{-9}	2.1×10^{-12}	1.2×10^{-5}	6.2×10^{-9}
			95th percentile	6.8×10^{-8}	2.7×10^{-11}	1.3×10^{-8}	6.5×10^{-12}	5.6×10^{-5}	2.8×10^{-8}
Hydrogen explosion	3.8×10^{-4}	Unlikely	Mean	4.3×10^{-5}	1.7×10^{-8}	5.9×10^{-6}	2.9×10^{-9}	1.7×10^{-2}	8.6×10^{-6}
			95th percentile	9.5×10^{-5}	3.8×10^{-8}	1.8×10^{-5}	9.0×10^{-9}	7.8×10^{-2}	3.8×10^{-5}
Melter eruption	1.4×10^{-6}	Unlikely	Mean	1.6×10^{-7}	6.4×10^{-11}	2.2×10^{-8}	1.1×10^{-11}	6.4×10^{-5}	3.2×10^{-8}
			95th percentile	3.5×10^{-7}	1.4×10^{-10}	6.7×10^{-8}	3.3×10^{-11}	2.9×10^{-4}	1.4×10^{-7}
Melter spill	3.3×10^{-7}	Unlikely	Mean	3.8×10^{-8}	1.5×10^{-11}	5.1×10^{-9}	2.6×10^{-12}	1.5×10^{-5}	7.5×10^{-9}
			95th percentile	8.3×10^{-8}	3.3×10^{-11}	1.6×10^{-8}	7.8×10^{-12}	6.8×10^{-5}	3.3×10^{-8}
Design basis earthquake	3.3×10^{-4}	Unlikely	Mean	3.8×10^{-5}	1.5×10^{-8}	5.2×10^{-6}	2.6×10^{-9}	1.5×10^{-2}	7.6×10^{-6}
			95th percentile	8.3×10^{-5}	3.3×10^{-8}	1.6×10^{-5}	7.9×10^{-9}	6.9×10^{-2}	3.4×10^{-5}
Beyond-design-basis fire	3.8×10^{-4}	Beyond extremely unlikely	Mean	3.1×10^{-4}	1.2×10^{-7}	1.4×10^{-5}	6.8×10^{-9}	2.5×10^{-2}	1.3×10^{-5}
			95th percentile	1.1×10^{-3}	4.6×10^{-7}	4.4×10^{-5}	2.2×10^{-8}	1.0×10^{-1}	5.3×10^{-5}
Beyond-design-basis earthquake	1.7×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	1.4×10^{1}	5.5×10^{-3}	6.0×10^{-1}	3.0×10^{-4}	1.1×10^{3}	5.5×10^{-1}
			95th percentile	5.0×10^{1}	2.0×10^{-2}	2.0	9.8×10^{-4}	4.6×10^{3}	2.3

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: DWPF, Defense Waste Processing Facility; HYDOX, hydride oxidation.

Note: Calculated using the source terms in the immobilization data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1999d.

Table K–17. Accident Impacts of Ceramic Immobilization Facility in New Construction and DWPF at SRS (50-t Case)

Accident	Source Term (g)	Frequency (per year)	Meteorology	Impacts on Noninvolved Worker: Dose (rem)	Impacts on Noninvolved Worker: Probability of Cancer Fatality[a]	Impacts at Site Boundary: Dose (rem)	Impacts at Site Boundary: Probability of Cancer Fatality[a]	Impacts on Population Within 80 km: Dose (person-rem)	Impacts on Population Within 80 km: Latent Cancer Fatalities[b]
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	5.3×10^{-3}	2.1×10^{-6}	4.6×10^{-4}	2.3×10^{-7}	3.5×10^{-1}	1.8×10^{-4}
			95th percentile	1.0×10^{-2}	4.2×10^{-6}	1.6×10^{-3}	7.8×10^{-7}	1.5	7.5×10^{-4}
Explosion in HYDOX furnace	3.4×10^{-3}	Unlikely	Mean	3.9×10^{-4}	1.6×10^{-7}	5.3×10^{-5}	2.7×10^{-8}	1.6×10^{-1}	7.8×10^{-5}
			95th percentile	8.6×10^{-4}	3.4×10^{-7}	1.6×10^{-4}	8.1×10^{-8}	7.1×10^{-1}	3.5×10^{-4}
Glovebox fire (calcining furnace)	2.7×10^{-7}	Extremely unlikely	Mean	3.1×10^{-8}	1.2×10^{-11}	4.2×10^{-9}	2.1×10^{-12}	1.2×10^{-5}	6.2×10^{-9}
			95th percentile	6.8×10^{-8}	2.7×10^{-11}	1.3×10^{-8}	6.5×10^{-12}	5.6×10^{-5}	2.8×10^{-8}
Hydrogen explosion	3.8×10^{-4}	Unlikely	Mean	4.3×10^{-5}	1.7×10^{-8}	5.9×10^{-6}	2.9×10^{-9}	1.7×10^{-2}	8.6×10^{-6}
			95th percentile	9.5×10^{-5}	3.8×10^{-8}	1.8×10^{-5}	9.0×10^{-9}	7.8×10^{-2}	3.8×10^{-5}
Glovebox fire (sintering furnace)	1.5×10^{-6}	Extremely unlikely	Mean	1.7×10^{-7}	6.9×10^{-11}	2.4×10^{-8}	1.2×10^{-11}	6.9×10^{-5}	3.4×10^{-8}
			95th percentile	3.8×10^{-7}	1.5×10^{-10}	7.2×10^{-8}	3.6×10^{-11}	3.1×10^{-4}	1.5×10^{-7}
Design basis earthquake	3.8×10^{-4}	Unlikely	Mean	4.0×10^{-5}	1.6×10^{-8}	5.5×10^{-6}	2.7×10^{-9}	1.6×10^{-2}	8.0×10^{-6}
			95th percentile	8.8×10^{-5}	3.5×10^{-8}	1.7×10^{-5}	8.3×10^{-9}	7.2×10^{-2}	3.6×10^{-5}
Beyond-design-basis fire	2.1×10^{-3}	Beyond extremely unlikely	Mean	1.7×10^{-3}	6.9×10^{-7}	7.6×10^{-5}	3.8×10^{-8}	1.4×10^{-1}	7.0×10^{-5}
			95th percentile	6.3×10^{-3}	2.5×10^{-6}	2.5×10^{-4}	1.2×10^{-7}	5.8×10^{-1}	2.9×10^{-4}
Beyond-design-basis earthquake	1.9×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	1.4×10^{1}	5.7×10^{-3}	6.3×10^{-1}	3.1×10^{-4}	1.2×10^{3}	5.8×10^{-1}
			95th percentile	5.3×10^{1}	2.1×10^{-2}	2.1	1.0×10^{-3}	4.8×10^{3}	2.5

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: DWPF, Defense Waste Processing Facility; HYDOX, hydride oxidation.

Note: Calculated using the source terms in the immobilization data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1999c.

Table K–18. Accident Impacts of Glass Immobilization Facility in New Construction and DWPF at SRS (50-t Case)

Accident	Source Term (g)	Frequency (per year)	Meteorology	Impacts on Noninvolved Worker: Dose (rem)	Impacts on Noninvolved Worker: Probability of Cancer Fatality[a]	Impacts at Site Boundary: Dose (rem)	Impacts at Site Boundary: Probability of Cancer Fatality[a]	Impacts on Population Within 80 km: Dose (person-rem)	Impacts on Population Within 80 km: Latent Cancer Fatalities[b]
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	5.3×10^{-3}	2.1×10^{-6}	4.6×10^{-4}	2.3×10^{-7}	3.5×10^{-1}	1.8×10^{-4}
			95th percentile	1.0×10^{-2}	4.2×10^{-6}	1.6×10^{-3}	7.8×10^{-7}	1.5	7.5×10^{-4}
Explosion in HYDOX furnace	3.4×10^{-3}	Unlikely	Mean	3.9×10^{-4}	1.6×10^{-7}	5.3×10^{-5}	2.7×10^{-8}	1.6×10^{-1}	7.8×10^{-5}
			95th percentile	8.6×10^{-4}	3.4×10^{-7}	1.6×10^{-4}	8.1×10^{-8}	7.1×10^{-1}	3.5×10^{-4}
Glovebox fire (calcining furnace)	2.7×10^{-7}	Extremely unlikely	Mean	3.1×10^{-8}	1.2×10^{-11}	4.2×10^{-9}	2.1×10^{-12}	1.2×10^{-5}	6.2×10^{-9}
			95th percentile	6.8×10^{-8}	2.7×10^{-11}	1.3×10^{-8}	6.5×10^{-12}	5.6×10^{-5}	2.8×10^{-8}
Hydrogen explosion	3.8×10^{-4}	Unlikely	Mean	4.3×10^{-5}	1.7×10^{-8}	5.9×10^{-6}	2.9×10^{-9}	1.7×10^{-2}	8.6×10^{-6}
			95th percentile	9.5×10^{-5}	3.8×10^{-8}	1.8×10^{-5}	9.0×10^{-9}	7.8×10^{-2}	3.8×10^{-5}
Melter eruption	1.4×10^{-6}	Unlikely	Mean	1.6×10^{-7}	6.4×10^{-11}	2.2×10^{-8}	1.1×10^{-11}	6.4×10^{-5}	3.2×10^{-8}
			95th percentile	3.5×10^{-7}	1.4×10^{-10}	6.7×10^{-8}	3.3×10^{-11}	2.9×10^{-4}	1.4×10^{-7}
Melter spill	3.3×10^{-7}	Unlikely	Mean	3.8×10^{-8}	1.5×10^{-11}	5.1×10^{-9}	2.6×10^{-12}	1.5×10^{-5}	7.5×10^{-9}
			95th percentile	8.3×10^{-8}	3.3×10^{-11}	1.6×10^{-8}	7.8×10^{-12}	6.8×10^{-5}	3.3×10^{-8}
Design basis earthquake	3.3×10^{-4}	Unlikely	Mean	3.5×10^{-5}	1.4×10^{-8}	4.8×10^{-6}	2.4×10^{-9}	1.4×10^{-2}	7.0×10^{-6}
			95th percentile	7.7×10^{-5}	3.1×10^{-8}	1.5×10^{-5}	7.3×10^{-9}	6.4×10^{-2}	3.1×10^{-5}
Beyond-design-basis fire	3.8×10^{-4}	Beyond extremely unlikely	Mean	3.1×10^{-4}	1.2×10^{-7}	1.4×10^{-5}	6.8×10^{-9}	2.5×10^{-2}	1.3×10^{-5}
			95th percentile	1.1×10^{-3}	4.6×10^{-7}	4.4×10^{-5}	2.2×10^{-8}	1.0×10^{-1}	5.3×10^{-5}
Beyond-design-basis earthquake	1.7×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	1.3×10^{1}	5.1×10^{-3}	5.6×10^{-1}	2.8×10^{-4}	1.0×10^{3}	5.1×10^{-1}
			95th percentile	4.7×10^{1}	1.9×10^{-2}	1.8	9.1×10^{-4}	4.3×10^{3}	2.2

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: DWPF, Defense Waste Processing Facility; HYDOX, hydride oxidation.

Note: Calculated using the source terms in the immobilization data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1999d.

Table K–19. Accident Impacts of New MOX Facility at SRS

Accident	Source Term (g)	Frequency (per year)	Meteorology	Impacts on Noninvolved Worker: Dose (rem)	Impacts on Noninvolved Worker: Probability of Cancer Fatality[a]	Impacts at Site Boundary: Dose (rem)	Impacts at Site Boundary: Probability of Cancer Fatality[a]	Impacts on Population Within 80 km: Dose (person-rem)	Impacts on Population Within 80 km: Latent Cancer Fatalities[b]
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	8.8×10^{-2}	3.5×10^{-5}	4.0×10^{-3}	2.0×10^{-6}	3.9	1.9×10^{-3}
			95th percentile	3.0×10^{-1}	1.2×10^{-4}	1.6×10^{-2}	8.0×10^{-6}	1.6×10^{1}	8.0×10^{-3}
Explosion in sintering furnace	5.5×10^{-4}	Extremely unlikely	Mean	3.3×10^{-4}	1.3×10^{-7}	1.2×10^{-5}	6.1×10^{-9}	2.9×10^{-2}	1.4×10^{-5}
			95th percentile	1.2×10^{-3}	4.6×10^{-7}	4.8×10^{-5}	2.4×10^{-8}	1.2×10^{-1}	6.1×10^{-5}
Ion exchange exotherm	2.4×10^{-5}	Unlikely	Mean	1.4×10^{-5}	5.7×10^{-9}	5.3×10^{-7}	2.7×10^{-10}	1.2×10^{-3}	6.2×10^{-7}
			95th percentile	5.1×10^{-5}	2.0×10^{-8}	2.1×10^{-6}	1.1×10^{-9}	5.3×10^{-3}	2.7×10^{-6}
Fire	4.0×10^{-6}	Unlikely	Mean	2.4×10^{-6}	9.5×10^{-10}	8.9×10^{-8}	4.4×10^{-11}	2.1×10^{-4}	1.0×10^{-7}
			95th percentile	8.4×10^{-6}	3.4×10^{-9}	3.5×10^{-7}	1.8×10^{-10}	8.8×10^{-4}	4.4×10^{-7}
Spill	5.0×10^{-6}	Extremely unlikely	Mean	3.0×10^{-6}	1.2×10^{-9}	1.1×10^{-7}	5.6×10^{-11}	2.6×10^{-4}	1.3×10^{-7}
			95th percentile	1.1×10^{-5}	4.2×10^{-9}	4.4×10^{-7}	2.2×10^{-10}	1.1×10^{-3}	5.5×10^{-7}
Design basis earthquake	7.9×10^{-5}	Unlikely	Mean	4.6×10^{-5}	1.9×10^{-8}	1.7×10^{-6}	8.7×10^{-10}	4.1×10^{-3}	2.0×10^{-6}
			95th percentile	1.7×10^{-4}	6.6×10^{-8}	6.9×10^{-6}	3.5×10^{-9}	1.7×10^{-2}	8.7×10^{-6}
Beyond-design-basis fire	6.0×10^{-2}	Beyond extremely unlikely	Mean	3.9×10^{-2}	1.6×10^{-5}	1.7×10^{-3}	8.5×10^{-7}	3.2	1.6×10^{-3}
			95th percentile	1.4×10^{-1}	5.7×10^{-5}	5.6×10^{-3}	2.8×10^{-6}	1.3×10^{1}	6.7×10^{-3}
Beyond-design-basis earthquake	9.5×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	6.2×10^{1}	2.5×10^{-2}	2.7	1.4×10^{-3}	5.0×10^{3}	2.5
			95th percentile	2.3×10^{2}	9.1×10^{-2}	8.8	4.4×10^{-3}	2.1×10^{4}	1.1×10^{1}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or at the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Note: Calculated using the source terms in the MOX data report, as modified in Appendix K.1.5.1, site meteorology, projected regional population, and the MACCS2 computer code.

Source: UC 1998d.

K.6 LEAD ASSEMBLY ACCIDENT IMPACTS

Tables K–20 through K–25 present the source terms and accident impacts of fabrication of lead assemblies for the candidate sites.

Table K–20. Accident Impacts of Lead Assembly Fabrication at ANL–W

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	Source Term (g)	Frequency (per year)	Meteorology	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Latent Cancer Fatalities[b]
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	2.5×10^{-2}	9.9×10^{-6}	1.3×10^{-3}	6.4×10^{-7}	6.8×10^{-2}	3.4×10^{-5}
			95th percentile	7.7×10^{-2}	3.1×10^{-5}	4.9×10^{-3}	2.5×10^{-6}	3.4×10^{-1}	1.7×10^{-4}
Design basis earthquake	3.9×10^{-5}	Unlikely	Mean	5.0×10^{-5}	2.0×10^{-8}	2.0×10^{-6}	1.0×10^{-9}	5.1×10^{-4}	2.6×10^{-7}
			95th percentile	1.7×10^{-4}	6.8×10^{-8}	7.7×10^{-6}	3.9×10^{-9}	2.7×10^{-3}	1.4×10^{-6}
Design basis fire	1.7×10^{-5}	Unlikely	Mean	2.2×10^{-5}	8.6×10^{-9}	8.7×10^{-7}	4.4×10^{-10}	2.2×10^{-4}	1.1×10^{-7}
			95th percentile	7.4×10^{-5}	2.9×10^{-8}	3.3×10^{-6}	1.7×10^{-9}	1.2×10^{-3}	5.9×10^{-7}
Design basis explosion	2.7×10^{-4}	Extremely unlikely	Mean	3.5×10^{-4}	1.4×10^{-7}	1.4×10^{-5}	7.1×10^{-9}	3.6×10^{-3}	1.8×10^{-6}
			95th percentile	1.2×10^{-3}	4.8×10^{-7}	5.4×10^{-5}	2.7×10^{-8}	1.9×10^{-2}	9.6×10^{-6}
Beyond-design-basis earthquake	1.1×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	2.0×10^{1}	7.9×10^{-3}	7.7×10^{-1}	3.8×10^{-4}	1.5×10^{2}	7.4×10^{-2}
			95th percentile	7.4×10^{1}	3.0×10^{-2}	2.8	1.4×10^{-3}	7.9×10^{2}	3.9×10^{-1}
Beyond-design-basis fire	2.4×10^{-2}	Beyond extremely unlikely	Mean	4.4×10^{-2}	1.8×10^{-5}	1.7×10^{-3}	8.5×10^{-7}	3.3×10^{-1}	1.6×10^{-4}
			95th percentile	1.7×10^{-1}	6.6×10^{-5}	6.2×10^{-3}	3.1×10^{-6}	1.8	8.7×10^{-4}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: ANL–W, Argonne National Laboratory–West.

Source: O'Connor et al. 1998a.

Table K–21. Accident Impacts of Lead Assembly Fabrication at Hanford (27-m Stack Height)

Accident	Source Term (g)	Frequency (per year)	Meteorology	Impacts on Noninvolved Worker: Dose (rem)	Impacts on Noninvolved Worker: Probability of Cancer Fatality[a]	Impacts at Site Boundary: Dose (rem)	Impacts at Site Boundary: Probability of Cancer Fatalities[a]	Impacts on Population Within 80 km: Dose (person-rem)	Impacts on Population Within 80 km: Latent Cancer Fatalities[b]
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	1.4×10^{-2}	5.6×10^{-6}	1.4×10^{-3}	6.8×10^{-7}	8.7×10^{-1}	4.3×10^{-4}
			95th percentile	4.0×10^{-2}	1.6×10^{-5}	4.2×10^{-3}	2.1×10^{-6}	5.5	2.7×10^{-3}
Design basis earthquake	3.9×10^{-5}	Unlikely	Mean	1.6×10^{-5}	6.5×10^{-9}	1.9×10^{-6}	9.6×10^{-10}	2.9×10^{-3}	1.4×10^{-6}
			95th percentile	4.8×10^{-5}	1.9×10^{-8}	6.3×10^{-6}	3.2×10^{-9}	1.7×10^{-2}	8.6×10^{-6}
Design basis fire	1.7×10^{-5}	Unlikely	Mean	7.1×10^{-6}	2.8×10^{-9}	8.4×10^{-7}	4.2×10^{-10}	1.2×10^{-3}	6.2×10^{-7}
			95th percentile	2.1×10^{-5}	8.4×10^{-9}	2.7×10^{-6}	1.4×10^{-9}	7.4×10^{-3}	3.7×10^{-6}
Design basis explosion	2.7×10^{-4}	Extremely unlikely	Mean	1.1×10^{-4}	4.6×10^{-8}	1.4×10^{-5}	6.8×10^{-9}	2.0×10^{-2}	1.0×10^{-5}
			95th percentile	3.4×10^{-4}	1.4×10^{-7}	4.4×10^{-5}	2.2×10^{-8}	1.2×10^{-1}	6.0×10^{-5}
Beyond-design-basis earthquake	1.1×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	1.9×10^{1}	7.5×10^{-3}	7.4×10^{-1}	3.7×10^{-4}	1.0×10^{3}	5.1×10^{-1}
			95th percentile	7.1×10^{1}	8×10^{-2}	2.7	1.3×10^{-3}	6.5×10^{3}	3.2
Beyond-design-basis fire	2.4×10^{-2}	Beyond extremely unlikely	Mean	4.1×10^{-2}	1.7×10^{-5}	1.6×10^{-3}	8.2×10^{-7}	2.2	1.1×10^{-3}
			95th percentile	1.6×10^{-1}	6.3×10^{-5}	5.9×10^{-3}	3.0×10^{-6}	1.4×10^{1}	7.2×10^{-3}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Source: O'Connor et al. 1998b.

Table K–22. Accident Impacts of Lead Assembly Fabrication at Hanford (36-m Stack Height)

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	Source Term (g)	Frequency (per year)	Meteorology	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatalities[a]	Dose (person-rem)	Latent Cancer Fatalities[b]
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	1.1×10^{-2}	4.4×10^{-6}	1.2×10^{-3}	6.0×10^{-7}	8.5×10^{-1}	4.3×10^{-4}
			95th percentile	3.3×10^{-2}	1.3×10^{-5}	3.4×10^{-3}	1.7×10^{-6}	5.4	2.7×10^{-3}
Design basis earthquake	3.9×10^{-5}	Unlikely	Mean	9.1×10^{-6}	3.6×10^{-9}	1.7×10^{-6}	8.5×10^{-10}	2.8×10^{-3}	1.4×10^{-6}
			95th percentile	3.5×10^{-5}	1.4×10^{-8}	5.2×10^{-6}	2.6×10^{-9}	1.7×10^{-2}	8.5×10^{-6}
Design basis fire	1.7×10^{-5}	Unlikely	Mean	3.9×10^{-6}	1.6×10^{-9}	7.3×10^{-7}	3.7×10^{-10}	1.2×10^{-3}	6.1×10^{-7}
			95th percentile	1.5×10^{-5}	6.0×10^{-9}	2.3×10^{-6}	1.1×10^{-9}	7.4×10^{-3}	3.7×10^{-6}
Design basis explosion	2.7×10^{-4}	Extremely unlikely	Mean	6.4×10^{-5}	2.5×10^{-8}	1.2×10^{-5}	5.9×10^{-9}	2.0×10^{-2}	9.9×10^{-6}
			95th percentile	2.4×10^{-4}	9.8×10^{-8}	3.7×10^{-5}	1.8×10^{-8}	1.2×10^{-1}	5.9×10^{-5}
Beyond-design-basis earthquake	1.1×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	1.9×10^{1}	7.5×10^{-3}	7.4×10^{-1}	3.7×10^{-4}	1.0×10^{3}	5.1×10^{-1}
			95th percentile	7.1×10^{1}	2.8×10^{-2}	2.7	1.3×10^{-3}	6.5×10^{3}	3.2
Beyond-design-basis fire	2.4×10^{-2}	Beyond extremely unlikely	Mean	4.1×10^{-2}	1.7×10^{-5}	1.6×10^{-3}	8.2×10^{-7}	2.2	1.1×10^{-3}
			95th percentile	1.6×10^{-1}	6.3×10^{-5}	5.9×10^{-3}	3.0×10^{-6}	1.4×10^{1}	7.2×10^{-3}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (single noninvolved worker at a distance of 1,000 m [3,281 ft] or the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Source: O'Connor et al. 1998b.

Table K–23. Accident Impacts of Lead Assembly Fabrication at LLNL

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatalities[a]**	**Dose (person-rem)**	**Latent Cancer Fatalities[b]**
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	7.0×10^{-2}	2.8×10^{-5}	6.7×10^{-2}	3.3×10^{-5}	1.1×10^{1}	5.7×10^{-3}
			95th percentile	5.3×10^{-1}	2.1×10^{-4}	5.3×10^{-1}	2.7×10^{-4}	6.4×10^{1}	3.2×10^{-2}
Design basis earthquake	3.9×10^{-5}	Unlikely	Mean	1.8×10^{-4}	7.2×10^{-8}	2.2×10^{-4}	1.1×10^{-7}	5.5×10^{-2}	2.8×10^{-5}
			95th percentile	1.3×10^{-3}	5.3×10^{-7}	1.7×10^{-3}	8.5×10^{-7}	2.8×10^{-1}	1.4×10^{-4}
Design basis fire	1.7×10^{-5}	Unlikely	Mean	7.8×10^{-5}	3.1×10^{-8}	9.3×10^{-5}	4.7×10^{-8}	2.4×10^{-2}	1.2×10^{-5}
			95th percentile	5.7×10^{-4}	2.3×10^{-7}	7.4×10^{-4}	3.7×10^{-7}	1.2×10^{-1}	6.0×10^{-5}
Design basis explosion	2.7×10^{-4}	Extremely unlikely	Mean	1.3×10^{-3}	5.0×10^{-7}	1.5×10^{-3}	7.6×10^{-7}	3.9×10^{-1}	1.9×10^{-4}
			95th percentile	9.3×10^{-3}	3.7×10^{-6}	1.2×10^{-2}	6.0×10^{-6}	1.9	9.7×10^{-4}
Beyond-design-basis fire	2.4×10^{-2}	Beyond extremely unlikely	Mean	1.4×10^{-1}	5.7×10^{-5}	1.3×10^{-1}	6.7×10^{-5}	3.5×10^{1}	1.8×10^{-2}
			95th percentile	1.1	4.3×10^{-4}	1.1	5.3×10^{-4}	1.7×10^{2}	8.7×10^{-2}

[a] The closest point to the site boundary is 563 m (1,847 ft), which is less than 1,000 m (3,281 ft). Therefore, doses to the onsite worker are assessed at 1,000 m [3,281 ft] only in those directions where the site boundary is greater than 1,000 m (3,281 ft) away. For other directions, doses are assessed at the site boundary.

[b] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m (3,281 ft) or the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: LLNL, Lawrence Livermore National Laboratory.

Note: A beyond-design-basis earthquake was not evaluated for Building 332 at LLNL because extensive analyses of the seismic hazard at the site and the response of the building to those hazards indicate that the scenario is beyond the range of "reasonably foreseeable." Current estimates are that the frequency of collapse is on the order of 1.0×10^{-7} per year or less.

Source: Murray 1998; O'Connor et al. 1998c.

Table K–24. Accident Impacts of Lead Assembly Fabrication at LANL

Accident	Source Term (g)	Frequency (per year)	Meteorology	Impacts on Noninvolved Worker: Dose (rem)	Impacts on Noninvolved Worker: Probability of Cancer Fatality[a]	Impacts at Site Boundary: Dose (rem)	Impacts at Site Boundary: Probability of Cancer Fatalities[a]	Impacts on Population Within 80 km: Dose (person-rem)	Impacts on Population Within 80 km: Latent Cancer Fatalities[b]
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	2.2×10^{-2}	8.7×10^{-6}	1.1×10^{-2}	5.7×10^{-6}	1.5	7.5×10^{-4}
			95th percentile	6.5×10^{-2}	2.6×10^{-5}	2.8×10^{-2}	1.4×10^{-5}	6.6	3.2×10^{-3}
Design basis earthquake	3.9×10^{-5}	Unlikely	Mean	3.4×10^{-5}	1.4×10^{-8}	1.3×10^{-5}	6.5×10^{-9}	3.1×10^{-3}	1.5×10^{-6}
			95th percentile	1.1×10^{-4}	4.3×10^{-8}	4.1×10^{-5}	2.1×10^{-8}	1.4×10^{-2}	6.8×10^{-6}
Design basis fire	1.7×10^{-5}	Unlikely	Mean	1.5×10^{-5}	6.0×10^{-9}	5.7×10^{-6}	2.8×10^{-9}	1.3×10^{-3}	6.7×10^{-7}
			95th percentile	4.7×10^{-5}	1.9×10^{-8}	1.8×10^{-5}	9.0×10^{-9}	5.9×10^{-3}	2.9×10^{-6}
Design basis explosion	2.7×10^{-4}	Extremely unlikely	Mean	2.4×10^{-4}	9.7×10^{-8}	9.2×10^{-5}	4.6×10^{-8}	2.2×10^{-2}	1.1×10^{-5}
			95th percentile	7.6×10^{-4}	3.0×10^{-7}	2.9×10^{-4}	1.5×10^{-7}	9.5×10^{-2}	4.8×10^{-5}
Beyond-design-basis earthquake	1.1×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	1.3×10^{1}	5.3×10^{-3}	4.4	2.2×10^{-3}	9.5×10^{2}	4.8×10^{-1}
			95th percentile	5.1×10^{1}	2.1×10^{-2}	1.4×10^{1}	7.0×10^{-3}	4.2×103	2.1
Beyond-design-basis fire	2.4×10^{-2}	Beyond extremely unlikely	Mean	2.9×10^{-2}	1.2×10^{-5}	9.7×10^{-3}	4.9×10^{-6}	2.1	1.1×10^{-3}
			95th percentile	1.1×10^{-1}	4.6×10^{-5}	3.1×10^{-2}	1.6×10^{-5}	9.2	4.6×10^{-3}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Key: LANL, Los Alamos National Laboratory.

Source: O'Connor et al. 1998d.

Table K–25. Accident Impacts of Lead Assembly Fabrication at SRS H-Area

				Impacts on Noninvolved Worker		Impacts at Site Boundary		Impacts on Population Within 80 km	
Accident	**Source Term (g)**	**Frequency (per year)**	**Meteorology**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatalities[a]**	**Dose (person-rem)**	**Latent Cancer Fatalities[b]**
Criticality	1.0×10^{19} fissions	Extremely unlikely	Mean	5.2×10^{-3}	2.1×10^{-6}	3.4×10^{-4}	1.7×10^{-7}	3.0×10^{-1}	1.5×10^{-4}
			95th percentile	1.0×10^{-2}	4.0×10^{-6}	9.3×10^{-4}	4.6×10^{-7}	1.3	6.5×10^{-4}
Design basis earthquake	3.9×10^{-5}	Unlikely	Mean	3.5×10^{-6}	1.4×10^{-9}	4.4×10^{-7}	2.2×10^{-10}	1.3×10^{-3}	6.3×10^{-7}
			95th percentile	7.8×10^{-6}	3.1×10^{-9}	1.3×10^{-6}	6.7×10^{-10}	5.6×10^{-3}	2.8×10^{-6}
Design basis fire	1.7×10^{-5}	Unlikely	Mean	1.5×10^{-6}	6.1×10^{-10}	1.9×10^{-7}	9.5×10^{-11}	5.4×10^{-4}	2.7×10^{-7}
			95th percentile	3.4×10^{-6}	1.3×10^{-9}	5.8×10^{-7}	2.9×10^{-10}	2.4×10^{-3}	1.2×10^{-6}
Design basis explosion	2.7×10^{-4}	Extremely unlikely	Mean	2.5×10^{-5}	9.9×10^{-9}	3.1×10^{-6}	1.5×10^{-9}	8.8×10^{-3}	4.4×10^{-6}
			95th percentile	5.5×10^{-5}	2.2×10^{-8}	9.5×10^{-6}	4.7×10^{-9}	3.9×10^{-2}	2.0×10^{-5}
Beyond-design-basis earthquake	1.1×10^{1}	Extremely unlikely to beyond extremely unlikely	Mean	7.1	2.9×10^{-3}	2.0×10^{-1}	9.8×10^{-5}	5.1×10^{2}	2.6×10^{-1}
			95th percentile	2.6×10^{1}	1.0×10^{-2}	8.8×10^{-1}	4.4×10^{-4}	2.2×10^{3}	1.1
			95th percentile						
Beyond-design-basis fire	2.4×10^{-2}	Beyond extremely unlikely	Mean	1.6×10^{-2}	6.3×10^{-6}	4.4×10^{-4}	2.2×10^{-7}	1.1	5.7×10^{-4}
			95th percentile	5.8×10^{-2}	2.3×10^{-5}	2.0×10^{-3}	9.8×10^{-7}	4.9	2.4×10^{-3}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single noninvolved worker at a distance of 1,000 m [3,281 ft] or the site boundary, whichever is smaller, or to a hypothetical individual in the offsite population located at the site boundary) if exposed to the indicated dose. The value assumes that the accident has occurred.

[b] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) if exposed to the indicated dose. The value assumes that the accident has occurred.

Source: O'Connor et al. 1998e.

K.7 COMMERCIAL REACTOR ACCIDENT ANALYSIS

K.7.1 Introduction

Postulated design basis and beyond-design-basis accidents were analyzed using the MACCS2 computer code for each of the three proposed reactor sites, Catawba Nuclear Station, McGuire Nuclear Station, and North Anna Power Station (NRC 1990, SNL 1997). Only those accidents with the potential for substantial radiological releases to the environment were evaluated. Two design basis accidents (a loss-of-coolant accident [LOCA] and a fuel-handling accident) and four beyond-design-basis accidents (a steam generator tube rupture, an early containment failure, a late containment failure, and an interfacing systems loss-of-coolant accident [ISLOCA]) meet this criteria. Each of these accidents was analyzed twice, once using the current low-enriched uranium (LEU) core, and again, assuming a partial (40 percent) MOX core. Doses (consequences) and risks to a noninvolved worker, the offsite MEI, and the general public within 80 km (50 mi) of each plant from each accident scenario were calculated. These results were then compared, by plant, for each postulated accident.

The MEI dose is calculated at the exclusion area boundary of each plant. The exclusion area boundary is that area surrounding the reactor in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided any one of these is not so close to the facility that it interferes with normal operation of the facility, and appropriate and effective arrangements are made to control traffic and protect public health and safety on the highway, railroad, or waterway in an emergency. There are generally no residences within an exclusion area. However, if there were residents, they would be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety would result.

K.7.2 Reactor Accident Identification and Quantification

Catawba and McGuire are similar plants, both with two 3,411-MWt Westinghouse pressurized water reactors (PWRs) with ice condenser containments. Because of these similarities, the release paths and mitigating mechanisms for the two plants are almost identical. The conservative assumptions of the NRC regulatory guidance produce identical radiological releases to the environment (source terms) for the two plants. However, site-specific population and meteorological inputs result in different consequences from the two plants. The North Anna site has two 2,893 MWt Westinghouse PWRs with subatmospheric containments.

Both the design basis and beyond-design-basis accidents were identified from plant documents. Design basis accidents were selected by reviewing the Updated Final Safety Analysis Report (UFSAR) for each plant (Duke Power 1996, 1997; Virginia Power 1998). Beyond-design-basis accidents were identified from the submittals (Duke Power 1991, 1992; Virginia Power 1992) in response to the NRC's Generic Letter 88–20 (NRC 1988), which required reactor licensees to perform Individual Plant Examinations (IPEs) for severe accident vulnerabilities. Source terms for each accident for LEU-only cores were identified from these documents, source terms for partial MOX cores were developed based on these LEU source terms, and analyses were performed assuming both the current LEU-only cores and partial MOX cores containing 40 percent MOX fuel and 60 percent LEU fuel. After the source term is developed, the consequences (in terms of LCFs and prompt fatalities) can be determined. To determine the risk, however, the frequency (probability) of occurrence of the accident must be determined. Then the consequences are multiplied by the frequency to determine the risk.

For this analysis, the frequencies of occurrence for the accidents with a 40 percent MOX core are assumed to be the same as those with an LEU core. The National Academy of Sciences reported (NAS 1995) that "any approach to the use of MOX fuel in U.S. power reactors must and will receive a thorough, formal safety review before it is licensed. While we are not in a position to predict what if any modifications to existing reactor types

will be required as a result of such licensing reviews, we expect that the final outcome will be certification that whatever LWR type is chosen will be able, with modifications if appropriate, to operate within prevailing reactivity and thermal margins using sufficient plutonium loadings to accomplish the disposition mission in a small number of reactors. We believe, further, that under these circumstances no important overall adverse impact of MOX use ion the accident probabilities of the LWRs involved will occur; if there are adequate reactivity and thermal margins in the fuel, as licensing review should ensure, the main remaining determinants of accident probabilities will involve factors not related to fuel composition and hence unaffected by the use of MOX rather than LEU fuel." Considering the National Academy of Sciences statements, the lack of empirical data, and the degree of uncertainty associated with accident frequencies, this analysis assumes that the accident frequencies are the same for a 40 percent MOX core as those for a 100 percent LEU core.

K.7.2.1 MOX Source Term Development

MOX source terms were developed by applying the calculated ratio for individual radioisotopes present in both the MOX and LEU cores to the source term for each of the LEU accidents. MOX source term development required several steps. The analysis assumes that the initial isotopic composition of the plutonium is that delivered to the MOX facility for fabrication into MOX fuel. The MOX facility includes a polishing step that removes impurities, including americium 241, a major contributor to the dose from plutonium 235. This analysis conservatively assumes that the polishing step reduces the americium 241 to 1 part per million (ppm), then ages the plutonium for 1 year after polishing prior to being loaded into a reactor. Table K–26 provides the assumed isotopic composition for the plutonium source material.

Table K–26. Isotopic Breakdown of Plutonium

Isotope	Prior to Polishing (wt %)	After Polishing and Aging (wt %)
Plutonium 236	<1 ppb	1 ppb
Plutonium 238	0.03	0.03
Plutonium 239	92.2	93.28
Plutonium 240	6.46	6.54
Plutonium 241	0.05	0.05
Plutonium 242	0.1	0.1
Americium 241	0.9	25 ppm

Key: ppb, parts per billion; ppm, parts per million; wt %, weight percent.

The SPD EIS assumes that MOX fuel would be fabricated using depleted uranium (0.25 weight percent uranium 235) (White 1997). The MOX assemblies are assumed to be 4.37 percent plutonium/americium and the LEU assemblies are assumed to be 4.37 percent uranium 235. To simulate a normal plant refueling cycle, the MOX portion was assumed to be 50 percent once-burned and 50 percent twice-burned assemblies. The LEU portion of the MOX was assumed to be 33.3 percent once-burned, 33.3 percent twice-burned, and 33.3 percent thrice-burned assemblies. The LEU-only cores were assumed to be equally divided between once-, twice-, and thrice-burned assemblies. All analyses assumed end-of-cycle inventories to produce the highest consequences. Fuel cycles were based on an 18-month refueling schedule with a 40-day downtime between cycles. The source terms for the LEU-only accident analyses were those identified in plant documents. Source terms for the partial MOX cores were developed using the isotopic ratios in Table K–27 provided by Oak Ridge National Laboratory (ORNL 1999). The MOX core inventory for each isotope was divided by the LEU core inventory for that isotope to provide a MOX/LEU ratio for each isotope. These ratios were then applied to LEU releases for each accident to estimate the MOX releases.

Table K–27. MOX/LEU Core Inventory Isotopic Ratios

Isotope	Ratio	Isotope	Ratio	Isotope	Ratio
Americium 241	2.06	Krypton 85m	0.86	Strontium 91	0.86
Antimony 127	1.15	Krypton 87	0.85	Strontium 92	0.89
Antimony 129	1.07	Krypton 88	0.84	Technetium 99m	0.99
Barium 139	0.97	Lanthanum 140	0.97	Tellurium 127	1.16
Barium 140	0.98	Lanthanum 141	0.97	Tellurium 127m	1.20
Cerium 141	0.98	Lanthanum 142	0.97	Tellurium 129	1.08
Cerium 143	0.95	Molybdenum 99	0.99	Tellurium 129m	1.09
Cerium 144	0.91	Neodymium 147	0.98	Tellurium 131m	1.11
Cesium 134	0.85	Neptunium 239	0.99	Tellurium 132	1.01
Cesium 136	1.09	Niobium 95	0.94	Tritium	0.95
Cesium 137	0.91	Plutonium 238	0.76	Xenon 131m	1.02
Cobalt 58	0.86	Plutonium 239	2.06	Xenon 133	1.00
Cobalt 60	0.72	Plutonium 240	2.20	Xenon 133m	1.01
Curium 242	1.43	Plutonium 241	1.79	Xenon 135	1.28
Curium 244	0.94	Praseodymium 143	0.95	Xenon 135m	1.04
Iodine 131	1.03	Rhodium 105	1.19	Xenon 138	0.96
Iodine 132	1.02	Rubidium 86	0.77	Yttrium 90	0.76
Iodine 133	1.00	Ruthenium 103	1.11	Yttrium 91	0.85
Iodine 134	0.98	Ruthenium 105	1.18	Yttrium 92	0.89
Iodine 135	1.00	Ruthenium 106	1.28	Yttrium 93	0.91
Krypton 83m	0.89	Strontium 89	0.83	Zirconium 95	0.94
Krypton 85	0.78	Strontium 90	0.75	Zirconium 97	0.98

The NRC licensing process will thoroughly review precise enrichments and fuel management schemes. The enrichments and fuel management schemes analyzed in the SPD EIS were chosen as realistic upper bounds. The accidents also assumed a maximum 40 percent MOX core. Taken together, these assumptions are sufficiently conservative to account for uncertainties associated with the MOX/LEU ratios.

K.7.2.2 Meteorological Data

Meteorological data for each specific reactor site were used. The meteorological data characteristic of the site region are described by 1 year of hourly data (8,760 measurements). This data includes wind speed, wind direction, atmospheric stability, and rainfall (DOE 1999b).

K.7.2.3 Population Data

The population distribution around each plant was determined using 1990 census data extrapolated to the year 2015. The population was then split into segments that correspond to the chosen polar coordinate grid. The polar coordinate grid for this analysis consists of 12 radial intervals aligned with the 16 compass directions. For Catawba and McGuire, the distances (in kilometers) of the 12 radial intervals are: 0.64, 0.762, 1.61, 3.22, 4.83, 6.44, 8.05, 16.09, 32.18, 48.27, 64.36, 80.45. For North Anna, these distances (in kilometers) are: 0.64, 1.350, 1.61, 3.22, 4.83, 6.44, 8.05, 16.09, 32.18, 48.27, 64.36, 80.45. The first of the 12 segments represents the location of the noninvolved worker and the second is the location of the site boundary. Projected population data for the year 2015 corresponding to the grid segments at Catawba, McGuire, and North Anna are presented in Tables K–28, K–29, and K–30, respectively.

Table K–28. Projected Catawba Population for Year 2015

Direction	Distance in Kilometers From Release Point											
	0.64	**0.762**	**1.61**	**3.22**	**4.83**	**6.44**	**8.05**	**16.09**	**32.18**	**48.27**	**64.36**	**80.45**
N	0	0	6	14	73	469	800	2,642	51,540	31,112	49,551	33,306
NNE	0	0	6	112	250	334	362	9,394	173,036	135,229	102,558	66,298
NE	0	0	7	119	239	394	595	6,442	212,814	143,650	22,571	20,108
ENE	0	0	11	81	504	1,409	1,042	5,842	72,488	52,784	32,588	10,919
E	0	0	21	5	863	1,059	570	7,959	12,144	27,800	22,844	10,995
ESE	0	0	23	47	295	388	679	7,449	8,607	18,196	12,293	9,290
SE	0	0	20	25	284	893	1,060	37,300	14,279	14,657	12,776	3,692
SSE	0	0	6	80	278	706	891	16,458	10,249	4,190	1,599	11,376
S	0	0	24	165	275	606	819	4,529	4,457	15,062	1,579	1,874
SSW	0	0	17	137	245	238	346	2,268	3,563	2,093	12,970	4,245
SW	0	0	20	114	162	208	267	5,538	9,559	2,040	11,272	12,302
WSW	0	0	21	84	159	205	257	2,493	4,756	8,947	31,712	80,518
W	0	0	23	113	202	272	345	4,979	6,978	17,182	26,070	35,091
WNW	0	0	23	103	199	283	363	3,011	17,814	32,751	29,031	8,706
NW	0	0	23	96	165	274	363	3,099	65,856	28,474	33,819	45,793
NNW	0	0	21	85	125	1,153	1,296	3,404	48,431	24,219	32,537	52,530

Table K–29. Projected McGuire Population for Year 2015

Direction	Distance in Kilometers From Release Point											
	0.64	**0.762**	**1.61**	**3.22**	**4.83**	**6.44**	**8.05**	**16.09**	**32.18**	**48.27**	**64.36**	**80.45**
N	0	0	44	0	269	110	203	3,153	14,870	28,254	12,987	15,726
NNE	0	0	28	0	124	569	1,728	9,493	21,903	12,317	24,826	43,937
NE	0	0	30	0	5	832	1,016	6,944	30,939	44,064	55,186	44,691
ENE	0	0	184	144	405	684	591	4,289	51,928	37,373	13,039	28,160
E	0	0	217	180	448	381	493	7,575	26,495	21,992	16,957	14,635
ESE	0	0	65	69	271	381	507	7,423	119,345	79,039	36,221	26,552
SE	0	0	15	59	130	244	273	8,387	219,183	204,614	46,100	24,527
SSE	0	0	15	59	99	138	100	9,530	90,900	95,688	79,859	15,954
S	0	0	14	83	165	182	165	6,429	35,178	21,241	41,638	9,071
SSW	0	0	18	101	169	240	221	3,261	61,514	29,814	10,774	9,327
SW	0	0	26	101	169	236	305	5,338	20,195	31,064	47,641	43,067
WSW	0	0	19	101	169	236	296	2,741	20,873	17,334	15,815	15,077
W	6	0	14	112	184	252	312	2,048	24,932	11,715	12,705	43,357
WNW	0	0	3	101	444	811	338	2,187	14,985	57,262	74,708	60,953
NW	0	0	0	224	200	1,005	793	4,260	8,528	22,380	26,093	12,511
NNW	0	0	0	0	4	0	36	1,989	8,570	40,993	13,101	10,686

Table K–30. Projected North Anna Population for Year 2015

Direction	Distance in Kilometers From Release Point											
	0.64	**1.35**	**1.61**	**3.22**	**4.83**	**6.44**	**8.05**	**16.09**	**32.18**	**48.27**	**64.36**	**80.45**
N	0	0	0	39	98	122	153	576	7,816	5,149	17,803	42,233
NNE	0	0	2	37	58	160	206	1,236	7,634	10,765	25,976	172,658
NE	0	0	2	30	43	94	100	1,122	38,833	90,820	34,429	77,097
ENE	0	0	0	15	103	40	64	1,373	5,822	6,693	11,426	17,324
E	0	0	0	17	112	42	34	1,183	6,128	5,175	1,839	4,296
ESE	0	0	2	7	17	97	135	950	5,595	5,454	5,161	7,909
SE	0	0	1	18	77	9	12	575	2,989	19,343	59,057	76,396
SSE	0	0	3	50	29	27	40	919	5,051	15,259	443,326	392,420
S	0	0	0	42	20	30	40	669	4,413	11,763	20,254	34,375
SSW	0	0	0	10	12	54	65	554	3,098	5,803	5,616	6,222
SW	0	0	0	4	14	54	86	1,186	2,678	2,845	5,482	4,576
WSW	0	0	0	19	42	31	63	1,381	4,402	6,729	8,905	8,094
W	0	0	0	31	24	24	29	466	2,883	4,529	109,205	21,748
WNW	0	0	0	30	79	52	29	606	2,725	8,371	17,931	9,934
NW	0	0	1	35	52	92	81	662	3,327	11,604	11,816	3,090
NNW	0	0	0	28	64	13	25	771	4,725	9,040	25,534	10,041

K.7.2.4 Design Basis Events

Design basis events are defined by the American Nuclear Society as Condition IV occurrences or limiting faults. Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of substantial radioactive material. These are the most serious events which must be designed against and represent limiting design cases.

The accident analyses presented in the UFSARs are conservative design basis analyses and therefore the dose consequences are bounding (i.e., a realistically based analysis would result in lower doses). The results, however, provide a comparison of the potential consequences resulting from design basis accidents. The consequences also provide insight into which design basis accidents should be analyzed in an environmental impact statement, such as the SPD EIS. After reviewing the UFSAR accident analyses, the design basis accidents chosen for evaluation in the SPD EIS are a large-break LOCA and a fuel-handling accident.

LOCA. A design basis large-break LOCA was chosen for evaluation because it is the limiting reactor design basis accident at each of the three plants. The analysis was performed in accordance with the methodology and assumptions in Regulatory Guide 1.4 (NRC 1974). The large-break LOCA is defined as a break equivalent in size to a double-ended rupture of the largest pipe of the reactor coolant system. Following a postulated double-ended rupture of a reactor coolant pipe, the emergency core cooling system keeps cladding temperatures well below melting, ensuring that the core remains intact and in a coolable geometry. As a result of the increase in cladding temperature and rapid depressurization of the core, however, some cladding failure may occur in the hottest regions of the core. Thus, a fraction of the fission products accumulated in the pellet-cladding gap may be released to the reactor coolant system and thereby to the containment. Although no core melting would occur for the design basis LOCA, a gross release of fission products is evaluated. The only postulated mechanism for such a release would require a number of simultaneous and extended failures to occur in the engineered safety feature systems, producing severe physical degradation of core geometry and partial melting of the fuel.

Development of the LOCA source term is based on the conservative assumptions specified in Regulatory Guide 1.4. Consistent with this Regulatory Guide, 100 percent of the noble gas inventory and 25 percent of the iodine inventory in the core are assumed to be immediately available for leakage from the primary containment.

However, all of this radioactivity is not released directly to the environment because there are a number of mitigating mechanisms which can delay or retain radioisotopes. The principal mechanism, the primary containment, substantially restricts the release rate of the radioisotopes. Following a postulated LOCA, another potential source of fission product release to the environment is the leakage of radioactive water from engineered safety feature equipment located outside containment. The fission products could then be released from the water into the atmosphere, resulting in offsite radiological consequences that contribute to the total dose from the LOCA.

The LOCA radiological consequence analysis for the LEU cores was performed assuming a ground-level release based on offeror-supplied plant-specific radioisotope release data. All possible leak paths (containment, bypass, and the emergency core cooling system) were included. Were a LOCA to occur, a substantial percentage of the releases would be expected to be elevated, which would be expected to reduce the consequences from those calculated in this analysis. To analyze the accident for a partial MOX core, the LEU isotopic activity was multiplied by the MOX/LEU ratios (from Table K–27) to provide a MOX core activity for each isotope. The LEU and MOX LOCA releases for Catawba and McGuire are provided in Table K–31 and for North Anna in Table K–32.

Table K–31. Catawba and McGuire LOCA Source Term

Isotope	LEU LOCA Release (Ci)	MOX/LEU Ratio	40% MOX Core Release (Ci)
Iodine 131	2.42×10^4	1.03	2.49×10^4
Iodine 132	7.76×10^2	1.02	7.92×10^2
Iodine 133	3.22×10^3	1.00	3.22×10^3
Iodine 134	6.55×10^2	0.98	6.42×10^2
Iodine 135	2.51×10^3	1.00	2.51×10^3
Krypton 83m	3.62×10^3	0.89	3.22×10^3
Krypton 85	1.96×10^4	0.78	1.53×10^4
Krypton 85m	1.96×10^4	0.86	1.68×10^4
Krypton 87	1.04×10^4	0.85	8.82×10^3
Krypton 88	3.23×10^4	0.84	2.72×10^4
Xenon 131m	2.79×10^4	1.02	2.84×10^4
Xenon 133	2.33×10^6	1.00	2.33×10^6
Xenon 133m	3.45×10^4	1.01	3.49×10^4
Xenon 135	2.90×10^5	1.28	3.71×10^5
Xenon 135m	1.40×10^3	1.04	1.46×10^3
Xenon 138	7.21×10^3	0.96	6.92×10^3

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Fuel-Handling Accident. The fuel-handling accident analysis was performed in a conservative manner, in accordance with Regulatory Guide 1.25 methodology (NRC 1972). In the fuel-handling accident scenario, a spent fuel assembly is dropped. The drop results in a breach of the fuel rod cladding, and a portion of the volatile fission gases from the damaged fuel rods is released. A fuel-handling accident would realistically result in only a fraction of the fuel rods being damaged. However, consistent with NRC methodology, all the fuel rods in the assembly are assumed to be damaged.

Table K–32. North Anna LOCA Source Term

Isotope	LEU LOCA Release (Ci)	MOX/LEU Ratio	40% MOX Core Release (Ci)
Iodine 131	3.68×10^{2}	1.03	3.79×10^{2}
Iodine 132	3.45×10^{2}	1.02	3.52×10^{2}
Iodine 133	5.87×10^{2}	1.00	5.87×10^{2}
Iodine 134	5.10×10^{2}	0.98	5.00×10^{2}
Iodine 135	5.01×10^{2}	1.00	5.01×10^{2}
Krypton 83m	4.26×10^{2}	0.89	3.79×10^{2}
Krypton 85	5.06×10^{1}	0.78	3.95×10^{1}
Krypton 85m	1.48×10^{3}	0.86	1.27×10^{3}
Krypton 87	2.22×10^{3}	0.85	1.89×10^{3}
Krypton 88	3.50×10^{3}	0.84	2.94×10^{3}
Xenon 131m	3.20×10^{1}	1.02	3.26×10^{1}
Xenon 133	6.91×10^{3}	1.00	6.91×10^{3}
Xenon 133m	1.70×10^{2}	1.01	1.72×10^{2}
Xenon 135	6.37×10^{3}	1.28	8.15×10^{3}
Xenon 135m	6.72×10^{2}	1.04	6.99×10^{2}
Xenon 138	1.90×10^{3}	0.96	1.82×10^{3}

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

The accident is assumed to occur at the earliest time fuel-handling operations may begin after shutdown as identified in each plant's Technical Specifications.[8] The assumed accident time is 72 hr after shutdown at Catawba and McGuire. North Anna Technical Specifications require a minimum of 150 hr between shutdown and the initiation of fuel movement, but assumed an accident time of 100 hr.

As assumed in Regulatory Guide 1.25, the damaged assembly is the highest powered assembly being removed from the reactor. The values for individual fission product inventories in the damaged assembly are calculated assuming full power operation at the end of core life immediately preceding shutdown. All of the gap activity in the damaged rods is assumed to be released to the spent fuel pool. Noble gases released to the spent fuel pool are immediately released at ground level to the environment, but the water in the spent fuel pool greatly reduces the iodine available for release to the environment. It is assumed that all of the iodine escaping from the spent fuel pool is released to the environment at ground level over a 2-hr time period through the fuel-handling building ventilation system. The Catawba and McGuire UFSARs assume iodine filter efficiencies of 95 percent for both the inorganic and organic species. The North Anna UFSAR assumes a filter efficiency of 90 percent for the inorganic iodine and 70 percent for the organic iodine. The LEU and MOX source terms for Catawba and McGuire are provided in Table K–33 and the source terms for North Anna are provided in Table K–34.

The frequencies for the design basis LOCAs, obtained from the IPEs, are Catawba, 7.50×10^{-6}; McGuire, 1.50×10^{-5}; and North Anna, 2.10×10^{-5}. The frequencies of the fuel-handling accidents were estimated in lieu of plant-specific data. For conservatism, a frequency of 1×10^{-4} was chosen for the analysis.

[8] Technical Specifications are plant-specific operating conditions that control safety-related parameters of plant operation. Technical Specifications are part of the operating license and require an operating license amendment to change.

Table K–33. Catawba and McGuire Fuel-Handling Accident Source Term

Nuclide	LEU Release (Ci)	MOX/LEU Ratio	40% MOX Core Release
Iodine 131	3.83×10^{1}	1.03	3.94×10^{1}
Iodine 132	5.55×10^{1}	1.02	5.66×10^{1}
Iodine 133	8.00×10^{1}	1.00	8.00×10^{1}
Iodine 134	8.80×10^{1}	0.98	8.62×10^{1}
Iodine 135	7.55×10^{1}	1.00	7.55×10^{1}
Krypton 83m	9.47×10^{3}	0.89	8.43×10^{3}
Krypton 85	1.11×10^{3}	0.78	8.66×10^{2}
Krypton 85m	2.16×10^{4}	0.86	1.86×10^{4}
Krypton 87	4.04×10^{4}	0.85	3.43×10^{4}
Krypton 88	5.58×10^{4}	0.84	4.69×10^{4}
Xenon 133	1.60×10^{5}	1.00	1.60×10^{5}
Xenon 133m	4.81×10^{3}	1.01	4.86×10^{3}
Xenon 135	1.65×10^{5}	1.28	2.11×10^{5}
Xenon 135m	2.96×10^{4}	1.04	3.08×10^{4}
Xenon 138	1.34×10^{5}	0.96	1.29×10^{5}

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

Table K–34. North Anna Fuel-Handling Accident Source Term

Nuclide	LEU Release (Ci)	MOX/LEU Ratio	40% MOX Core Release
Iodine 131	9.05×10^{1}	1.03	9.32×10^{1}
Iodine 132	1.37×10^{2}	1.02	1.40×10^{2}
Iodine 133	2.01×10^{2}	1.00	2.01×10^{2}
Iodine 134	2.36×10^{2}	0.98	2.31×10^{2}
Iodine 135	1.82×10^{2}	1.00	1.82×10^{2}
Krypton 85	2.60×10^{3}	0.78	2.03×10^{3}
Krypton 85m	2.65×10^{4}	0.86	2.28×10^{4}
Krypton 87	5.10×10^{4}	0.85	4.34×10^{4}
Krypton 88	7.25×10^{4}	0.84	6.09×10^{4}
Xenon 131m	4.56×10^{2}	1.02	4.65×10^{2}
Xenon 133	1.36×10^{5}	1.00	1.36×10^{5}
Xenon 133m	3.46×10^{3}	1.01	3.49×10^{3}
Xenon 135	3.70×10^{4}	1.28	4.74×10^{4}
Xenon 135m	3.74×10^{4}	1.04	3.89×10^{4}
Xenon 138	1.22×10^{5}	0.96	1.17×10^{5}

Key: LEU, low-enriched uranium; LOCA, loss-of-coolant accident.

K.7.2.5 Beyond-Design-Basis Events

Beyond-design-basis accidents (severe reactor accidents) are less likely to occur than reactor design basis accidents. In the reactor design basis accidents, the mitigating systems are assumed to be available. In the severe reactor accidents, even though the initiating event could be a design basis event (e.g., large-break LOCA), additional failures of mitigating systems would cause some degree of physical deterioration of the fuel in the

reactor core and a possible breach of the containment structure leading to the direct release of radioactive materials to the environment.

The beyond-design-basis accident evaluation in the SPD EIS included a review of each plant's IPE. In 1988, the NRC required all licensees of operating plants to perform IPEs for severe accident vulnerabilities (Generic Letter 88-20) (NRC 1988), and indicated that a Probabalistic Risk Assessment (PRA) would be an acceptable approach to performing the IPE. A PRA evaluates, in full detail (quantitatively), the consequences of all potential events caused by the operating disturbances (known as internal initiating events) within each plant. The state-of-the-art PRA uses realistic criteria and assumptions in evaluating the accident progression and the systems required to mitigate each accident.

A plant-specific PRA for severe accident vulnerabilities starts with identification of initiating events (i.e., challenges to normal plant operation or accidents) that require successful mitigation to prevent core damage. These events are grouped into initiating event classes that have similar characteristics and require the same overall plant response.

Event trees are developed for each initiating event class. These event trees depict the possible sequence of events that could occur during the plant's response to each initiating event class. The trees delineate the possible combinations (sequences) of functional and/or system successes and failures that lead to either successful mitigation of the initiating event or core damage. Functional and/or system success criteria are developed based on the plant response to the class of accident sequences. Failure modes of systems that are functionally important to preventing core damage are modeled. This modeling process is usually done with fault trees that define the combinations of equipment failures, equipment outages, and human errors that could cause the failure of systems to perform the desired functions.

Quantification of the event trees leads to hundreds, or even thousands, of different end states representing various accident sequences that are either mitigated or lead to core damage. Each accident sequence and its associated end state has a unique "signature" because of the particular combination of system successes and failures. These end states are grouped together into plant damage states, each of which collects sequences for which the progression of core damage, the release of fission products from the fuel, the status of containment and its systems, and the potential for mitigating source terms are similar. The sum of all core damage accident sequences will then represent an estimate of plant core damage frequency. The analysis of core damage frequency calculations is called a Level 1 PRA, or front-end analysis.

Next, an analysis of accident progression, containment loading[9] resulting from the accident, and the structural response to the accident loading is performed. The primary objective of this analysis, which is called a Level 2 PRA, is to characterize the potential for, and magnitude of, a release of radioactive material from the reactor fuel to the environment, given the occurrence of an accident that damages the core. The analysis includes an assessment of containment performance in response to a series of severe accidents. Analysis of the progression of an accident (an accident sequence within a plant damage state) generates a time history of loads imposed on the containment pressure boundary. These loads would then be compared against the containment's structural performance limits. If the loads exceed the performance limits, the containment would be expected to fail; conversely, if the containment performance limits exceed the calculated loads, the containment would be expected to survive. Four modes of containment failure are defined: containment isolation failure, containment bypass, early containment failure, and late containment failure.

[9] Challenges to containment integrity such as elevated temperature or pressure are referred to as containment loading.

The magnitude of the radioactive release to the atmosphere in an accident is dependent on the timing of the reactor vessel failure and the containment failure. To determine the magnitude of the release, a containment event tree representing the time sequence of major phenomenological events that could occur during the formation and relocation of core debris (after core melt), availability of the containment heat removal system, and the expected mode of containment failures (i.e., bypass, early, and late), is developed. A reduced set of plant damage states is defined by culling the lower frequency plant damage states into higher frequency ones that have relatively similar severity and consequence potential. This condensed set is known as the key plant damage states. These key plant damage states would then become the initiating events for the containment event tree. The outcome of each sequence in this event tree represents a specific release category. Release categories that can be represented by similar source terms are grouped. Source terms associated with various release categories describe the fractional releases for representative radionuclide groups, as well as the timing, duration, and energy of release.

Beyond-design-basis accidents evaluated in the SPD EIS included only those scenarios that lead to containment bypass or failure because the public and environmental consequences would be significantly less for accident scenarios that do not lead to containment bypass or failure. The accidents evaluated consisted of a steam generator tube rupture, an early containment failure, a late containment failure, and an ISLOCA.

Steam Generator Tube Rupture. A beyond-design-basis steam generator tube rupture induced by high temperatures represents a containment bypass event. Analyses have indicated a potential for very high gas temperatures in the reactor coolant system during accidents involving core damage when the primary system is at high pressure. The high temperature could fail the steam generator tubes. As a result of the tube rupture, the secondary side may be exposed to full Reactor Coolant System pressures. These pressures are likely to cause relief valves to lift on the secondary side as they are designed to do. If these valves fail to close after venting, an open pathway from the reactor vessel to the environment can result.

Early Containment Failure. This accident is defined as the failure of containment prior to or very soon (within a few hours) after breach of the reactor vessel. A variety of mechanisms such as direct contact of core debris with the containment, rapid pressure and temperature loads, hydrogen combustion, and fuel-coolant interactions can cause structural failure of the containment. Early containment failure can be important because it tends to result in shorter warning times for initiating public protective measures, and because radionuclide releases would generally be more severe than if the containment fails late.

Late Containment Failure. A late containment failure involves structural failure of the containment several hours after breach of the reactor vessel. A variety of mechanisms such as gradual pressure and temperature increase, hydrogen combustion, and basemat melt-through by core debris can cause late containment failure.

ISLOCA. An ISLOCA refers to a class of accidents in which the reactor coolant system pressure boundary interfacing with a supporting system of lower design pressure is breached. If this occurs, the lower pressure system will be overpressurized and could rupture outside the containment. This failure would establish a flow path directly to the environment or, sometimes, to another building of small-pressure capacity.

For each of the proposed reactors, an assessment was made of the pre-accident inventories of each radioactive species in the reactor fuel, using information on the thermal power and refueling cycles. For the source term and offsite consequence analysis, the radioactive species were collected into groups that exhibit similar chemical behavior. The following groups represent the radionuclides considered to be most important to offsite consequences: noble gases, iodine, cesium, tellurium, strontium, ruthenium, lanthanum, cerium, and barium.

The LEU end-of-cycle isotopic activities (inventories) were multiplied by the MOX/LEU ratio to provide a MOX end-of-cycle activity for each isotope. The LEU and MOX core activities for Catawba and McGuire are provided in Table K–35. The activities for North Anna are provided in Table K–36.

Table K–35. Catawba and McGuire End-of-Cycle Core Activities

Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)	Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)
Americium 241	3.13×10^{3}	2.06	6.45×10^{3}	Niobium 95	1.41×10^{8}	0.94	1.33×10^{8}
Antimony 127	7.53×10^{6}	1.15	8.66×10^{6}	Plutonium 238	9.90×10^{4}	0.76	7.53×10^{4}
Antimony 129	2.67×10^{7}	1.07	2.85×10^{7}	Plutonium 239	2.23×10^{4}	2.06	4.60×10^{4}
Barium 139	1.70×10^{8}	0.97	1.65×10^{8}	Plutonium 240	2.82×10^{4}	2.20	6.20×10^{4}
Barium 140	1.68×10^{8}	0.98	1.65×10^{8}	Plutonium 241	4.74×10^{6}	1.79	8.49×10^{6}
Cerium 141	1.53×10^{8}	0.98	1.50×10^{8}	Praseodymium 143	1.46×10^{8}	0.95	1.39×10^{8}
Cerium 143	1.48×10^{8}	0.95	1.41×10^{8}	Rhodium 105	5.53×10^{7}	1.19	6.58×10^{7}
Cerium 144	9.20×10^{7}	0.91	8.37×10^{7}	Rubidium 86	5.10×10^{4}	0.77	3.93×10^{4}
Cesium 134	1.17×10^{7}	0.85	9.93×10^{6}	Ruthenium 103	1.23×10^{8}	1.11	1.36×10^{8}
Cesium 136	3.56×10^{6}	1.09	3.88×10^{6}	Ruthenium 105	7.98×10^{7}	1.18	9.42×10^{7}
Cesium 137	6.53×10^{6}	0.91	5.94×10^{6}	Ruthenium 106	2.79×10^{7}	1.28	3.57×10^{7}
Cobolt 58	8.71×10^{5}	0.86	7.49×10^{5}	Strontium 89	9.70×10^{7}	0.83	8.05×10^{7}
Cobolt 60	6.66×10^{5}	0.72	4.80×10^{5}	Strontium 90	5.24×10^{6}	0.75	3.93×10^{6}
Curium 242	1.20×10^{6}	1.43	1.71×10^{6}	Strontium 91	1.25×10^{8}	0.86	1.07×10^{8}
Curium 244	7.02×10^{4}	0.94	6.60×10^{4}	Strontium 92	1.30×10^{8}	0.89	1.16×10^{8}
Iodine 131	8.66×10^{7}	1.03	8.92×10^{7}	Technetium 99m	1.42×10^{8}	0.99	1.41×10^{8}
Iodine 132	1.28×10^{8}	1.02	1.30×10^{8}	Tellurium 127	7.28×10^{6}	1.16	8.44×10^{6}
Iodine 133	1.83×10^{8}	1.00	1.83×10^{8}	Tellurium 127m	9.63×10^{5}	1.20	1.16×10^{6}
Iodine 134	2.01×10^{8}	0.98	1.97×10^{8}	Tellurium 129	2.50×10^{7}	1.08	2.70×10^{7}
Iodine 135	1.73×10^{8}	1.00	1.73×10^{8}	Tellurium 129m	6.60×10^{6}	1.09	7.20×10^{6}
Krypton 85	6.69×10^{5}	0.78	5.22×10^{5}	Tellurium 131m	1.26×10^{7}	1.11	1.40×10^{7}
Krypton 85m	3.13×10^{7}	0.86	2.69×10^{7}	Tellurium 132	1.26×10^{8}	1.01	1.27×10^{8}
Krypton 87	5.72×10^{7}	0.85	4.87×10^{7}	Xenon 133	1.83×10^{8}	1.00	1.83×10^{8}
Krypton 88	7.74×10^{7}	0.84	6.50×10^{7}	Xenon 135	3.44×10^{7}	1.28	4.40×10^{7}
Lanthanum 140	1.72×10^{8}	0.97	1.67×10^{8}	Yttrium 90	5.62×10^{6}	0.76	4.27×10^{6}
Lanthanum 141	1.57×10^{8}	0.97	1.53×10^{8}	Yttrium 91	1.18×10^{8}	0.85	1.00×10^{8}
Lanthanum 142	1.52×10^{8}	0.97	1.47×10^{8}	Yttrium 92	1.30×10^{8}	0.89	1.16×10^{8}
Molybdenum 99	1.65×10^{8}	0.99	1.63×10^{8}	Yttrium 93	1.47×10^{8}	0.91	1.34×10^{8}
Neodymium 147	6.52×10^{7}	0.98	6.39×10^{7}	Zirconium 95	1.49×10^{8}	0.94	1.40×10^{8}
Neptunium 239	1.75×10^{9}	0.99	1.73×10^{9}	Zirconium 97	1.56×10^{8}	0.98	1.53×10^{8}

Key: LEU, low-enriched uranium.

Table K–36. North Anna End-of-Cycle Core Activities

Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)	Isotope	LEU Core Activity (Ci)	MOX/ LEU Ratio	40% MOX Core Activity (Ci)
Americium 241	1.03×10^{4}	2.06	2.13×10^{4}	Plutonium 238	1.99×10^{5}	0.76	1.51×10^{5}
Antimony 127	6.36×10^{6}	1.15	7.31×10^{6}	Plutonium 239	2.70×10^{4}	2.06	5.57×10^{4}
Antimony 129	2.41×10^{7}	1.07	2.58×10^{7}	Plutonium 240	3.43×10^{4}	2.20	7.54×10^{4}
Barium 139	1.39×10^{8}	0.97	1.35×10^{8}	Plutonium 241	9.82×10^{6}	1.79	1.76×10^{7}
Barium 140	1.37×10^{8}	0.98	1.34×10^{8}	Praseodymium 143	1.17×10^{8}	0.95	1.11×10^{8}
Cerium 141	1.25×10^{8}	0.98	1.22×10^{8}	Rhodium 105	7.22×10^{7}	1.19	8.59×10^{7}
Cerium 143	1.18×10^{8}	0.95	1.12×10^{8}	Rubidium 86	1.45×10^{4}	0.77	1.12×10^{4}
Cerium 144	9.70×10^{7}	0.91	8.82×10^{7}	Rubidium 103	1.16×10^{8}	1.11	1.28×10^{8}
Cesium 134	1.28×10^{7}	0.85	1.09×10^{7}	Rubidium 105	7.84×10^{7}	1.18	9.25×10^{7}
Cesium 136	3.42×10^{6}	1.09	3.72×10^{6}	Rubidium 106	3.83×10^{7}	1.28	4.90×10^{7}
Cesium 137	8.41×10^{6}	0.91	7.66×10^{6}	Strontium 89	7.48×10^{7}	0.83	6.21×10^{7}
Curium 242	2.72×10^{6}	1.43	3.88×10^{6}	Strontium 90	6.22×10^{6}	0.75	4.66×10^{6}
Curium 244	2.75×10^{5}	0.94	2.58×10^{5}	Strontium 91	9.36×10^{7}	0.86	8.05×10^{7}
Iodine 131	7.33×10^{7}	1.03	7.55×10^{7}	Strontium 92	1.04×10^{8}	0.89	9.23×10^{7}
Iodine 132	1.07×10^{8}	1.02	1.09×10^{8}	Technetium 99m	1.26×10^{8}	0.99	1.25×10^{8}
Iodine 133	1.52×10^{8}	1.00	1.52×10^{8}	Tellurium 127	6.21×10^{6}	1.16	7.21×10^{6}
Iodine 134	1.75×10^{8}	0.98	1.71×10^{8}	Tellurium 127m	9.87×10^{5}	1.20	1.18×10^{6}
Iodine 135	1.49×10^{8}	1.00	1.49×10^{8}	Tellurium 129	2.29×10^{7}	1.08	2.47×10^{7}
Krypton 85	3.51×10^{6}	0.78	2.74×10^{6}	Tellurium 129m	4.20×10^{6}	1.09	4.58×10^{6}
Krypton 85m	8.69×10^{5}	0.86	7.48×10^{5}	Tellurium 132	1.07×10^{8}	1.01	1.08×10^{8}
Krypton 87	3.86×10^{7}	0.85	3.28×10^{7}	Xenon 133	1.59×10^{8}	1.00	1.59×10^{8}
Krypton 88	5.46×10^{7}	0.84	4.59×10^{7}	Xenon 133m	4.69×10^{6}	1.01	4.73×10^{6}
Lanthanum 140	1.42×10^{8}	0.97	1.37×10^{8}	Xenon 135	4.47×10^{7}	1.28	5.72×10^{7}
Lanthanum 141	1.28×10^{8}	0.97	1.24×10^{8}	Yttrium 90	6.21×10^{6}	0.76	4.72×10^{6}
Lanthanum 142	1.24×10^{8}	0.97	1.21×10^{8}	Yttrium 91	9.93×10^{7}	0.85	8.44×10^{7}
Molybdenum 99	1.43×10^{8}	0.99	1.42×10^{8}	Yttrium 92	1.01×10^{8}	0.89	8.97×10^{7}
Neodymium 147	5.12×10^{7}	0.98	5.02×10^{7}	Yttrium 93	1.16×10^{8}	0.91	1.05×10^{8}
Neptunium 239	1.51×10^{9}	0.99	1.50×109	Zirconium 95	1.27×10^{8}	0.94	1.20×10^{8}
Niobium 95	1.31×10^{8}	0.94	1.23×10^{8}	Zirconium 97	1.28×10^{8}	0.98	1.26×10^{8}

Key: LEU, low-enriched uranium.

The source term for each accident, taken from each plant's PRA, is described by the release height, timing, duration, and heat content of the plume, the fraction of each isotope group released, and the warning time (time when offsite officials are warned that an emergency response should be initiated). The PRAs included several release categories for each bypass and failure scenario. These release categories were screened for each accident scenario to determine which release category resulted in the highest risk. The risk was determined by multiplying the consequences by the frequency for each release category. The release category with the highest risk for each scenario was used in the SPD EIS analysis. The highest risk release category source terms for Catawba, McGuire, and North Anna are presented in Table K–37. Also included in each release category characterization is the frequency of occurrence.

The overall risk from beyond-design-basis accidents can be described by the sum of risks from all beyond-design-basis accidents. The group of accidents derived from the screening process results in the highest risks from the containment bypass and failure scenarios. The screened-out accidents in these categories not only

Table K–37. Beyond-Design-Basis Accident Source Terms

Accident	Parameters	Release Category	Frequency	Release Fractions								
				Xe/Kr	I	Cs/Rb	Te/Sb	Sr	Ru/Mo	La	Ce	Ba
				CATAWBA								
SG tube rupture[a]	Time: 20 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 7.5 hr	1.04	6.31×10^{-10}	1.0	7.7×10^{-1}	7.9×10^{-1}	7.3×10^{-1}	5.0×10^{-3}	9.4×10^{-2}	1.3×10^{-4}	NA	4.0×10^{-2}
Early containment failure	Time: 6.0 hr Duration: 0.5 hr Energy: 2.0×10^{7} cal/sec (8.37×10^{7} W) Elevation: 10.0 m Warning time: 5.5 hr	5.01	3.42×10^{-8}	1.0	5.5×10^{-2}	4.8×10^{-2}	3.0×10^{-2}	2.5×10^{-4}	2.2×10^{-3}	1.2×10^{-4}	NA	1.7×10^{-3}
Late containment failure	Time: 18.5 hr Duration: 0.5 hr Energy: 1.0×10^{7} cal/sec (4.2×10^{7} W) Elevation: 10.0 m Warning time: 18.0 hr	6.01	1.21×10^{-5}	1.0	3.6×10^{-3}	3.9×10^{-3}	1.8×10^{-3}	5.2×10^{-5}	3.8×10^{-4}	2.6×10^{-5}	NA	1.6×10^{-4}
Interfacing systems LOCA	Time: 6.0 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 5.5 hr	2.04	6.9×10^{-8}	1.0	8.2×10^{-1}	8.2×10^{-1}	7.9×10^{-1}	5.8×10^{-2}	2.1×10^{-1}	3.1×10^{-2}	NA	1.4×10^{-1}

Table K–37. Beyond-Design-Basis Accident Source Terms (Continued)

Accident	Parameters	Release Category	Frequency	Release Fractions								
				Xe/Kr	I	Cs/Rb	Te/Sb	Sr	Ru/Mo	La	Ce	Ba
McGUIRE												
SG tube rupture	Time: 20.0 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 7.5 hr	1.04	5.81×10^{-9}	1.0	7.7×10^{-1}	7.9×10^{-1}	7.3×10^{-1}	5.0×10^{-3}	9.4×10^{-2}	1.3×10^{-4}	NA	4.0×10^{-2}
Early containment failure	Time: 6.0 hr Duration: 0.5 hr Energy: 2.0×10^{7} cal/sec (8.37×10^{7} W) Elevation: 10.0 m Warning time: 5.5 hr	5.01	9.89×10^{-8}	1.0	4.4×10^{-2}	3.5×10^{-2}	2.1×10^{-2}	1.4×10^{-4}	4.3×10^{-3}	2.0×10^{-5}	NA	1.4×10^{-3}
Late containment failure	Time: 32.0 hr Duration: 0.5 hr Energy: 1.0×10^{7} cal/sec (4.2×10^{7} W) Elevation: 10.0 m Warning time: 31.5 hr	6.01	7.21×10^{-6}	1.0	3.2×10^{-3}	2.4×10^{-3}	3.3×10^{-3}	1.0×10^{-8}	5.8×10^{-8}	1.0×10^{-9}	NA	1.8×10^{-7}
Interfacing systems LOCA	Time: 3.0 hr Duration: 1.0 hr Energy: 1.0×10^{4} cal/sec (4.2×10^{4} W) Elevation: 10.0 m Warning time: 2.0 hr	2.04	6.35×10^{-7}	1.0	7.5×10^{-1}	7.5×10^{-1}	6.6×10^{-1}	4.2×10^{-2}	1.5×10^{-1}	2.0×10^{-2}	NA	9.8×10^{-2}

Table K–37. Beyond-Design-Basis Accident Source Terms (Continued)

Accident	Parameters	Release Category	Frequency	Release Fractions								
				Xe/Kr	I	Cs/Rb	Te/Sb	Sr	Ru/Mo	La	Ce	Ba
NORTH ANNA												
SG tube rupture	Time: 20.3 hr Duration: 1.0 hr Energy: 8.48×10^{3} cal/sec (3.55×10^{4} W) Elevation: 10.0 m Warning time: 7.8 hr	24	7.38×10^{-6}	9.96×10^{-1}	5.2×10^{-1}	5.4×10^{-1}	2.6×10^{-3}/ 6.8×10^{-1}	3.4×10^{-2}	1.4×10^{-1}	5.5×10^{-5}	5.2×10^{-3}	2.1×10^{-2}
Early containment failure	Time: 3.056 hr Duration: 0.5 hr Energy: 1.696×10^{7} cal/sec (7.1×10^{7} W) Elevation: 10.0 m Warning time: 2.556 hr	7	1.60×10^{-7}	9.0×10^{-1}	7.4×10^{-2}	9.7×10^{-2}	1.4×10^{-2}/ 1.3×10^{-1}	1.5×10^{-2}	2.5×10^{-2}	8.1×10^{-6}	9.7×10^{-5}	8.7×10^{-3}
Late containment failure	Time: 8.33 hr Duration: 0.5 hr Energy: 8.48×10^{6} cal/sec (3.55×10^{7} W) Elevation: 10.0 m Warning time: 7.83 hr	9	2.46×10^{-6}	8.2×10^{-1}	2.3×10^{-6}	1.4×10^{-5}	1.6×10^{-5}/ 1.2×10^{-4}	3.2×10^{-4}	3.9×10^{-4}	1.8×10^{-11}	1.4×10^{-11}	1.3×10^{-5}
Interfacing systems LOCA[b]	Time: 5.56 hr Duration: 1.0 hr Energy: 8.48×10^{3} cal/sec (3.55×10^{4} W) Elevation: 10.0 m Warning time: 4.56 hr	23	2.40×10^{-7}	9.4×10^{-1}	2.9×10^{-1}	3.1×10^{-1}	1.6×10^{-5}/ 5.0×10^{-1}	2.3×10^{-1}	2.8×10^{-1}	3.6×10^{-4}	3.7×10^{-2}	1.5×10^{-1}

[a] McGuire data was used for the Catawba steam generator tube rupture event to compare similar scenarios.
[b] McGuire release duration, elevation, and warning time span were used for North Anna in lieu of plant-specific information.
Key: LOCA, loss-of-coolant accident; NA, not applicable; SG, steam generator.

result in lower consequences, but also have much lower probabilities, often resulting in risks several orders of magnitude lower. The other type of severe accident scenario for these reactors results in an intact containment. The risks from these events are several orders of magnitude lower than the risks from the bypass and failure scenarios. Therefore, a summation of the severe accident risks presented in the SPD EIS is a good indicator of overall risk.

Evacuation Information. This analysis conservatively assumes that 95 percent of the population within the 16-km (10-mi) emergency planning zone participated in an evacuation. It was also assumed that the five percent of the population that did not participate in the initial evacuation was relocated within 12 to 24 hr after plume passage, based on the measured concentrations of radioactivity in the surrounding area and the comparison of projected doses with Environmental Protection Agency (EPA) guidelines. Longer term countermeasures (e.g., crop or land interdiction) were based on EPA Protective Action Guides.

Each beyond-design-basis accident scenario has a warning time and a subsequent release time. The warning time is the time at which notification is given to offsite emergency response officials to initiate protective measures for the surrounding population. The release time is the time when the release to the environment begins. The minimum time between the warning time and the release time is one-half hour. The minimum time of one-half hour is enough time to evacuate onsite personnel (i.e., noninvolved workers). This also conservatively assumes that an onsite emergency has not been declared prior to initiating an offsite notification. Intact containment severe accident scenarios, which were not analyzed because of their insignificant offsite consequences, take place on an even longer time frame.

K.7.2.6 Accident Impacts

Accident impacts are presented in terms of increased risk. Increased risk is defined as the additional risk resulting from using a partial MOX core rather than an LEU core. For example, if the risk of an LCF from an accident with an LEU core is 1.0×10^{-6} and the risk of an LCF from the same accident with a MOX core is 1.1×10^{-6}, then the increased risk of an LCF is 1.0×10^{-7} ($1.1\times10^{-6} - 1.0\times10^{-6} = 1.0\times10^{-7}$).

Tables K–38 through K–43 present the consequences and risks of the postulated set of accidents at Catawba, McGuire, and North Anna, respectively. The receptors include a noninvolved worker located 640 m (0.4 mi) from the release point, the MEI, and the population within an 80-km (50-mi) radius of the reactor site. The consequences and risks are presented for both the current LEU-only and the proposed 40 percent MOX core configurations.

Table K–44 shows the ratios of accident impacts with the proposed 40 percent MOX core to the impacts with the current LEU core. This table shows that the increased risk from accidents to the surrounding population from a MOX core is, on average, less than 5 percent. For the fuel-handling accident at all three plants, the risk is reduced when using MOX fuel.

Severe accident scenarios that postulate large abrupt releases could result in prompt fatalities if the radiation dose is sufficiently high. Of the accidents analyzed in the SPD EIS, the ISLOCA and steam generator tube rupture at Catawba and McGuire, and the ISLOCA at North Anna were the only accidents that resulted in doses high enough to cause prompt fatalities. However, the number of prompt fatalities is expected to increase only for the ISLOCA scenarios. Table K–45 shows the estimated number of prompt fatalities estimated to result from these accidents.

Table K–38. Design Basis Accident Impacts for Catawba With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
Loss-of-coolant accident	7.50×10^{-6}	LEU	3.78	1.51×10^{-3}	1.81×10^{-7}	1.44	7.20×10^{-4}	8.64×10^{-8}	3.64×10^{3}	1.82	2.19×10^{-4}
		MOX	3.85	1.54×10^{-3}	1.86×10^{-7}	1.48	7.40×10^{-4}	8.88×10^{-8}	3.75×10^{3}	1.88	2.26×10^{-4}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.275	1.10×10^{-4}	1.78×10^{-7}	0.138	6.90×10^{-5}	1.10×10^{-7}	1.12×10^{2}	5.61×10^{-2}	8.98×10^{-5}
		MOX	0.262	1.05×10^{-4}	1.68×10^{-7}	0.131	6.55×10^{-5}	1.05×10^{-7}	1.10×10^{2}	5.48×10^{-2}	8.77×10^{-5}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary—given exposure (762 m [2,500 ft]) to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table K–39. Beyond-Design-Basis Accident Impacts for Catawba With LEU and MOX Fuels

			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities[c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
SG tube rupture[e]	6.31×10^{-10}	LEU	3.46×10^{2}	0.346	3.49×10^{-9}	5.71×10^{6}	5.20×10^{3}	5.25×10^{-5}
		MOX	3.67×10^{2}	0.367	3.71×10^{-9}	5.93×10^{6}	5.42×10^{3}	5.47×10^{-5}
Early containment failure	3.42×10^{-8}	LEU	5.97	2.99×10^{-3}	1.63×10^{-9}	7.70×10^{5}	4.62×10^{2}	2.53×10^{-4}
		MOX	6.01	3.01×10^{-3}	1.65×10^{-9}	8.07×10^{5}	4.84×10^{2}	2.66×10^{-4}
Late containment failure	1.21×10^{-5}	LEU	3.25	1.63×10^{-3}	3.15×10^{-7}	3.93×10^{5}	1.97×10^{2}	3.81×10^{-2}
		MOX	3.48	1.74×10^{-3}	3.38×10^{-7}	3.78×10^{5}	1.90×10^{2}	3.68×10^{-2}
ISLOCA	6.90×10^{-8}	LEU	1.40×10^{4}	1	1.10×10^{-6}	2.64×10^{7}	1.56×10^{4}	1.73×10^{-2}
		MOX	1.60×10^{4}	1	1.10×10^{-6}	2.96×10^{7}	1.69×10^{4}	1.87×10^{-2}

[a] Likelihood (or probability) of cancer fatality to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] McGuire timing and release fractions were used to compare like scenarios.

Key: ISLOCA, interfacing systems loss-of-coolant accident; LEU, low-enriched uranium; SG, steam generator.

Table K–40. Design Basis Accident Impacts for McGuire With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundaries			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
Loss-of-coolant accident	1.50×10^{-5}	LEU	5.31	2.12×10^{-3}	5.10×10^{-7}	2.28	1.14×10^{-3}	2.74×10^{-7}	3.37×10^{3}	1.69	4.06×10^{-4}
		MOX	5.46	2.18×10^{-3}	5.25×10^{-7}	2.34	1.17×10^{-3}	2.82×10^{-7}	3.47×10^{3}	1.74	4.18×10^{-4}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.392	1.57×10^{-4}	2.51×10^{-7}	0.212	1.06×10^{-4}	1.70×10^{-7}	99.1	4.96×10^{-2}	7.94×10^{-5}
		MOX	0.373	1.49×10^{-4}	2.38×10^{-7}	0.201	1.01×10^{-4}	1.62×10^{-7}	97.3	4.87×10^{-2}	7.79×10^{-5}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual–a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table K–41. Beyond-Design-Basis Accident Impacts for McGuire With LEU and MOX Fuels

			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	**Frequency (per year)**	**LEU or MOX Core**	**Dose (rem)**	**Probability of Latent Cancer Fatality[a]**	**Risk of Latent Cancer Fatality (over campaign)[b]**	**Dose (person-rem)**	**Latent Cancer Fatalities[c]**	**Risk of Latent Cancer Fatalities (over campaign)[d]**
SG tube rupture[e]	5.81×10^{-9}	LEU	6.10×10^{2}	0.610	5.66×10^{-8}	5.08×10^{6}	4.65×10^{3}	4.32×10^{-4}
		MOX	6.47×10^{2}	0.647	6.02×10^{-8}	5.28×10^{6}	4.85×10^{3}	4.51×10^{-4}
Early containment failure	9.89×10^{-8}	LEU	12.2	6.10×10^{-3}	9.65×10^{-9}	7.90×10^{5}	4.57×10^{2}	7.23×10^{-4}
		MOX	12.6	6.30×10^{-3}	9.97×10^{-9}	8.04×10^{5}	4.67×10^{2}	7.39×10^{-4}
Late containment failure	7.21×10^{-6}	LEU	2.18	1.09×10^{-3}	1.26×10^{-7}	3.04×10^{5}	1.52×10^{2}	1.76×10^{-2}
		MOX	2.21	1.11×10^{-3}	1.28×10^{-7}	2.96×10^{5}	1.48×10^{2}	1.71×10^{-2}
ISLOCA	6.35×10^{-7}	LEU	1.95×10^{4}	1	1.02×10^{-5}	1.79×10^{7}	1.19×10^{4}	0.121
		MOX	2.19×10^{4}	1	1.02×10^{-5}	1.97×10^{7}	1.27×10^{4}	0.129

[a] Likelihood (or probability) of cancer fatality to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (762 m [2,500 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of cancer fatalities over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
[e] McGuire timing and release fractions were used to compare like scenarios.

Key: ISLOCA, interfacing systems loss-of-coolant accident; LEU, low-enriched uranium; SG, steam generator.

Table K–42. Design Basis Accident Impacts for North Anna With LEU and MOX Fuels

			Impacts on Noninvolved Worker			Impacts at Site Boundary			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
Loss-of-coolant accident	2.10×10^{-5}	LEU	0.114	4.56×10^{-5}	1.53×10^{-8}	3.18×10^{-2}	1.59×10^{-5}	5.34×10^{-9}	39.4	1.97×10^{-2}	6.62×10^{-6}
		MOX	0.115	4.60×10^{-5}	1.55×10^{-8}	3.20×10^{-2}	1.60×10^{-5}	5.38×10^{-9}	40.3	2.02×10^{-2}	6.78×10^{-6}
Spent-fuel-handling accident[e]	1.00×10^{-4}	LEU	0.261	1.04×10^{-4}	1.66×10^{-7}	9.54×10^{-2}	4.77×10^{-5}	7.63×10^{-8}	29.4	1.47×10^{-2}	2.35×10^{-5}
		MOX	0.239	9.56×10-5	1.53×10^{-7}	8.61×10^{-2}	4.31×10^{-5}	6.90×10^{-8}	27.5	1.38×10^{-2}	2.21×10^{-5}

[a] Likelihood (or probability) of cancer fatality for a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft])—given exposure to the indicated dose.

[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—a noninvolved worker at a distance of 640 m (2,100 ft) or the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft]).

[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.

[d] Risk of a cancer fatality over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).

[e] Postulated design basis accidents at commercial reactors are considered extremely unlikely events. They are estimated to have a frequency of between 1.0×10^{-4} and 1.0×10^{-6} per year. Because a spent-fuel-handling accident does not have a calculated frequency associated with it, it has been estimated to have the highest frequency for the purposes of this analysis.

Key: LEU, low-enriched uranium.

Table K–43. Beyond-Design-Basis Accident Impacts for North Anna With LEU and MOX Fuels

			Impacts on Site Boundary			Impacts on Population Within 80 km		
Accident	Frequency (per year)	LEU or MOX Core	Dose (rem)	Probability of Latent Cancer Fatality[a]	Risk of Latent Cancer Fatality (over campaign)[b]	Dose (person-rem)	Latent Cancer Fatalities[c]	Risk of Latent Cancer Fatalities (over campaign)[d]
SG tube rupture[e]	7.38×10^{-6}	LEU	2.09×10^{2}	0.209	2.46×10^{-5}	1.73×10^{6}	1.22×10^{3}	0.144
		MOX	2.43×10^{2}	0.243	2.86×10^{-5}	1.84×10^{6}	1.33×10^{3}	0.157
Early containment failure[e]	1.60×10^{-7}	LEU	19.6	1.96×10^{-2}	5.02×10^{-8}	8.33×10^{5}	4.52×10^{2}	1.16×10^{-3}
		MOX	21.6	2.16×10^{-2}	5.54×10^{-8}	8.42×10^{5}	4.61×10^{2}	1.18×10^{-3}
Late containment failure[e]	2.46×10^{-6}	LEU	1.12	5.60×10^{-4}	2.21×10^{-8}	4.04×10^{4}	20.2	7.95×10^{-4}
		MOX	1.15	5.75×10^{-4}	2.26×10^{-8}	4.43×10^{4}	22.1	8.70×10^{-4}
ISLOCA[e]	2.40×10^{-7}	LEU	1.00×10^{4}	1	3.84×10^{-6}	4.68×10^{6}	2.98×10^{3}	1.14×10^{-2}
		MOX	1.22×10^{4}	1	3.84×10^{-6}	5.41×10^{6}	3.39×10^{3}	1.30×10^{-2}

[a] Likelihood (or probability) of cancer fatality to a hypothetical individual—the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft])—given exposure to the indicated dose.
[b] Risk of cancer fatality over the estimated 16-year campaign to a hypothetical individual—the maximally exposed offsite individual at the site boundary (1,349 m [4,426 ft]).
[c] Estimated number of cancer fatalities in the entire offsite population out to a distance of 80 km (50 mi) given exposure to the indicated dose.
[d] Risk of cancer fatalities over the estimated 16-year campaign in the entire offsite population out to a distance of 80 km (50 mi).
[e] McGuire release durations and warning times were used in lieu of site specific data.
Key: ISLOCA, interfacing systems loss-of-coolant accident; LEU, low-enriched uranium; SG, steam generator.

Table K–44. Ratio of Accident Impacts for MOX-Fueled and LEU-Fueled Reactors (MOX Impacts/Uranium Impacts)

	Catawba			McGuire			North Anna		
Accident	Worker	MEI	Population	Worker	MEI	Population	Worker	MEI	Population
LOCA	1.019	1.028	1.033	1.028	1.026	1.030	1.009	1.006	1.025
FHA	0.953	0.949	0.977	0.952	0.948	0.982	0.916	0.903	0.939
SGTR	NA	1.061	1.042	NA	1.061	1.043	NA	1.163	1.090
Early	NA	1.007	1.048	NA	1.033	1.022	NA	1.102	1.020
Late	NA	1.071	0.964	NA	1.014	0.974	NA	1.027	1.094
ISLOCA	NA	1.143	1.083	NA	1.123	1.067	NA	1.220	1.138

Key: Early, early containment; FHA, fuel-handling accident; ISLOCA, interfacing systems loss-of-coolant accident; Late, late containment; LEU, low-enriched uranium; LOCA, loss-of-coolant accident; MEI, maximally exposed individual; NA, not applicable; SGTR, steam generator tube rupture.

K.7.2.6.1 Catawba

Design Basis Accidents. Table K–38 shows the risks and consequences associated with a LOCA and spent-fuel-handling accident at Catawba. The greatest risk increase to the surrounding population for a design basis accident with a MOX core configuration is approximately 3.3 percent from the LOCA. If this accident were to occur, the consequences in terms of LCFs in the surrounding population within 80 km (50 mi) would be 1.82 LCFs for an LEU core and 1.88 LCFs for a partial MOX core. The increased risk, in terms of an LCF, to the noninvolved worker is 1 in 200 million (5.0×10^{-9}) per 16-year campaign; the MEI, one 1 in 420 million (2.4×10^{-9}) per 16-year campaign; and the population, 1 in 140,000 (7.0×10^{-6}) per 16-year campaign.

Table K–45. Prompt Fatalities for MOX-Fueled and LEU-Fueled Reactors

Accident Scenario	LEU	MOX
Steam generator tube rupture		
Catawba	1	1
McGuire	1	1
North Anna	0	0
Interfacing systems loss-of-coolant accident		
Catawba	815	843
McGuire	398	421
North Anna	54	60

Key: LEU, low-enriched uranium.

Beyond-Design-Basis Accidents. Table K–39 shows the risks and consequences associated with four beyond-design-basis accidents at Catawba. Table K–45 shows prompt fatalities. The greatest risk increase to the surrounding population from a beyond-design-basis accident with a MOX core configuration is approximately 8.3 percent from the ISLOCA. If this accident were to occur, the consequences in terms of LCFs and prompt fatalities in the surrounding population within 80 km (50 mi) would be approximately 16,400 fatalities for an LEU core and 17,700 fatalities for a partial MOX core. The increased risk, in terms of an LCF, to the population is 1 in 710 (1.4×10^{-3}) per 16-year campaign. The increased risk of a prompt fatality is 1 in 32,000 (3.1×10^{-5}) per 16-year campaign.

K.7.2.6.2 McGuire

Design Basis Accidents. Table K–40 shows the risks and consequences associated with a LOCA and spent-fuel-handling accident at McGuire. The greatest risk increase to the surrounding population for a design basis accident with a MOX core configuration is 3.0 percent from the LOCA. If this accident were to occur, the consequences in terms of LCFs in the surrounding population within 80 km (50 mi) would be 1.69 LCFs for an LEU core and 1.74 LCFs for a partial MOX core. The increased risk, in terms of an LCF, to the noninvolved worker is 1 in 67 million (1.5×10^{-8}) per 16-year campaign; the MEI, 1 in 120 million (8.0×10^{-9}) per 16-year campaign; and the population, 1 in 83,000 (1.2×10^{-5}) per 16-year campaign.

Beyond-Design-Basis Accidents. Table K–41 shows the risks and consequences associated with four beyond-design-basis accidents at McGuire. Table K–45 shows prompt fatalities. The greatest risk increase to the surrounding population for a beyond-design-basis accident with a MOX core configuration is approximately 6.6 percent from the ISLOCA. If this accident were to occur, the consequences in terms of LCFs and prompt fatalities in the surrounding population within 80 km (50 mi) would be approximately 12,300 fatalities with an LEU core and 13,100 with a partial MOX core. The increased risk of an LCF to the population is 1 in 120 (8.0×10^{-3}) per 16-year campaign. The increased risk of a prompt fatality is 1 in 4,300 (2.3×10^{-4}) per 16-year campaign.

K.7.2.6.3 North Anna

Design Basis Accidents. Table K–42 shows the risks and consequences associated with a LOCA and spent-fuel-handling accident at North Anna. The greatest risk increase to the surrounding population for a design-basis-accident with a MOX core configuration is approximately 2.5 percent from the LOCA. If this accident were to occur, the consequences in terms of LCFs in the surrounding population within 80 km (50 mi) would be 1.97×10^{-2} LCF for an LEU core and 2.02×10^{-2} LCF for a partial MOX core. The increased risk, in

terms of an LCF, to the noninvolved worker is 1 in 5.0 billion (2.0×10^{-10}) per 16-year campaign; the MEI, 1 in 25 billion (4.0×10^{-11}) per 16-year campaign; and the population, 1 in 6.2 million (1.6×10^{-7}) per 16-year campaign.

Beyond-Design-Basis Accidents. Table K–43 shows the risks and consequences associated with four beyond-design-basis accidents at North Anna. Table K–45 shows prompt fatalities. The greatest risk increase to the surrounding population from a beyond-design-basis accident with a MOX core configuration is approximately 14 percent from the ISLOCA. If this accident were to occur, the consequences in terms of LCFs and prompt fatalities in the surrounding populations within 80 km (50 mi) would be approximately 3,000 fatalities for an LEU core and 3,450 fatalities for a partial MOX core. The increased risk of an LCF to the population is 1 in 620 (1.6×10^{-3}) per 16-year campaign. The increased risk of a prompt fatality is 1 in 43,000 (2.3×10^{-5}) per 16-year campaign.

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Appendix L
Evaluation of Human Health Effects From Transportation

L.1 INTRODUCTION

The overland transportation of any commodity involves a risk to both transportation crew members and members of the public. This risk results directly from transportation-related accidents and indirectly from the increased levels of pollution from vehicle emissions, regardless of the cargo. The transportation of certain materials, such as hazardous or radioactive waste, can pose an additional risk due to the unique nature of the material. In order to permit a complete appraisal of the environmental impacts of the proposed action and alternatives, the human health risks associated with the overland transportation of plutonium and other hazardous materials have been assessed.

This appendix provides an overview of the approach used to assess the human health risks that may result from the overland transportation. The appendix includes a discussion of the scope of the assessment, analytical methods used for the risk assessment (i.e., computer models), important assessment assumptions, and a determination of potential transportation routes. It also presents the results of the assessment. In addition, to aid in the understanding and interpretation of the results, specific areas of uncertainty are described, with an emphasis on how the uncertainties may affect comparisons of the alternatives.

The approach used in this appendix is modeled after that used in the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic Environmental Impact Statement* (PEIS) (DOE 1996a). The fundamental assumptions used in the analysis for the *Surplus Plutonium Disposition Environmental Impact Statement* are consistent with those used in the PEIS, and the same computer codes and generic release and accident data are used.

The risk assessment results are presented in this appendix in terms of "per-shipment" risk factors, as well as for the total risks associated with each alternative. Per-shipment risk factors provide an estimate of the risk from a single hazardous material shipment between a specific origin and destination. The total risks for a given alternative are found by multiplying the expected number of shipments by the appropriate per-shipment risk factors.

L.2 SCOPE OF ASSESSMENT

The scope of the overland transportation human health risk assessment, including the alternatives and options, transportation activities, potential radiological and nonradiological impacts, transportation modes considered, and receptors, is described below. Additional details of the assessment are provided in the remaining sections of the appendix.

- Proposed Action and Alternatives—The transportation risk assessment conducted for the SPD EIS estimates the human health risks associated with the transportation of plutonium and other hazardous materials for a number of disposition alternatives.

- Radiological Impacts—For each alternative, radiological risks (i.e., those risks that result from the radioactive nature of the plutonium and other hazardous materials) are assessed for both incident-free (i.e., normal) and accident transportation conditions. The radiological risk associated with incident-free transportation conditions would result from the potential exposure of people to external radiation in the vicinity of a loaded shipment. The radiological risk from transportation accidents would come from the potential release and dispersal of radioactive material into the environment during an accident and the

subsequent exposure of people through multiple exposure pathways (i.e., exposure to contaminated ground or air, or ingestion of contaminated food).

- All radiological impacts are calculated in terms of effective dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent, which is the sum of the effective dose equivalent from external radiation exposure and the 50-year committed effective dose equivalent from internal radiation exposure (NRC 1998). Radiation doses are presented in units of roentgen equivalent man (rem) for individuals and person-rem for collective populations. The impacts are further expressed as health risks in terms of latent cancer fatalities (LCFs) and cancer incidence in exposed populations. The health risk conversion factors (expected health effects per dose absorbed) were taken from the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991).

- Nonradiological Impacts—In addition to the radiological risks posed by overland transportation activities, vehicle-related risks are also assessed for nonradiological causes (i.e., related to the transport vehicles and not the radioactive cargo) for the same transportation routes. The nonradiological transportation risks are independent of the radioactive nature of the cargo and would be incurred for similar shipments of any commodity. The nonradiological risks are assessed for both incident-free and accident conditions. Nonradiological risks during incident-free transportation conditions would be caused by potential exposure to increased vehicle exhaust emissions. The nonradiological accident risk refers to the potential occurrence of transportation accidents that directly result in fatalities unrelated to the cargo. State-specific transportation fatality rates are used in the assessment. Nonradiological risks are presented in terms of estimated fatalities.

- Transportation Modes—All overland shipments were assumed to take place by truck.

- Receptors—Transportation-related risks are calculated and presented separately for workers and members of the general public. The workers considered are truck crew members involved in the actual overland transportation. The general public includes all persons who could be exposed to a shipment while it is moving or stopped enroute. Potential risks are estimated for the collective populations of exposed people, as well as for the hypothetical maximally exposed individual. The collective population risk is a measure of the radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing various alternatives.

L.3 PACKAGING AND REPRESENTATIVE SHIPMENT CONFIGURATIONS

Regulations that govern the transportation of radioactive materials are designed to protect the public from the potential loss or dispersal of radioactive materials as well as from routine radiation doses during transit. The primary regulatory approach to promote safety is through the specification of standards for the packaging of radioactive materials. Because packaging represents the primary barrier between the radioactive material being transported and radiation exposure to the public and the environment, packaging requirements are an important consideration for the transportation risk assessment. Regulatory packaging requirements are discussed briefly below and in Chapter 5. In addition, the representative packaging and shipment configurations assumed for the SPD EIS are described.

L.3.1 Packaging Overview

Although several Federal and State organizations are involved in the regulation of radioactive materials transportation, primary regulatory responsibility resides with the U.S. Department of Transportation (DOT) and the U.S. Nuclear Regulatory Commission (NRC). All transportation activities must take place in accordance with

the applicable regulations of these agencies specified in Title 49 of the Code of Federal Regulations (CFR) Part 173 (DOT 1992a) and 10 CFR 71 (NRC 1996).

Transportation packaging for small quantities of radioactive materials must be designed, constructed, and maintained to contain and shield their contents during normal transport conditions. For large quantities and for more highly radioactive material, such as spent nuclear fuel or plutonium, they must contain and shield their contents in the event of severe accident conditions. The type of packaging used is determined by the total radioactive hazard presented by the material within the packaging; 10 CFR 71 (NRC 1996) provides the rules for this determination. Four basic types of packaging are used: Excepted, Industrial, Type A, and Type B. Another packaging option, Strong and Tight, is still available for some domestic shipments.

Excepted packagings are limited to transporting materials with extremely low levels of radioactivity. Industrial packagings are used to transport materials that, because of their low concentration of radioactive materials, present a limited hazard to the public and the environment. Type A packagings are designed to protect and retain their contents under normal transport conditions and must maintain sufficient shielding to limit radiation exposure to handling personnel. These packagings are used to transport radioactive materials with higher concentrations or amounts of radioactivity than Excepted or Industrial packagings. Strong and Tight packagings are used in the United States for shipment of certain materials with low levels of radioactivity, such as natural uranium and rubble from the decommissioning of nuclear reactors. Type B packages are described in detail in Appendix L.3.1.6.

L.3.1.1 Uranium Hexafluoride Packaging

DOE would ship uranium hexafluoride in a commercial vehicle from the Portsmouth Gaseous Diffusion Plant to a fuel fabrication facility in Model 30B cylinders, which are Type A packages (for the purposes of the SPD EIS). Uranium hexafluoride shipments are regulated under 49 CFR 173.420, which requires the packaging to be in accordance with ANSI N14.1, *Uranium Hexafluoride–Packaging for Transport*. Because uranium hexafluoride breaks down into hydrofluoric acid and uranyl fluoride when exposed to air, packages would be marked with the primary hazard label as "Radioactive Yellow-II" and a secondary hazard label as "Corrosive." The transport vehicle would be required to show the primary placard "Radioactive" and the secondary placard "Corrosive."

L.3.1.2 Uranium Dioxide Packaging

DOE would ship uranium dioxide in a commercial vehicle from the fuel fabrication facility to DOE's mixed oxide (MOX) facility in gasketed, open-head, 208-l (55-gal) drums with heavy plastic liners, which are Industrial Package Type 1 packages. Uranium dioxide shipments are regulated under 49 CFR 173.425. Because uranium dioxide is a low-specific-activity material, no primary hazard label would be required, and because it is chemically stable, no secondary hazard label would be required. The transport vehicle would be required to show the primary placard "Radioactive" and no secondary placard.

L.3.1.3 MOX Fuel Packaging

DOE will design the container for the MOX fuel assemblies. For analysis purposes, it is assumed that DOE would ship the unirradiated MOX fuel bundles in a safe, secure trailer/SafeGuards Transport (SST/SGT) to the reactor site(s) in Type B packages. Two conceptual packaging ideas are end-loading and lateral-loading packages (Ludwig et al. 1997). The fuel assembly weight per container is approximately 2800 kg (6,000 lb) for either pressurized water reactor (PWR) or boiling water reactor (BWR) fuel. The container could hold either four PWR or eight BWR assemblies.

L.3.1.4 Highly Enriched Uranium Packaging

DOE would ship highly enriched uranium (HEU) in an SST/SGT from the pit conversion facility to the Y–12 facility near Oak Ridge, Tennessee. The DOE-approved container type for these shipments is the DT–22.

L.3.1.5 Plutonium Packaging

DOE would ship all plutonium in Type B containers. DOE would ship nonpit plutonium in an SST/SGT from DOE sites (Hanford, Idaho National Engineering and Environmental Laboratory [INEEL], Lawrence Livermore National Laboratory [LLNL], Los Alamos National Laboratory [LANL], Rocky Flats Environmental Technology Site [RFETS], and Savannah River Site [SRS]) to the immobilization facility (Hanford or SRS) in a variety of containers, such as Type 3013, Type 2R, and Foodpac containers, which would be transported inside various casks, such as radial reflector, SAFEKEG (Type 9517), Model 60 FFTA DFA pins shipping or Specification 6M packages. DOE would ship plutonium pits from DOE sites to the pit conversion facility in DOE-approved FL containers and the piece parts resulting from pit disassembly in DOE-approved UC–609 and USA/9975 containers. Plutonium dioxide produced at the pit conversion facility would be loaded into packaging that meets DOE-STD-3013-96, *Criteria for Preparing and Packaging Plutonium Metals and Oxides for Long-Term Storage* (DOE 1996b) or equivalent. This package provides for safe storage of plutonium oxides for at least 50 years or until final disposition and serves as the primary containment vessel for shipping. DOE-STD-3013-96 specifies a design goal that the Type 3013 container could be shipped in a qualified shipping container without further reprocessing or repackaging. The Type 3013 primary containment vessel is designed for shipping and would be compatible with a Type B package. No Type B package has been specifically constructed or licensed for shipping DOE-STD-3013-96 primary containment vessels.

A Type B package is required when transporting commercial quantities of plutonium materials, including unirradiated MOX fuel assemblies. DOE is developing a conceptual design for a MOX container that optimizes SST/SGT load-carrying capacity and ensures compatibility with fuel-handling systems at commercial reactors (Ludwig et al. 1997).

L.3.1.6 Overview of Type B Containers

The transportation of highway-route controlled quantities of plutonium (more than a few grams, depending on activity level) requires the use of Type B packaging. In addition to meeting the standards for Type A packaging, Type B packaging must provide a high degree of assurance that, even in severe accidents, the integrity of the package will be maintained with essentially no loss of the radioactive contents or serious impairment of the shielding and maintain subcriticality capability. Type B packaging must satisfy stringent testing criteria specified in 10 CFR 71 (NRC 1996). The testing criteria were developed to simulate severe accident conditions, including impact, puncture, fire, and water immersion.

Beyond meeting DOT standards showing it can withstand normal conditions of transport without loss or dispersal of its radioactive contents or allowance of significant radiation fields, Type B packaging must also meet the 10 CFR 71 requirements administered by the NRC. The complete sequence of tests is listed below:

- Free-Drop Test—A 9-m (30-ft) free-drop onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage to the package is expected.

- Puncture Test—A 1-m (40-in) drop onto the upper end of a 15-cm (6-in) diameter solid, vertical, cylindrical, mild steel bar (at least 20-cm [8-in] long) mounted on an essentially unyielding, horizontal surface.

- Thermal Test—Exposure to a heat flux of no less than that of a thermal radiation environment of 800 °C (1,475 °F) with an emissivity coefficient of at least 0.9 for a period of 30 minutes.

- Water Immersion Test—A separate, undamaged package specimen is subjected to water pressure equivalent to immersion under a head of water of at least 15-m (50-ft) for no less than 8 hours.

Effective April 1, 1996, 10 CFR 71 was revised to require an additional immersion test in 200 m (660 ft) of water for Type B casks designed to contain material with activity levels greater than 1 million curies (Ci) (NRC 1996). Containers used for shipping plutonium will not necessarily be subject to this test because they will contain much less than one million curies. The packaging may also be required to undergo the crush test if it is considered a light-weight, low-density package as most drum-type packages are. The crush test consists of dropping a 500-kg (1100-lb) steel plate from 9 m (30 ft) onto the package, which is resting on an essentially unyielding surface.

Additional restrictions apply to package surface contamination levels, but these restrictions are not limiting for the transportation radiological risk assessment. For risk assessment purposes, it is important to note that all packaging of a given type is designed to meet the same performance criteria. Therefore, two different Type B designs would be expected to perform similarly during incident-free and accident transportation conditions. The specific containers selected, however, will determine the total number of shipments necessary to transport a given quantity of plutonium.

External radiation from a package must be below specified limits that minimize the exposure of the handling personnel and general public. For these types of shipments, the external radiation dose rate during normal transportation conditions must be maintained below the following limits of 49 CFR 173 (DOT 1992a):

- 10 mrem/hr at any point 2 m (6.6 ft) from the vertical planes projected by the outer lateral surfaces of the transport vehicle (referred to as the regulatory limit throughout this document)

- 2 mrem/hr in any normally occupied position in the transport vehicle

L.3.2 Safe, Secure Transportation

DOE anticipates that any transportation of plutonium pits, nonpit plutonium, plutonium dioxide, MOX fuel, or HEU would be required to be made through use of the Transportation Safeguards System and shipped using SST/SGTs. The SST/SGT is a fundamental component of the Transportation Safeguards System. The Transportation Safeguards System is operated by the DOE Transportation Safeguards Division of the Albuquerque Operations Office for the DOE Headquarters Office of Defense Programs. Based on operational experience between FY84 and FY98, the mean probability of an accident requiring the tow-away of the SST/SGT was 0.058 accident per million kilometers (0.096 accident per million miles). By contrast, the rate for commercial trucking in 1989 was about 0.3 accident per million kilometers (0.5 accident per million miles). Commercial trucking accident rates (Saricks and Kvitek 1994) were used in the human health effects analysis. Since its establishment in 1975, the Transportation Safeguards Division has accumulated more than 151 million km (94 million mi) of over-the-road experience transporting DOE-owned cargo with no accidents resulting in a fatality or release of radioactive material.

The SST/SGT is a specially designed component of an 18-wheel tractor-trailer vehicle. Although details of vehicle enhancements and some operational aspects are classified, key characteristics of the SST/SGT system include the following:

- Enhanced structural characteristics and a highly reliable tie-down system to protect cargo from impact
- Heightened thermal resistance to protect the cargo in case of fire (newer SST/SGT models)
- Established operational and emergency plans and procedures governing the shipment of nuclear materials
- Various deterrents to prevent unauthorized removal of cargo
- An armored tractor component that provides courier protection against attack and contains advanced communications equipment
- Specially designed escort vehicles containing advanced communications and additional couriers
- 24-hour-a-day real-time communications to monitor the location and status of all SST/SGT shipments via DOE's Security Communication system
- Couriers who are armed Federal Officers, receive rigorous specialized training, and who are closely monitored through DOE's Personnel Assurance Program
- Significantly more stringent maintenance standards than those for commercial transport equipment
- Conduct of periodic appraisals of the Transportation Safeguards System operations by the DOE Office of Defense Programs to ensure compliance with DOE orders and management directives, and continuous improvement in transportation and emergency management programs

L.3.3 Ground Transportation Route Selection Process

According to DOE guidelines, plutonium shipments must comply with both NRC and DOT regulatory requirements. Commercial shipments are also required by law to comply with both NRC and DOT requirements. NRC regulations cover the packaging and transport of plutonium, whereas DOT specifically regulates the carriers and the conditions of transport, such as routing, handling and storage, and vehicle and driver requirements. The highway routing of nuclear material is systematically determined according to DOT regulations 49 CFR 171–179 and 49 CFR 397 for commercial shipments. The dates and times that specific transportation routes would be used are classified information and would not be publicized before a shipment.

The DOT routing regulations require that a shipment of a "highway route-controlled quantity" of radioactive material be transported over a preferred highway network including interstate highways, with preference toward interstate system bypasses around cities, and State-designated preferred routes. A State or tribe may designate a preferred route to replace or supplement the interstate highway system in accordance with DOT guidelines (DOT 1992b).

Carriers of highway route-controlled quantities are required to use the preferred network, unless moving from origin to the nearest interstate or from the interstate to the destination, when making necessary repair or rest stops, or when emergency conditions render the interstate unsafe or impassible. The primary criterion for selecting the preferred route for a shipment is travel time. Preferred routing takes into consideration accident rate, transit time, population density, activities, time of day, and day of week.

The HIGHWAY computer code (Johnson et al. 1993) may be used for selecting highway routes in the United States. The HIGHWAY database is a computerized road atlas that currently describes about 386,400 km (240,000 mi) of roads. The Interstate System and all U.S. (U.S.-designated) highways are completely described in the database. In addition, most of the principal State highways and many local and community roads are also identified. The code is updated periodically to reflect current road conditions and has been benchmarked against reported mileages and observations of commercial truck firms. Features in the HIGHWAY code allow the user to select routes that conform to DOT regulations. Additionally, the HIGHWAY code contains data on the population densities along the routes. The distance and population data from the HIGHWAY code are part of the information used for the transportation impact analysis in the SPD EIS.

L.4 METHODS FOR CALCULATING TRANSPORTATION RISKS

The overland transportation risk assessment methodology is summarized in Figure L–1. After the alternatives were identified and goals of the shipping campaign were understood, the first step was to collect data on material characteristics and accident parameters. Physical, radiological, and packaging data were provided in reports from the DOE national laboratories. Accident parameters are largely based on the DOE-funded study of transportation accidents (Saricks and Kvitek 1994).

Representative routes that may be used for the shipment of plutonium were selected using the HIGHWAY code. These routes were selected for risk assessment purposes. They do not necessarily represent the actual routes that would be used to transport nuclear materials. Specific routes cannot be identified in advance because the routes would not be finalized until DOE has actually planned the shipping campaign. The selection of the actual route would be responsive to environmental and other conditions that would be in effect or could be predicted at the time of shipment. Such conditions could include adverse weather conditions, road conditions, bridge closures, and local traffic problems. For security reasons, details about a planned shipment would not be publicized before the shipment.

The first analytic step in the ground transportation analysis was to determine the incident-free and accident risk factors, on a per-shipment basis, for transportation. Risk factors, as any risk estimate, are the product of the probability of exposure and the magnitude of the exposure. Accident risk factors were calculated for radiological and nonradiological traffic accidents. The probabilities, which are much lower than 1, and the magnitudes of exposure were multiplied, yielding risk numbers. Incident-free risk factors were calculated for crew and public exposure to radiation emanating from the shipping container (cask) and public exposure to the chemical toxicity of the transportation vehicle exhaust. The probability of incident-free exposure is unity (one).

Radiological risk factors are expressed in units of rem. Later in the analysis, they are multiplied by the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991) conversion factors and estimated number of shipments to give risk estimates in units of LCFs. The vehicle emission risk factors are calculated in LCFs, and the vehicle accident risk factors are calculated in fatalities.

For each alternative, risks were assessed for both incident-free transportation and accident conditions. For the incident-free assessment, risks were calculated for collective populations of potentially exposed individuals and for maximally exposed individuals. The accident assessment consists of two components: (1) a probabilistic accident risk assessment that considers the probabilities and consequences of a range of possible transportation accident environments, including low-probability accidents that have high consequences and

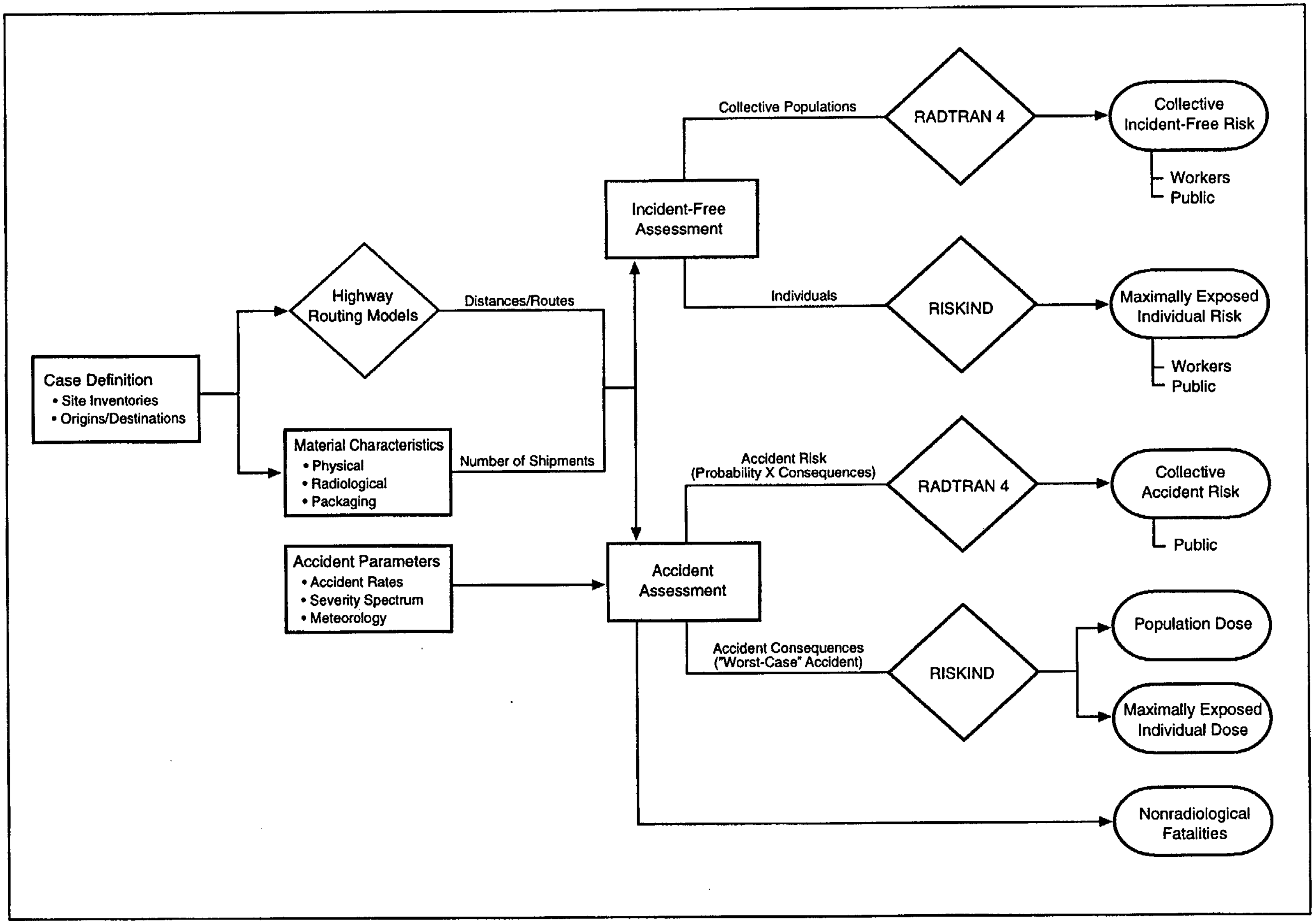

Figure L–1. Overland Transportation Risk Assessment

high-probability accidents that have low consequences, and (2) an accident consequence assessment that considers only the consequences of the most severe transportation accidents postulated.

The RADTRAN 4 computer code (Neuhauser and Kanipe 1995) is used for incident-free and accident risk assessments to estimate the impacts on collective populations. RADTRAN 4 was developed by Sandia National Laboratories to calculate population risks associated with the transportation of radioactive materials by a variety of modes, including truck, rail, air, ship, and barge.

The RADTRAN 4 population risk calculations take into account both the consequences and probabilities of potential exposure events. The collective population risk is a measure of the total radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing the various alternatives. The RISKIND computer code (Yuan et al. 1995) is used to estimate the incident-free doses to maximally exposed individuals and for estimating impacts for the accident consequence assessment. The RISKIND computer code was developed for DOE's Office of Civilian Radioactive Waste Management to analyze the exposure of individuals during incident-free transportation. In addition, the RISKIND code was designed to allow a detailed assessment of the consequences to individuals and population subgroups from severe transportation accidents under various environmental settings.

The RISKIND calculations were conducted to supplement the collective risk results calculated with RADTRAN 4. Whereas the collective risk results provide a measure of the overall risks of each alternative, the RISKIND calculations are meant to address areas of specific concern to individuals and population subgroups. Essentially, the RISKIND analyses are meant to address "What if" questions, such as "What if I live next to a site access road?" or "What if an accident happens near my town?"

If highly specialized analytic codes had been used to model SST/SGT behavior in an accident (*DOE-Developed Analysis of Dispersal Risk Occurring in Transportation* or ADROIT [Clauss et al. 1995:689–696]), the code would have provided a probabilistic risk analysis of special nuclear materials shipped in an SST/SGT. ADROIT is designed to provide a focused analysis of a release caused by partial detonation of explosive material. The approach and the code could be tailored for the materials shipped as part of the surplus plutonium disposition program. However, detailed thermal and mechanical models have not been created for most of the packages used in the SPD EIS.

L.5 ALTERNATIVES, PARAMETERS, AND ASSUMPTIONS

The transportation risk assessment is designed to ensure—through uniform and judicious selection of models, data, and assumptions—that relative comparisons of risk among the various alternatives are meaningful. The major input parameters and assumptions used in the transportation risk assessment are discussed below.

L.5.1 Transportation Alternatives

The proposed action would involve transporting plutonium and other nuclear materials between DOE and commercial sites. Except for the No Action Alternative, each alternative in the SPD EIS has extensive and unique requirements for the transportation of hazardous materials. In this section, the assumptions and logic used to model the intersite transportation requirements are described.

Alternatives 2 through 12 require transporting plutonium metal and pits from various DOE sites to the pit conversion facility at Hanford, INEEL, Pantex, or SRS. The pit conversion facility would disassemble pits and convert the plutonium metal into plutonium dioxide. During the pit disassembly process, HEU would be recovered and shipped from the pit conversion facility to the Y–12 facility at Oak Ridge. In addition, some pit parts would be recovered and shipped to LANL. The plutonium dioxide would be shipped to the MOX facility

or the immobilization facility depending on the alternative. In many of the alternatives, the pit conversion facility is located on the same site as the MOX facility or immobilization facility, limiting the need for intersite transportation of the plutonium dioxide. In these alternatives, the plutonium dioxide would be transported between the facilities via a secure tunnel between the facilities.

In addition to reducing the number of trips required and the distance that would have to be traveled to transport surplus pits to the pit conversion facility, by placing the pit conversion facility at Pantex the dose associated with repackaging pits for intersite shipment could be reduced by nearly 40 percent. This is because pits can be transferred to the pit conversion facility at Pantex in their current storage containers (mainly the AL–R8 container) without having to be repackaged. If the pits are transported to another site, they have to be moved to a shipping container (e.g., FL-type, 9975).

Based on estimates presented in the *Final EIS for the Continued Operation of Pantex and Associated Storage of Nuclear Weapons Components (Pantex Sitewide EIS)* (DOE 1996c), about 50 workers would be needed to repackage approximately 13,000 pits from their current storage containers into containers that could also be used for shipping.[1] Work is currently under way to repackage pits from the AL–R8 container into the AL–R8 sealed insert (SI) container as discussed in the *Supplement Analysis for the Final Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapons Components—AL–R8 Sealed Insert Container* (DOE 1998). This effort could be completed over 10 years, and the estimated annual dose received from repackaging activities would be about 208 mrem per worker (Low 1999). By locating the pit conversion facility at Pantex, it is expected that the additional dose associated with repackaging the surplus pits into shipping containers could be avoided. This would effectively reduce the total expected dose for these activities by 50 percent. If the pit conversion facility were sited at Pantex, the pits would be slowly moved from storage locations in storage containers on specially designed vehicles to the pit conversion facility instead of having to be put into offsite shipping containers. Over the 10-year operating life of the pit conversion facility, this would reduce the total estimated dose to involved Pantex transportation and staging workers by 104 person-rem from 208 person-rem to 104 person-rem.[2] Under either scenario, the estimated number of excess cancer fatalities associated with repackaging activities would be 0.1 or less.

In August 1998, DOE prepared a supplement analysis (DOE 1998) for the *Pantex Sitewide EIS* that compares all environmental impact parameters to those analyzed in the *Pantex Sitewide EIS* and final determinations made in the Record of Decision that was signed on January 17, 1997, with respect to the use of the AL–R8 SI. Results of the analysis indicated that both the AT–400A container and the modified AL–R8 container, or AL–R8 SI, comply with the latest pit storage specifications to provide an improved storage environment for the pits and would be considered feasible solutions to long-term pit storage at Pantex. The containers were further analyzed with respect to the parameters established in the *Pantex Sitewide EIS* for public, personnel, and environmental impact potential. Based on conclusions drawn from this analysis, DOE concluded that the use of the AL–R8 SI containers does not constitute new circumstances or information or substantial change in the proposed action relevant to environmental concerns; therefore, no supplemental EIS, no new EIS, nor further NEPA documentation is required.

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[1] In the analysis presented in the *Pantex Sitewide EIS* (DOE 1996c), pits are assumed to be repackaged in AT–400A containers. The amount of effort involved in repackaging a pit in an AT–400A container is more intense than the effort needed to repackage a pit in an FL-type container or equivalent; therefore, the doses would be expected to be higher. Since the *Pantex Sitewide EIS* was completed, it has been decided that surplus pits would not be repackaged in AT–400A containers. As a result, the dose estimates associated with repackaging pits as presented in the *Pantex Sitewide EIS* are conservatively high for the SPD EIS. No effort has been made to reestimate the dose associated with repackaging pits. The doses presented in the SPD EIS are based on using the AT–400A container, and therefore represent upper bounds on the expected dose to involved workers.

[2] Extremity doses are estimated to be approximately nine times higher than the whole body dose, but would be expected to stay within DOE's administrative limit of 2 rem/yr, or in the case at Pantex, 5 rem/yr (Low 1999).

Alternatives 2 through 12 involve immobilization of nonpit plutonium at Hanford (Alternative 2, 4, 8, 10, or 11) or SRS (Alternative 3, 5, 6, 7, 9, or 12). This material would be transported from its current location at various DOE sites to the chosen immobilization facility. If the immobilization facility uses a ceramic process, uranium oxide would be required. One of the United States Enrichment Corporation's gaseous diffusion plants would fill cylinders with depleted uranium hexafluoride, which would be transported to a commercial facility for conversion to uranium oxide. (For the purpose of this analysis, the gaseous diffusion plant in Portsmouth, Ohio, and the nuclear fuel fabrication facility in Wilmington, North Carolina, were chosen as representative sites for these activities.) The uranium oxide would be transported to the immobilization facility at Hanford or SRS. After the material is immobilized, it is assumed that the additional canisters of high level waste would be shipped to a potential geologic repository consistent with the assumptions made in the *Final Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste* (WM PEIS) (DOE 1997a). Figure L–2 shows the transportation requirements for the proposed immobilization disposition activities.

The production of MOX fuel (Alternatives 2 through 10) requires transporting plutonium dioxide from the pit conversion facility to the MOX facility at Hanford, INEEL, Pantex, or SRS. However, in every alternative except Alternatives 4 and 5, the pit conversion facility and MOX facility are collocated so there would not be any intersite transportation required for the plutonium dioxide as discussed above. In the case of Alternative 4, the pit conversion facility would be located at Pantex and the plutonium dioxide would be shipped to Hanford. Under Alternative 5, the pit conversion facility would also be at Pantex but the plutonium dioxide would be shipped to SRS. Uranium oxide needed to produce MOX fuel would be converted from uranium hexafluoride, originally from Portsmouth, at Wilmington, and then transported to the MOX facility. If MOX fuel rods are bundled with low-enriched uranium fuel rods, the uranium fuel rods may come from a separate fabrication facility. Transportation of the uranium fuel rods to the MOX facility is equivalent to transportation of uranium fuel to a commercial reactor site. This transportation activity is covered under the *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes* (NRC 1977). The MOX fuel would be transported to a domestic, commercial reactor for power production. For the purposes of this analysis, all MOX fuel was assumed to be transported to North Anna, the commercial reactor farthest from the MOX facility. Because the proposed reactor sites are in the same general area of the country, this approach closely models the risk of implementing each alternative. Figure L–3 shows the transportation requirements for the proposed MOX disposition activities.

Alternatives 2 through 10 include the production of MOX fuel. If this alternative is chosen by DOE, lead assembly fabrication and irradiation may precede the actual production of MOX fuel. Plutonium dioxide at LANL would be shipped to one of five DOE facilities (Argonne National Laboratory–West [ANL–W], Hanford, LLNL, LANL, or SRS). Low-enriched uranium (LEU) oxide would be produced from LEU hexafluoride, originally from Portsmouth, at Wilmington, and then transported to the lead assembly fabrication facility. From the fabrication facility, the MOX fuel lead assemblies would be transported overland to the McGuire reactor. After irradiation in the reactor, the MOX spent fuel lead assemblies would be transported to a DOE site (either ANL–W or Oak Ridge National Laboratory) for postirradiation examination. Figure L–4 shows the transportation requirements for the proposed lead assembly activities.

Table L–1 shows the container type, vehicle type, and number of shipments required for each material form. This table can be used along with Figures L–2 through L–4 to determine which shipments and how many shipments are required for each alternative. The container type and vehicle type are based on currently available containers, and current practices, regulations, and DOE Orders. If a MOX production alternative is selected, DOE would have to design and construct a container to transport MOX fuel to the commercial, domestic reactor. The estimated number of shipments is based on the best available information and could change slightly as material is prepared for transportation.

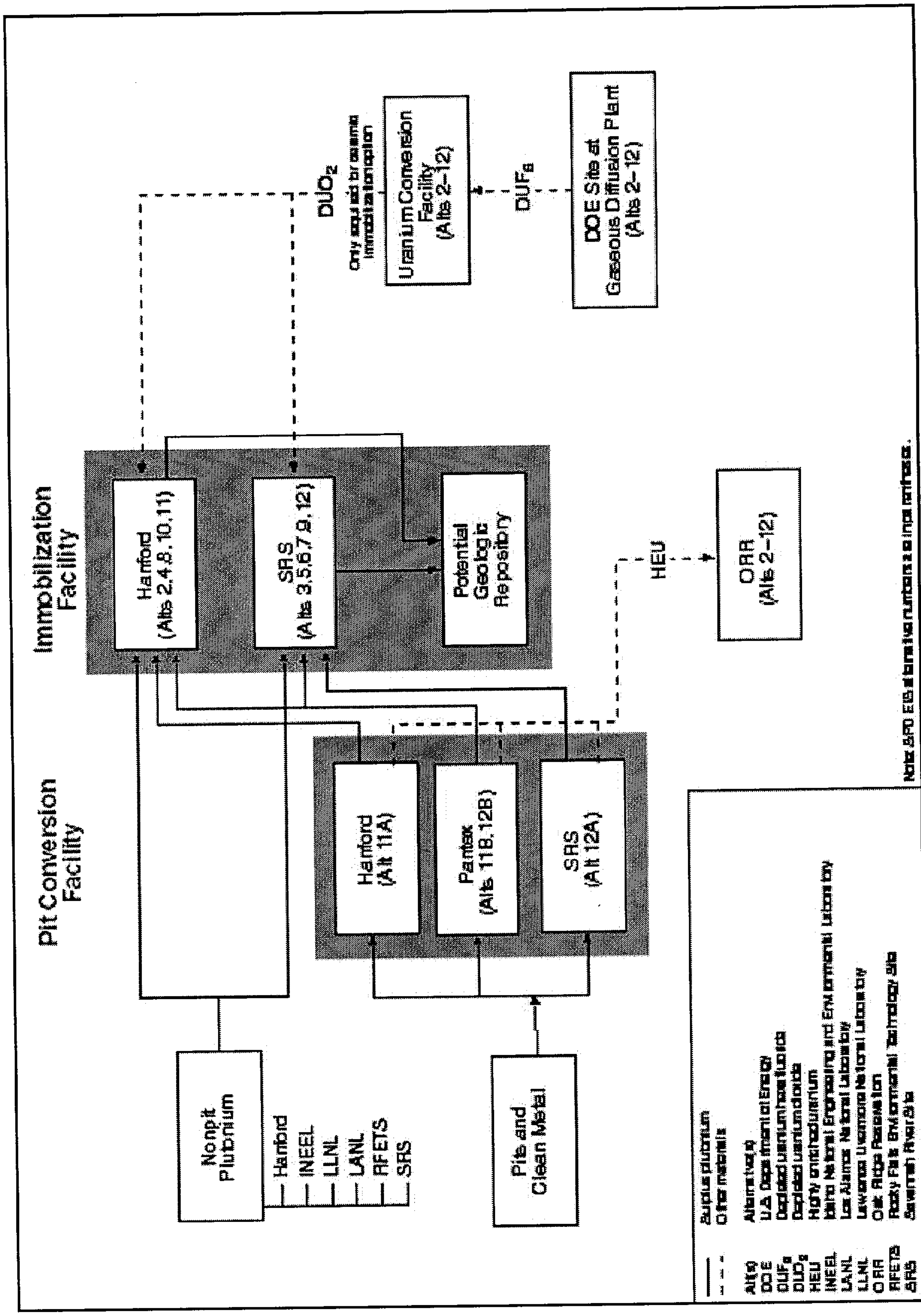

Figure L–2. Transportation Requirements for Plutonium Conversion and Immobilization

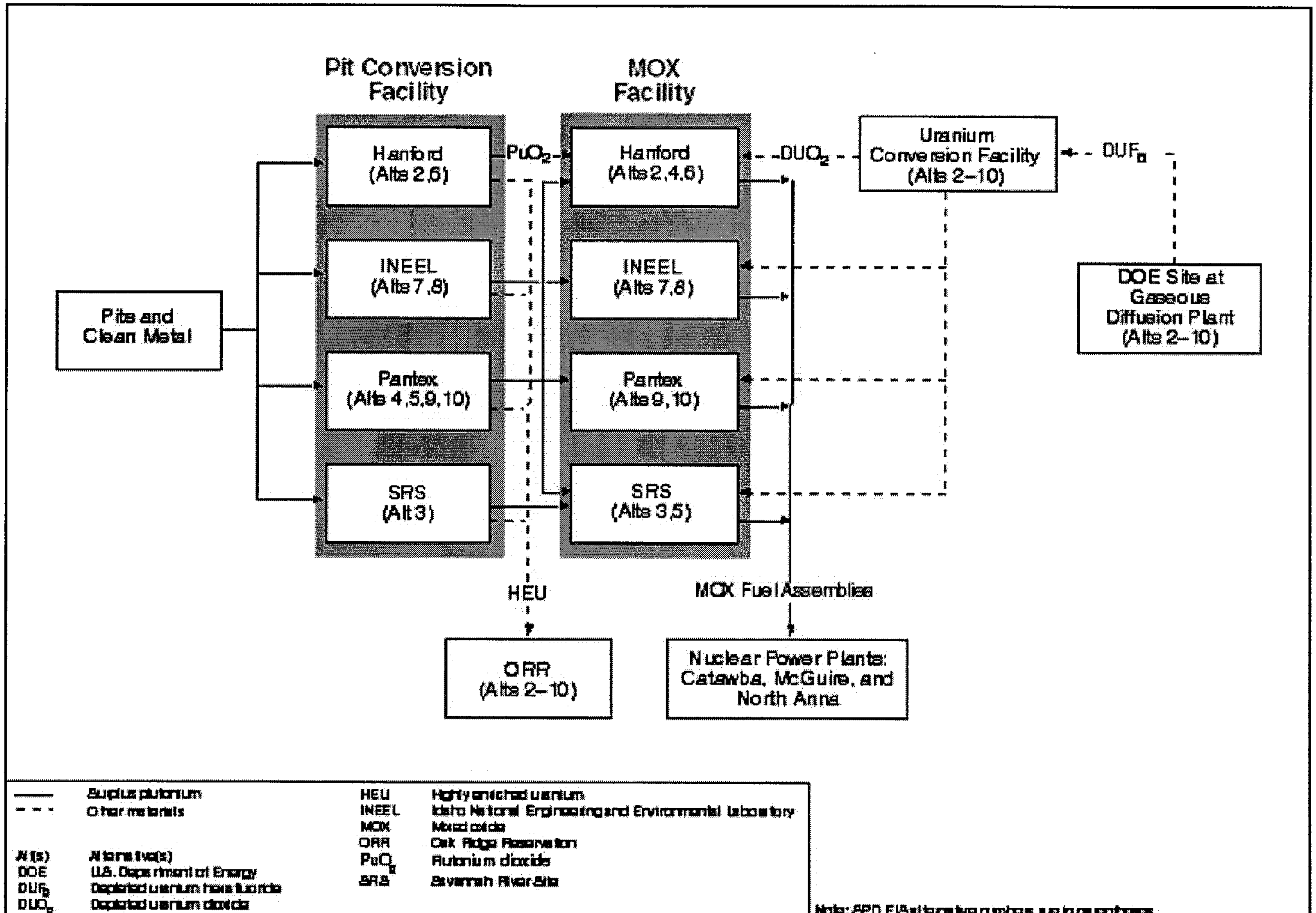

Figure L–3. Transportation Requirements for MOX Fuel Fabrication

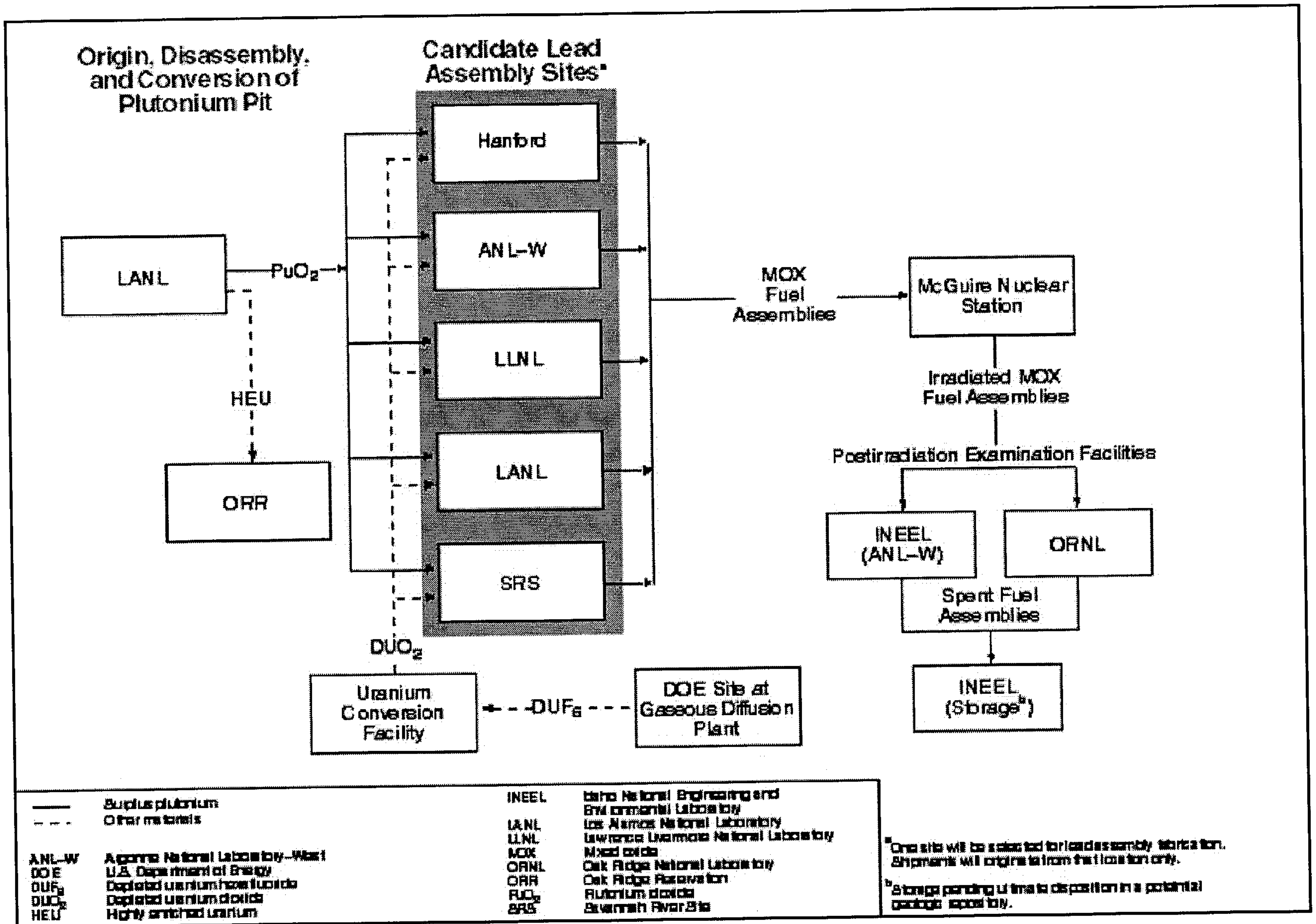

Figure L–4. Transportation Requirements for Lead Assembly Fabrication

Table L–1. Summary of Material Shipments

Origin	Destination	Material Form	Container	Vehicle	No. of Shipments
Surplus plutonium [a,b]					
Pantex	PDCF	Pits	To be designed	SST/SGT	530
Hanford	Immobilization	Oxide	9975	SST/SGT	104
		FFTF pins	M60	SST/SGT	13
		FFTF assemblies	RRSC	SST/SGT	14
ANL–W	Immobilization	ZPPR plates	9975	SST/SGT	116
		ZPPR pins	9975	SST/SGT	40
SRS	Immobilization	SRS material	9975	SST/SGT	48
LANL	Immobilization	Oxide	SAFEKEG	SST/SGT	7
		Metal	SAFEKEG	SST/SGT	4
LLNL	Immobilization	Various	9975	SST/SGT	8
RFETS	Immobilization	Oxide	9975	SST/SGT	104
Pit conversion facility [a,b]					
PDCF	Y–12	HEU	DT-22	SST/SGT	160
PDCF	LANL	Piece parts	UC-609	SST/SGT	20
PDCF	LANL	Piece parts	9968	SST/SGT	10
PDCF	Immobilization or MOX facility	Oxide	SAFEKEG	SST/SGT	254
Immobilization facility					
GDP	UO_2 facility	UF_6 [(c)]	30B cylinder	Commercial	2/2[(d)]
UO_2 facility	Immobilization	UO_2 [(c)]	55-gal drum	Commercial	2/5[(d)]
Immobilization	Potential geologic repository	Vitrified HLW[b]	TRUPACT	Commercial	145/395[(d)]
MOX facility [e]					
GDP	UO_2 facility	UF_6 [(c)]	30B	Commercial	80
UO_2 facility	MOX facility	UO_2 [(c)]	55-gal drum	Commercial	60
MOX facility	Reactors	MOX fuel bundles [a,b]	To be designed	SST/SGT	830
Lead assembly fabrication facility[f]					
LANL	Lead assembly	Pu oxide	SAFEKEG	SST/SGT	12
GDP	UO_2 facility	UF_6	30B cylinder	Commercial	1
UO_2 facility	MOX facility	UO_2	55-gal drum	Commercial	2
MOX facility	Reactors	MOX fuel bundles	MO-1	SST/SGT	4
Reactor	Examination site	Irradiated fuel	Type -B	Commercial	8

[a] From Didlake 1998.
[b] From UC 1998a–h, 1999a–d.
[c] From White 1997.
[d] 17-ton cases/50-ton cases.
[e] Some equipment for the MOX facility may be manufactured in Europe and shipped to the United States. No nuclear or radiologically contaminated materials would be transported. Any such shipments would be made by commercial vessel, and no impacts other than those occurring from routine commercial shipping would be expected.
[f] From O'Connor et al. 1998a–e.

Key: ANL–W, Argonne National Laboratory–W; FFTF, Fast Flux Test Facility; GDP, Gaseous Diffusion Plant; HEU, highly enriched uranium; HLW, high-level waste; LANL, Los Alamos National Laboratory; LLNL, Lawrence Livermore National Laboratory; PDCF, pit disassembly and conversion facility; Pu, plutonium; RFETS, Rocky Flats Environmental Technology Site; SST/SGT, safe, secure trailer/SafeGuards Transport; UF_6, uranium hexafluoride; UO_2, uranium dioxide; ZPPR, Zero Power Physics Reactor.

L.5.2 Representative Routes and Populations

Representative overland truck routes were selected for the origin and destination points identified in Figures L–2, L–3, and L–4 are shown in Table L–2. The routes (which were determined for risk assessment purposes) were

selected consistent with current routing practices and all applicable routing regulations and guidelines. They do not necessarily represent the actual routes that would be used to transport plutonium and other hazardous materials in the future. Details about a planned shipment cannot be identified in advance, as explained in Appendix L.3.3.

Route characteristics that are important to the radiological risk assessment include the total shipment distance and the population distribution along the route. The specific route selected determines both the total potentially exposed population and the expected frequency of transportation-related accidents. Route characteristics are summarized in Table L–2. The population densities along each route are derived from 1990 U.S. Bureau of the Census data and projected forward to the year 2010 using State-specific projections. Rural, suburban, and urban areas are characterized according to the following breakdown: rural population densities range from 0 to 54 persons per square kilometer (0 to 139 person per square mile); the suburban range is from 55 to 1,284 persons per square kilometer (140 to 3,326 persons per square mile); and the urban includes all population densities greater than 1,284 persons per square kilometer (3,326 persons per square mile). The exposed population includes all persons living within 800 m (0.5 mi) of each side of the road.

L.5.3 Distance Traveled by Alternative

Table L–3 shows the number of shipments, the total mileage traveled by the trucks carrying nuclear materials, and the affected populations. The affected population is designed to show the number of people potentially exposed to nuclear material shipments. The measure is calculated by multiplying the number of shipments by the number of people living within 800 m (0.5 mi) of the route used to transport the material. The highest possible lead test assembly mileages and populations from Table L–3 are used in the alternative totals. The number of trips in Table L–3 comes from the SPD EIS data reports (UC 1998a–h, 1999a–d).

[Text deleted.]

L.5.4 Shipment External Dose Rates

The dose and corresponding risk to populations and maximally exposed individuals during incident-free transportation conditions are directly proportional to the assumed shipment external dose rate. The Federal regulations for maximum allowable dose rates for exclusive-use shipments were presented in Appendix L.3.1.

The actual shipment dose rate is a complex function of the composition and configuration of shielding and containment used in the cask, the geometry of the loaded shipments, and characteristics of the material shipped. DOE has years of experience handling the materials that would be required to be shipped under the alternatives assessed in the SPD EIS, and has regularly conducted radiation level measurements while handling these materials. The maximum predicted dose from individual packages, based on experience at DOE facilities, would yield a dose rate less than the Federal regulatory limit in every case. Spent nuclear fuel and nonpit plutonium were conservatively assumed to have dose rates equal to the regulatory limit of 10 mrem/hr at 2 m (6.6 ft) from the vehicle. This DOE experience was used in the preparation of the dose rates given in the data reports (UC 1998a–h, 1999a–d) and used in the analysis.

Table L–2. Potential Shipping Legs Evaluated in the SPD EIS

From	To	Distance (km)	Percentage in Zones: Rural	Suburban	Urban	Population Density (person/km^2): Rural	Suburban	Urban	Affected Population
ANL–W	INEEL	34	100	0	0	2	0	0	84
ANL–W	Hanford	1,035	91.7	7.6	0.6	9	570	2,883	113,482
ANL–W	Pantex	2,395	90.1	8.3	1.6	6	561	2,963	380,038
ANL–W	SRS	3,756	82.8	15.4	1.8	9	453	2,787	767,529
Hanford	INEEL	967	91.6	7.9	0.6	8	559	2,898	107,214
Hanford	ORR	3,981	87.6	11.1	1.3	8	461	2,830	604,916
Hanford	Pantex	3,032	90.6	8.0	1.4	6	574	2,979	450,511
Hanford	Onsite	24	100	0	0	10	0	0	538
Hanford	Geologic repository[a]	1,907	87.8	10.3	1.9	4	485	2,098	397,534
Hanford	LANL	2,511	90.2	8.6	1.2	6	569	2,952	361,442
INEEL	SRS	3,719	82.7	15.4	1.8	9	450	2,788	757,940
INEEL	ORR	3,312	86.7	11.9	1.4	8	437	2,778	518,875
INEEL	LANL	1,841	89.6	9.1	1.4	6	553	2,962	286,387
LANL	Pantex	647	90.7	6.8	2.5	6	676	3,061	132,446
LANL	LLNL	1,218	88.8	7.8	3.4	5	634	3,634	346,679
LANL	INEEL	1,841	89.6	9.1	1.4	6	553	2,962	286,387
LANL	Hanford	2,511	90.2	8.6	1.2	6	569	2,952	361,442
LANL	SRS	2,787	80.8	16.9	2.4	12	455	2,786	684,441
LANL	ORR	2,390	85.8	12.3	1.9	10	435	2,764	439,696
LANL	ANL–W	1,873	89.1	9.5	1.4	4.5	386	2,085	296,222
LLNL	Hanford	1,429	76.0	20.5	3.5	12	487	2,868	478,115
LLNL	INEEL	1,566	85.7	10.3	4.0	6	713	3,546	552,834
LLNL	Pantex	2,327	89.8	6.7	3.5	5	674	3,525	643,591
LLNL	SRS	4,416	80.6	16.4	3.0	10	482	3,165	1,284,987
LLNL	NTS	1,143	85.8	8.6	5.6	5	716	3,771	506,575
Pantex	ORR	1,762	84.4	14.0	1.6	12	392	2,657	302,418
Pantex	SRS	2,169	78.1	19.6	2.3	14	426	2,706	543,092
Pantex	INEEL	2,363	90.2	8.2	1.6	6	561	2,988	373,420
Pantex	WIPP	713	93.1	6.0	0.8	4	697	2,631	75,392
Pantex	NTS	1,997	94.0	4.8	1.2	4	634	3,086	228,159
Pantex	LANL	647	90.7	6.8	2.5	6	676	3,061	132,446
Portsmouth, OH	Fuel fabrication[b]	1,014	63.5	34.6	1.7	20	380	2,446	301,445
RFETS	INEEL	1,178	91.4	7.4	1.2	6	505	3,329	156,394
RFETS	Pantex	1,255	87.2	10.0	2.9	5	634	3,143	319,338
RFETS	Hanford	1,848	91.6	7.4	1.0	6	547	3,228	232,380
RFETS	SRS	2,609	78.1	19.3	2.5	11	439	2,741	674,965
SRS	ORR	575	68.7	30.5	0.8	18	374	2,306	132,959
SRS	Hanford	4,389	84.2	14.2	1.6	9	467	2,823	835,727
SRS	Onsite	6	100	0	0	10	0	0	134
SRS	Geologic repository[a]	3,936	83.2	19.9	1.9	9	510	3,069	893,080
SRS	LANL	2,787	80.8	16.9	2.4	12	455	2,786	684,441
Fuel fabrication[b]	SRS	581	72.8	26.8	0.3	23	301	2,202	97,034
Fuel fabrication[b]	Pantex	2,577	76.2	22.4	1.4	14	392	2,690	651,769
Fuel fabrication[b]	Hanford	4,796	82.6	16.1	1.2	10	435	2,806	856,223

Table L–2. Potential Shipping Legs Evaluated in the SPD EIS (Continued)

From	To	Distance (km)	Percentage in Zones			Population Density (person/km²)			Affected Population
			Rural	Suburban	Urban	Rural	Suburban	Urban	
Fuel fabrication[b]	ANL–W	4,165	81.0	17.7	1.3	10	418	2,769	787,474
Fuel fabrication[b]	LLNL	4,880	82.5	15.1	2.4	10	457	3,192	1,199,169
Fuel fabrication[b]	LANL	3,201	78.0	19.8	1.6	13	413	2,766	696,023
Generic 4,000 km		4,000	84.0	15.0	1.0	6	719	3,861	969,600
Generic 5,000 km		5,000	84.0	15.0	1.0	6	719	3,861	1,212,000
Hanford	Catawba	4,498	84.5	14.1	1.3	9	447	2,776	765,850
INEEL/ANL	Catawba	3,793	83.0	15.5	1.5	9	429	2,737	697,959
SRS	Catawba	251	69.0	29.8	1.2	17	418	2,373	66,154
LANL	Catawba	2,844	81.1	17.0	1.8	11	428	2,722	595,856
LLNL	Catawba	4,539	84.3	13.1	2.6	9	477	3,167	1,105,526
Pantex	Catawba	2,243	78.6	19.7	1.7	13	397	2,626	477,319
Catawba	ORR	497	58.3	39.8	2.0	20	405	2,546	177,922
Hanford	McGuire	4,458	84.8	13.9	1.2	9	428	2,802	716,024
INEEL/ANL–W	McGuire	3,753	83.4	15.3	1.3	9	409	2,767	636,712
SRS	McGuire	296	66.4	31.6	2.1	15	441	2,438	94,828
LANL	McGuire	2,821	81.5	16.9	1.7	11	401	2,753	559,307
LLNL	McGuire	4,500	84.6	12.9	2.5	9	458	3,207	1,055,765
Pantex	McGuire	2,203	79.3	19.3	1.4	13	370	2,661	419,295
McGuire	ORR	457	59.5	39.9	0.5	21	343	2,504	118,268
Hanford	N. Anna	4,575	86.1	12.4	1.4	9	449	2,717	744,228
INEEL/ANL–W	N. Anna	3,870	85.0	13.4	1.6	10	429	2,666	671,048
SRS	N. Anna	837	72.7	26.8	0.5	21	306	2,167	145,069
LANL	N. Anna	3,117	83.6	14.7	1.7	13	397	2,711	574,877
LLNL	N. Anna	4,797	84.7	12.7	2.7	9	492	2,886	1,134,405
Pantex	N. Anna	2,499	82.0	16.6	1.4	14	364	2,619	435,744
N. Anna	ORR	753	76.3	22.7	1.0	22	317	2,503	137,224

[a] Potential geologic repository assumed to be located at Yucca Mountain, Nevada, for the purposes of analysis.
[b] Assumed to be located at Wilmington, North Carolina, for the purposes of analysis.

Key: ANL–W, Argonne National Laboratory–W; LANL, Los Alamos National Laboratory; LLNL, Lawrence Livermore National Laboratory; NTS, Nevada Test Site; ORR, Oak Ridge Reservation; RFETS, Rocky Flats Environmental Technology Site; WIPP, Waste Isolation Pilot Plant.

Table L–3. Summary of SPD EIS Transportation Requirements

Alternative	Number of Trips	Cumulative Distance (km)	Affected Population (millions)
2	2,447	7.5 M	5.4
3	2,530	4.3 M	7.0
4	2,171	6.3 M	4.9
5	2,254	3.8 M	6.7
6	2,530	8.7 M	8.5
7	2,530	7.6 M	8.1
8	2,447	6.4 M	5.3
9	2,000	4.8 M	6.4
10	1,917	3.6 M	4.2
11A	2,153	3.7 M	4.7
11B	1,877	2.5 M	4.1
12A	2,236	4.4 M	6.8
12B	1,960	3.9 M	6.4
Lead assembly			
ANL–W	27	77 K	2.5
Hanford	27	89 K	2.7
LLNL	27	73 K	3.4
LANL	15	49 K	2.1
SRS	27	67 K	1.7

Key: ANL–W, Argonne National Laboratory–W; K, thousands; LANL, Los Alamos National Laboratory; LLNL, Lawrence Livermore National Laboratory; M, million.

L.5.5 Health Risk Conversion Factors

The health risk conversion factors used to estimate expected cancer fatalities were taken from the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991): 0.0005 and 0.0004 fatal cancer cases per person-rem for members of the public and workers, respectively. Cancer fatalities occur during the lifetimes of the exposed populations and, thus, are called LCFs.

L.5.6 Accident Involvement Rates

For the calculation of accident risks, vehicle accident and fatality rates are taken from data provided in other reports (Saricks and Kvitek 1994). Accident rates are generically defined as the number of accident involvements (or fatalities) in a given year per unit of travel in that same year. Therefore, the rate is a fractional value, with the accident-involvement count as the numerator of the fraction and vehicular activity (total travel distance) as its denominator. Accident rates are generally determined for a multiyear period. For assessment purposes, the total number of expected accidents or fatalities is calculated by multiplying the total shipment distance for a specific case by the appropriate accident or fatality rate.

For truck transportation, the rates presented are specifically for heavy combination trucks involved in interstate commerce (Saricks and Kvitek 1994). Heavy combination trucks are rigs composed of a separable tractor unit containing the engine and one to three freight trailers connected to each other. Heavy combination trucks are typically used for radioactive waste shipments. The truck accident rates are computed for each State based on statistics compiled by the DOT Office of Motor Carriers for 1986 to 1988. Saricks and Kvitek present accident involvement and fatality counts; estimated kilometers of travel by State; and the corresponding average accident involvement, fatality, and injury rates for the 3 years investigated. Fatalities are deaths (including crew members)

attributable to the accident or that occurred at any time within 30 days thereafter. SST/SGT accident rates are based on operational experience (Claus and Shyr 1999) and influence factors (Phillips et al. 1994).

L.5.7 Container Accident Response Characteristics and Release Fractions

The transportation accident model assigns accident probabilities to a set of accident categories. Eight accident-severity categories defined in the NRC's *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes,* NUREG-0170 (NRC 1977), were used. The least severe categories (Categories I and II) represent low magnitudes of crush force, accident-impact velocity, fire duration, and puncture-impact speed. The most severe category (Category VIII) represents a large crush force, high accident-impact velocity, long fire duration, and a high puncture-impact speed. The fraction of material released and material aerosolized, and the fraction of that material that is respirable (particles smaller than 10 microns), was assigned based on the accident categories and container types. Because all plutonium shipments will use the previously described Type B containers and the SST/SGT system, even severe accidents release, at the most, a portion of the material being transported. The risks associated with other materials are significantly lower.

L.6 RISK ANALYSIS RESULTS

L.6.1 Per-Shipment Risk Factors

Per-shipment risk factors have been calculated for the collective populations of exposed persons and the crew for all anticipated routes and shipment configurations. The radiological risks are presented in doses per shipment for each unique route, material, and container combination. Doses are calculated for the crew, off-link public (i.e., people living along the route), on-link public (i.e., pedestrians and drivers along the route), and public at rest and fueling stops (i.e., stopped cars, buses, and trucks, workers, and other bystanders). The accident risk factors are called "dose risk" because the values incorporate the spectrum of accident severity probabilities and associated consequences. Separate risk factors are provided for fatalities resulting from hydrocarbon emissions (known to contain carcinogens) and transportation accidents (fatalities resulting from impact).

L.6.2 Evaluation of Shipment Risks

Tables L–4 and L–5 show the human health risks and maximum human health risks, respectively, of transporting materials for the lead assembly alternatives. As shown, the risks include the risk of transporting uranium dioxide, uranium hexafluoride, plutonium dioxide, fuel assemblies, and spent fuel. Table L–6 shows the results of similar calculations that give the risks for each alternative. The risk estimates in Table L–6 include the maximum risk for the lead assembly transportation (Alternatives 2 through 10), plutonium pit shipments, pit material shipments (HEU and nonplutonium bearing pit parts), uranium hexafluoride, uranium dioxide, fuel assemblies, and nonpit plutonium. The risks are calculated by multiplying the per-shipment factors by the number of shipments and, in the case of the radiological doses, by the health risk conversion factors.

Table L–4. Human Health Risks of Transport to Lead Assembly Facilities

	DUO_2 and LEU Fuel Assemblies From FFF					PuO_2 From LANL				
	Routine Transport Impacts					Routine Transport Impacts				
	Radiological			Accident Risks		Radiological			Accident Risks	
Site	Crew	Public	Nonrad[a]	Rad	Nonrad	Crew	Public	Nonrad[a]	Rad	Nonrad
LANL	5.6E-6	4.5E-5	2.0E-5	3.8E-4	2.5E-4	–	–	–	–	–
ANL–W	7.3E-6	5.8E-5	2.2E-5	1.6E-4	3.2E-4	2.1E-6	2.2E-6	8.2E-5	2.3E-4	1.6E-4
SRS	9.8E-7	7.9E-6	1.3E-6	1.2E-5	4.3E-5	3.2E-6	4.2E-6	2.1E-4	5.3E-4	2.3E-4
Hanford	8.4E-6	6.7E-5	2.3E-5	1.7E-4	3.7E-4	2.8E-6	2.9E-6	9.4E-5	2.8E-4	2.1E-4
LLNL	8.5E-6	6.8E-5	4.7E-5	3.4E-4	3.8E-4	1.4E-6	1.4E-6	1.3E-4	2.9E-4	1.0E-4

[a] Toxic emissions.

Key: ANL–W, Argonne National Laboratory–West; DUO_2, depleted uranium dioxide; FFF, Uranium Fuel Fabrication Facility; LANL, Los Alamos National Laboratory; LEU, low-enriched uranium; LLNL, Lawrence Livermore National Laboratory; Rad, radiological; Nonrad, nonradiological; PuO_2, plutonium dioxide; UO_2, uranium dioxide.

Note: All risks are expressed in latent cancer fatalities during the implementation of the proposed action, except for the Nonrad Accident Risks column, which is the number of fatalities.

Table L–5. Maximum Human Health Risks of Transport to Lead Assembly Facilities

	Routine Transport Impacts			Accident Risks	
	Radiological				
Shipment	Crew	Public	Nonradiological[a]	Radiological	Nonradiological
Depleted UO_2 and LEU fuel assemblies from FFF and PuO_2 from LANL	1.1E-5	7.0E-5	2.1E-4	6.3E-4	5.8E-4
Depleted UF_6 from gaseous diffusion plant to FFF	2.5E-8	2.0E-7	3.4E-6	5.2E-5	4.0E-5
Lead assemblies to reactor site	3.7E-7	2.2E-7	1.2E-4	2.1E-6	1.3E-4
Spent fuel to postirradiation examination site	5.5E-4	4.8E-3	7.8E-5	2.3E-3	1.2E-3

[a] Toxic emissions.

Key: FFF, Uranium Fuel Fabrication Facility; LANL, Los Alamos National Laboratory; LEU, low-enriched uranium; PuO_2, plutonium dioxide; UF_6, uranium hexafluoride; UO_2, uranium dioxide.

Note: All risks are expressed in latent cancer fatalities during the implementation of the proposed action, except for the Nonradiological Accident Risks column, which is the number of fatalities.

L.6.3 Maximally Exposed Individuals

The risks to maximally exposed individuals under incident-free transportation conditions were estimated for hypothetical exposure scenarios. The estimated dose to inspectors and the public is presented in Table L–7 on a per-event basis (person-rem per event). Note that the potential exists for individual exposures if multiple exposure events occur. For instance, the dose to a person stuck in traffic next to a shipment for 30 minutes is calculated to be 11 mrem. (This conservatively assumes the person in a car is 1.2 m [4 ft] from the edge of the truck.) If the exposure duration was longer, the dose would rise proportionally. In addition, a person working at a truck service station could receive a significant dose if trucks were to use the same stops repeatedly. The dose to a person fueling a truck could be as much as 1 mrem. Administrative controls could be instituted to control the location and duration of truck stops if multiple exposures were to occur routinely. However, it is DOE's normal practice to have SST/SGT guard force members (trained, monitored radiation workers) perform fueling and routine on-road maintenance checks (i.e., check oil or windshield wiper fluid).

Table L–6. Total Risks for All SPD EIS Alternatives

				Routine Transport Impacts				Accident Risks
				Radiological		Nonradiological		Radiological
Alternative	**Pit Conversion**	**MOX**	**Immobilization**	**Crew**	**Public**	**Emission**	**Traffic**	**Accident**
2	Hanford	Hanford	Hanford	0.012	0.020	0.025	0.074	0.004
3	SRS	SRS	SRS	0.024	0.034	0.019	0.053	0.004
4	Pantex	Hanford	Hanford	0.012	0.020	0.021	0.065	0.004
5	Pantex	SRS	SRS	0.024	0.033	0.016	0.050	0.004
6	Hanford	Hanford	SRS	0.024	0.035	0.033	0.091	0.004
7	INEEL	INEEL	SRS	0.024	0.035	0.032	0.083	0.004
8	INEEL	INEEL	Hanford	0.012	0.020	0.024	0.065	0.003
9	Pantex	Pantex	SRS	0.024	0.034	0.019	0.052	0.004
10	Pantex	Pantex	Hanford	0.012	0.019	0.012	0.043	0.003
11A	Hanford	NA	Hanford	0.027	0.036	0.011	0.054	0.0003
11B	Pantex	NA	Hanford	0.027	0.036	0.007	0.045	0.0007
12A	SRS	NA	SRS	0.057	0.074	0.021	0.081	0.0006
12B	Pantex	NA	SRS	0.057	0.073	0.018	0.078	0.0012

Key: NA, not applicable.

Note: All risks are expressed in latent cancer fatalities during the implementation of the proposed action, except for the Nonradiological Accident Risks column, which is the number of fatalities.

Table L–7. Estimated Dose to Maximally Exposed Individuals During Incident-Free Transportation Conditions[a,b]

Receptor	Dose to Maximally Exposed Individual
Workers	
Crew member	0.1 rem/yr[c]
Inspector	0.0029 rem/event
Public	
Resident	4.0×10^{-7} rem/event
Person in traffic construction	0.011 rem/event
Person at service station	0.001 rem/event

[a] The exposure scenario assumptions are described in Appendix L.6.3.

[b] Doses are calculated assuming that the shipment external dose rate is equal to the maximum expected dose 10 mrem/hr at 2 m (6.6 ft) from the package.

[c] Dose to truck drivers could exceed the legal limit of 100 mrem/yr in the absence of administrative controls.

The cumulative dose to a resident was calculated assuming all shipments passed his or her home. The cumulative doses assume that the resident is present for every shipment and is unshielded at a distance of 30 m (98 ft) from the route. Therefore, the cumulative dose is only a function of the number of shipments passing a particular point and is independent of the actual route being considered. The maximum dose to this resident, would be about 1 mrem. The annual individual dose can be estimated by assuming that shipments would occur uniformly over a 15-year time period.

The accident consequence assessment is intended to provide an estimate of the maximum potential impacts posed by the most severe potential transportation accidents involving a shipment. The accident consequence results are presented in Table L–8 for the maximum severity accidents involving plutonium dioxide shipments,

Table L–8. Estimated Dose to the Population and to Maximally Exposed Individuals During the Most Severe Accident Conditions (Plutonium Dioxide)[a, b]

	Neutral Conditions[c]				Stable Conditions[f]			
	Population[d]		Maximally Exposed Individual[e]		Population[d]		Maximally Exposed Individual[e]	
Mode and Accident Location	Dose (person-rem)	Consequences (Cancer Fatalities)	Dose (rem)	Consequences (Probability of Cancer Fatality)	Dose (person-rem)	Consequences (Cancer Fatalities)	Dose (rem)	Consequences (Probability of Cancer Fatality)
Truck								
Urban	228,760	114	684	0.68	40,420	20.2	23.2	0.023
Suburban	49,880	25	684	0.68	8,815	4.4	23.2	0.023
Rural	624	0.31	684	0.68	581	0.29	23.2	0.023

[a] The most severe accidents correspond to the NUREG-0170 accident severity Category VIII (NRC 1977).
[b] Buoyant plume rise resulting from fire for a severe accident was included in the exposure model.
[c] Neutral weather conditions result in moderate dispersion and dilution of the release plume. Neutral conditions were taken to be Pasquill stability Class D with a wind speed of 4 m/sec (9 mph). Neutral conditions occur approximately 50 percent of the time in the United States.
[d] Populations extend at a uniform density to a radius of 80 km (50 mi) from the accident site. Population exposure pathways include acute inhalation, acute cloudshine, groundshine, resuspended inhalation, resuspended cloudshine, and ingestion of food, including initially contaminated food (RISKIND assumes that all food is grown in rural areas) (Yuan et al. 1995). It is assumed that decontamination or mitigative actions are taken.
[e] The maximally exposed individual is assumed to be at the location of maximum exposure. The locations of maximum exposure would be 100 m (330 ft) and 500 m (1,650 ft) from the accident site under neutral and stable atmospheric conditions, respectively. Individual exposure pathways include acute inhalation, acute cloudshine, and groundshine during passage of the plume. No ingested dose is considered. Note that the maximally exposed individual receives more dose than the population in a rural location. This analytic phenomena is caused by probabilistic calculations. It is very unlikely that an individual will be nearby in a rural population zone.
[f] Stable weather conditions result in minimal dispersion and dilution of the release plume and are thus unfavorable. Stable conditions were taken to be Pasquill stability Class F with a wind speed of 1 m/sec (2.2 mph). Stable conditions occur approximately one-third of the time in the United States.

and Table L–9 for maximum severity accidents involving plutonium pits. Table L–8 applies to alternatives in which the pit conversion facility is located at Pantex, and large amounts of plutonium dioxides are shipped to a MOX or conversion facility. Table L–9 applies to alternatives in which plutonium pits and metals are shipped to a pit conversion facility at a site other than Pantex. In either table, the accident frequency in rural locations is about 1×10^{-7}per year (once in 10 million years). The frequency of accidents in urban and suburban zones was evaluated. Accidents are much less likely to occur in urban and suburban zones because the total distance traveled is much lower than in rural zones. The impacts represent the most severe accidents hypothesized.

The hypothetical accidents described in Tables L–8 and L–9 involve either a long-term fire or tremendous impact or crushing forces. In the case of crushing forces, a fire would have to be burning in order to spread the plutonium as modeled. These accidents are assumed to cause a ground-level release of 10 percent of the radioactive material in the truck. These accidents are more likely on rural interstates where speeds are higher and where the vehicles spend most of their travel time. NUREG-0170 (NRC 1977) describes the analytic approach in more detail.

The population doses are for a uniform population density within an 80-km (50-mi) radius (Neuhauser and Kanipe 1995). The location of the maximally exposed individual is determined based on atmospheric conditions

at the time of the accident and the buoyant characteristics of the released plume. The locations of maximum exposure would be 100 m (330 ft) and 500 m (1,650 ft) from the accident site for neutral (average)

Table L–9. Estimated Dose to the Population and to Maximally Exposed Individuals During the Most Severe Accident Conditions (Plutonium Pits)[a, b]

	Neutral Conditions [c]				Stable Conditions [f]			
	Population [d]		Maximally Exposed Individual [e]		Population [d]		Maximally Exposed Individual [e]	
Mode and Accident Location	Dose (person-rem)	Consequences (Cancer Fatalities)	Dose (rem)	Consequences (Probability of Cancer Fatality)	Dose (person-rem)	Consequences (Cancer Fatalities)	Dose (rem)	Consequences (Probability of Cancer Fatality)
Truck								
Urban	31,920	16	96	0.096	5,640	2.8	3.3	0.0016
Suburban	6,960	3.5	96	0.096	1,230	0.62	3.3	0.0016
Rural	87	0.044	96	0.096	81	0.041	3.3	0.0016

[a] The most severe accidents correspond to the NUREG-0170 accident severity Category VIII (NRC 1977).

[b] Buoyant plume rise resulting from fire for a severe accident was included in the exposure model.

[c] Neutral weather conditions result in moderate dispersion and dilution of the release plume. Neutral conditions were taken to be Pasquill stability Class D with a wind speed of 4 m/sec (9 mph). Neutral conditions occur approximately 50 percent of the time in the United States.

[d] Populations extend at a uniform density to a radius of 80 km (50 mi) from the accident site. Population exposure pathways include acute inhalation, acute cloudshine, groundshine, resuspended inhalation, resuspended cloudshine, and ingestion of food, including initially contaminated food (RISKIND assumes that all food is grown in rural areas) (Yuan et al. 1995). It is assumed that decontamination or mitigative actions are taken.

[e] The maximally exposed individual is assumed to be at the location of maximum exposure. The locations of maximum exposure would be 100 m (330 ft) and 500 m (1,650 ft) from the accident site under neutral and stable atmospheric conditions, respectively. Individual exposure pathways include acute inhalation, acute cloudshine, and groundshine during passage of the plume. No ingested dose is considered. Note that the maximally exposed individual receives more dose than the population in a rural location. This analytic phenomena is caused by probabilistic calculations. It is very unlikely that an individual will be nearby in a rural population zone.

[f] Stable weather conditions result in minimal dispersion and dilution of the release plume and are thus unfavorable. Stable conditions were taken to be Pasquill stability Class F with a wind speed of 1 m/sec (2.2 mph). Stable conditions occur approximately one-third of the time in the United States.

and stable conditions, respectively. The dose to the maximally exposed individual is independent of the location of the accident. No acute or early fatalities would be expected from radiological causes.

L.6.4 Waste Transportation

Under all of the alternatives being considered in the SPD EIS, some transportation would be required to support routine shipments of wastes from the proposed surplus plutonium disposition facilities to treatment, storage, or disposal facilities located on the sites. All DOE sites have plans and procedures for handling and transporting waste. This transportation would be handled in the same manner as other site waste shipments and would not represent a large increase in the amount of wastes generated at these sites. The shipments would not represent any additional risks beyond the ordinary waste shipments at these sites, as analyzed in the WM PEIS (DOE 1997a).

However, in four specific cases, waste would be generated that is not covered in the WM PEIS (DOE 1997a): (1) transuranic (TRU) waste generated at Pantex from the pit conversion facility; (2) low-level waste (LLW) generated at Pantex from the pit conversion facility; (3) LLW generated at Pantex from the MOX facility, and (4) LLW generated at LLNL during lead assembly fabrication.

TRU waste generated at Pantex was not covered by the WM PEIS Record of Decision (ROD) because there was no TRU waste at Pantex at the time the ROD was issued, and none was anticipated to be generated by ongoing

site operations. Location of the pit conversion and MOX facilities at Pantex would result in the generation of TRU waste as described in Section 4.17.2.2 of the SPD EIS. Shipment of TRU waste to WIPP was analyzed using the methodology and parameters found in Appendix E of the Waste Isolation Pilot Plant *Disposal Phase Final Supplemental Environmental Impact Statement* (DOE 1997b). In order to support the transportation of TRU waste from Pantex to WIPP, 76 additional shipments were analyzed in the SPD EIS.

A fairly large increase in the amount of LLW (i.e., 25 percent of the site's current storage capacity) would be expected if the pit conversion facility were located at Pantex. Currently, this type of waste is shipped to the Nevada Test Site (NTS) for disposal. In order to support the transportation of pit conversion facility LLW from Pantex to NTS, 21 additional shipments were analyzed in the SPD EIS. The impacts were calculated from LLW transportation impacts presented in the WM PEIS (DOE 1997a).

An additional increase in the amount of LLW (i.e., 14 percent, for a total of 39 percent of the site's current storage capacity) would be expected if the pit conversion and MOX facilities are located at Pantex. Currently, this type of waste is shipped to NTS for disposal. In order to support the transportation of MOX LLW from Pantex to NTS, 38 additional shipments have been analyzed in the SPD EIS. The impacts were calculated from LLW transportation impacts presented in the WM PEIS (DOE 1997a).

Further, an increase in the LLW at LLNL would be expected if the lead assembly were done at LLNL. Currently, this type of waste is shipped to NTS for disposal. In order to support transportation of lead assembly LLW from LLNL to NTS, 44 additional shipments were analyzed in the SPD EIS. The impacts were calculated from LLW transportation impacts presented in the WM PEIS (DOE 1997a). Table L–10 shows the impacts of transporting LLW and TRU waste. The radiological risks to the public are larger for TRU than for LLW because of the larger amount of radioactive material in TRU. The dose to the crew are about the same, because the truck carrying TRU would require some shielding or spacing to ensure that the dose rate to the truck crew is less than 2 mrem/hr.

Table L–10. Impacts of Transporting LLW and Transuranic Waste

					Routine Transport Impacts				Accidental Risks
					Radiological		Nonradiological		Radiological
Waste Type	Origin	Destination	Trips	Kilometers Traveled	Crew	Public	Emission	Traffic	
LLW	Pantex, pit conversion facility	NTS	38	76,000	0.0011	0.0015	0.00018	0.0029	5.8×10^{-7}
LLW	Pantex, MOX	NTS	21	42,000	0.0006	0.0008	0.00010	0.0016	3.2×10^{-7}
LLW	LLNL	NTS	44	50,000	0.0007	0.0010	0.00056	0.0020	3.8×10^{-7}
TRU	Pantex, pit conversion facility	WIPP	76	54,000	0.0008	0.0025	0.00013	0.0015	1.1×10^{-6}

Key: LLNL, Lawrence Livermore National Laboratory; LLW, low-level waste; NTS, Nevada Test Site; TRU, transuranic; WIPP, Waste Isolation Pilot Plant.

Note: All risks are expressed in latent cancer fatalities during the implementation of the proposed actions except for the Nonradiological Accidental Traffic column, which is the number of fatalities.

L.6.5 Consequences of Sabotage or Terrorist Attack During Transportation

This section provides an evaluation of impacts that could potentially result from a malicious act on a shipment of hazardous or radioactive material during transportation. In no instance, even in severe cases such as those discussed below, could a nuclear explosion or permanent contamination of the environment leading to condemnation of land occur. Because of the Transportation Safeguards System described in Appendix L.3.2,

DOE considers sabotage or terrorist attack on an SST/SGT to be unlikely enough such that no further risk analysis is required.

DOE analyzed the nonproliferation aspects (DOE 1997c) of the transportation associated with the alternatives in the SPD EIS. In this study, DOE realized that all plutonium disposition alternatives under consideration would involve processing and transport of plutonium, which will involve more risk of theft in the short term than if the material had remained in heavily guarded storage, in return for the long-term benefit of converting the material to more proliferation-resistant forms. DOE intends to use the same SST/SGTs for these shipments that are used for shipment of intact nuclear weapons, with similar security forces and other measures. The level of assurance against possible attack during transportation can be increased to essentially any desired level by applying more resources such as money, security forces, or technology. DOE concluded that transport of plutonium is the point in the disposition process when the material is most vulnerable to overt, armed attacks designed to steal plutonium. With sufficient resources devoted to security, high levels of protection against such overt attacks can be provided. International, and particularly overseas, shipments would involve greater transportation concerns than domestic shipments (DOE 1997c).

The *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (DOE 1996d) analyzed the spectrum of attacks on spent nuclear fuel casks. They fall into three categories or scenarios: (1) exploding a bomb near a shipping cask, (2) attacking a cask with a shaped charge or an armor-piercing weapon (i.e., an antitank weapon), and (3) hijacking (stealing) a shipping cask. None of the scenarios considered would lead to a criticality accident. DOE determined that, due to the security measures that would be in place for any spent nuclear fuel shipments, such attacks would be unlikely to occur. At a minimum, the extent or effects of any such attacks would be mitigated by the security measures. Additionally, the SPD EIS considered a comparatively few shipments (if the lead assembly program is implemented) of spent nuclear fuel. Other materials, including uranium hexaflouride, uranium dioxide, TRU waste, and LLW, are commonly shipped and do not represent particularly attractive targets for sabotage or terrorist attacks.

L.7 CUMULATIVE IMPACTS OF TRANSPORTATION

L.7.1 Radiological Impacts

The cumulative impacts of the transportation of radioactive material consist of impacts from (a) historical shipments of radioactive waste and spent nuclear fuel, (b) reasonably foreseeable actions that include transportation of radioactive material, (c) general radioactive materials transportation that is not related to a particular action, and (d) the alternatives evaluated in the SPD EIS. The assessment of cumulative transportation impacts concentrates on the cumulative impacts of offsite transportation because offsite transportation yields potential radiation doses to a greater portion of the general population than does onsite transportation. The collective dose to the general population and workers was the measure used to quantify cumulative transportation impacts. This measure of impact was chosen because it may be directly related to LCFs using a cancer risk coefficient and because of the difficulty in identifying a maximally exposed individual for shipments throughout the United States spanning the period 1943 through 2048 (106 years). The year 1943 corresponds to the start of operations at Hanford and the Oak Ridge Reservation.

Collective doses from historical shipments of spent nuclear fuel to NTS were summarized in *Summary of Doses and Health Effects* (Jones and Maheras 1994). Data for these shipments were available for 1971 through 1993 and were linearly extrapolated back to 1951, the start of operations at NTS, because data before 1971 were not available. The results of this analysis are summarized in Table L–11. Collective doses from historical shipments of low-level waste, mixed low-level waste, and TRU waste were also estimated (DOE 1996e). Over the time period 1974 through 1994, there were about 8,400 of these shipments. These

Table L–11. Cumulative Transportation-Related Radiological Collective Doses and Latent Cancer Fatalities (1943 to 2048) (person-rem)

Category	Collective Dose	
	Occupational Dose	General Population Dose
Historical shipments (DOE 1995a)	250	130
Radioactive waste to Nevada Test Site (DOE 1996e)	82	100
Reasonably foreseeable actions		
Nevada Test Site expanded use (DOE 1996e)	–	150[a]
Spent nuclear fuel management (DOE 1995a, 1996d)	360	810
Waste Management PEIS (DOE 1997a)[b]	16,000	20,000
Waste Isolation Pilot Plant (DOE 1997b)	790	5,900
Molybdenum-99 production (DOE 1996f)	240	520
Tritium supply and recycling (DOE 1995b)	–	–
Surplus highly enriched uranium disposition (DOE 1996g)	400	520
Storage and Disposition PEIS (DOE 1996a)	–	2,400[a]
Stockpile Stewardship (DOE 1996h)	–	38[a]
Pantex (DOE 1996c)	250[c]	490[c]
West Valley (DOE 1996i)	1,400	12,000
S3G and D1G prototype reactor plant disposal (DOE 1997d)	2.9–6.8	2.2–5.4
S1C prototype reactor plant disposal (DOE 1996j)	6.7	1.9
Container system for naval spent nuclear fuel (USN 1996a)	11	15
Cruiser and submarine reactor plant disposal (USN 1996b)	5.8	5.8
Submarine reactor compartment disposal (USN 1984)	–	0.053
Return of cesium 137 capsules (DOE 1994)	0.42	5.7
Uranium billets (DOE 1992)	0.50	0.014
Nitric acid (DOE 1995c)	0.43	3.1
General transportation		
1943 to 1982 (NRC 1977)	220,000	170,000
1983 to 2048 (Weiner, LaPlante, and Hageman 1991a:661–666; 1991b:655–660)	110,000	120,000
Shipments for alternatives evaluated in the SPD EIS	10	50
Summary		
Historical	330	230
Reasonably foreseeable actions	19,000	43,000
General transportation (1943 to 2048)	330,000	290,000
Shipments for alternatives evaluated in the SPD EIS	10	50
Total collective dose (rounded to nearest thousand)	349,000	333,000
Total latent cancer fatalities	140	170

[a] Includes public and occupational collective doses.
[b] Includes mixed low-level waste and low-level waste; transuranic waste included in DOE 1997b.
[c] Includes all highly enriched uranium shipped to Y–12.

shipments were estimated to result in a collective occupational dose of 82 person-rem and a collective dose for the general population of 100 person-rem.

Collective doses from other historical shipments of radioactive material were evaluated in the *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (DOE 1995a). These include historical shipments associated with Hanford, INEEL, Oak Ridge, SRS, and Naval spent nuclear fuel and test specimens.

There are considerable uncertainties in these historical estimates of collective dose. For example, the population densities and transportation routes used in the dose assessments were based on census data for 1990 and the U.S. highway and rail system as it existed in the 1990s. Using census data for 1990 tends to overestimate historical collective doses because the U.S. population has continuously increased over the time covered in these assessments. Basing collective dose estimates on the U.S. highway and rail system as it existed in the 1990s may slightly underestimate doses for shipments that occurred in the 1940s, 1950s, and 1960s, because a larger portion of the transport routes would have been on non-interstate highways where the population may have been closer to the road. Data were not available that correlated transportation routes and population densities for the 1940s, 1950s, 1960s, and 1970s; therefore, it was necessary to use more recent data to make dose estimates. By the 1970s, the structure of the interstate highway system was largely fixed and most shipments would have been made on interstates.

Shipment data were linearly extrapolated for years when data were unavailable, which also results in uncertainty. However, this technique was validated by linearly extrapolating the data in the *Historical Overview of Domestic Spent Fuel Shipments–Update* (SAIC 1991) for 1973 through 1989 to estimate the number of shipments that took place during the time period 1964 through 1972 (also contained in SAIC 1991). The data in the historical overview could not be used directly because only shipment counts are presented for 1964 through 1982, and no origins or destinations were listed for years before 1983. Based on the data in the historical overview, linearly extrapolating the data for 1973 through 1989 overestimates the shipments for 1964 through 1972 by 20 percent when compared to the actual shipment counts for 1964 through 1972.

Transportation impacts may also result from reasonably foreseeable projects, such as the transportation impacts contained in other DOE National Environmental Policy Act analyses. The results of these analyses are summarized in Table L–11. For some of these analyses, a preferred alternative was not identified nor a ROD issued. In those cases, the alternative that was estimated to result in the largest transportation impact was included in Table L–11.

There are also reasonably foreseeable projects that involve limited transportation of radioactive material: (a) shipment of submarine reactor compartments from the Puget Sound Naval Shipyard to Hanford for burial, (b) return of cesium 137 isotope capsules to Hanford, (c) shipment of uranium billets from Hanford to the United Kingdom, and (d) shipment of low-specific-activity nitric acid from Hanford to the United Kingdom. While this is not an exhaustive list of projects that may involve limited transportation of radioactive material, it does illustrate that the transportation impacts associated with these types of projects are extremely low when compared to major projects or general transportation.

There are also general transportation activities that take place that are unrelated to the alternatives evaluated in the SPD EIS or to reasonably foreseeable actions. Examples of these activities are shipments of radiopharmaceuticals to nuclear medicine laboratories and shipments of commercial low-level radioactive waste to commercial disposal facilities. The NRC evaluated these types of shipments based on a survey of radioactive materials transportation published in NUREG-0170 (NRC 1977). Categories of radioactive material evaluated in NUREG-0170 included: (a) limited quantity shipments, (b) medical, (c) industrial, (d) fuel cycle, and (e) waste.

The NRC estimated that the annual collective worker dose for these shipments was 5,600 person-rem. The annual collective general population dose for these shipments was estimated to be 4,200 person-rem. Because comprehensive transportation doses were not available, these collective dose estimates were used to estimate

transportation collective doses for 1943 through 1982 (40 years). These dose estimates included spent nuclear fuel and radioactive waste shipments made by truck and rail.

Based on the transportation dose assessments in NUREG-0170, the cumulative transportation collective doses for 1943 through 1982 were estimated to be 220,000 person-rem for workers and 170,000 person-rem for the general population.

In 1983, another survey of radioactive materials transportation in the United States was conducted (Javitz et al. 1985). This survey included NRC and Agreement State licensees. Both spent nuclear fuel and radioactive waste shipments were included in the survey. Weiner, LaPlante, and Hageman (1991a:661–666, 1991b:665–660) used the survey by Javitz et al. (1985) to estimate collective doses from general transportation. The transportation dose assessments in Weiner, LaPlante, and Hageman (1991a:661–666, 1991b:665–660) were used to estimate transportation doses for 1983 through 2048 (66 years). Weiner, LaPlante, and Hageman (1991a:661–666) evaluated eight categories of radioactive material shipments by truck: (a) industrial, (b) radiography, (c) medical, (d) fuel cycle, (e) research and development, (f) unknown, (g) waste, and (h) other. Based on a median external exposure rate, an annual collective worker dose of 1,400 person-rem and an annual collective general population dose of 1,400 person-rem were estimated. Over the 66-year time period from 1983 through 2048, both the collective worker and general population doses were estimated to be 92,000 person-rem.

Weiner, LaPlante, and Hageman (1991b:655–660) also evaluated six categories of radioactive material shipments by plane: (a) industrial, (b) radiography, (c) medical, (d) research and development, (e) unknown, and (f) waste. Based on a median external exposure rate, an annual collective worker dose of 290 person-rem and an annual collective general population dose of 450 person-rem were estimated. Over the 66-year time period from 1983 through 2048, the collective worker dose was estimated to be 19,000 person-rem and the general population collective dose was estimated to be 30,000 person-rem.

Like the historical transportation dose assessments, the estimates of collective doses from general transportation also exhibit considerable uncertainty. For example, data for 1975 were applied to general transportation activities from 1943 through 1982. This approach probably overestimates doses because the amount of radioactive material that was transported in the 1950s and 1960s was less than the amount shipped in the 1970s. For example, in 1968, the shipping rate for radioactive material packages was estimated to be 300,000 packages per year (Patterson 1968:199–209); in 1975, this rate was estimated to be 2,000,000 packages per year (NRC 1977). However, because comprehensive data that would enable a more realistic transportation dose assessment are not available, the dose estimates developed by NRC were used.

Total collective worker doses from all types of shipments (historical, reasonably foreseeable actions, and general transportation) were estimated to be approximately 350,000 person-rem (140 LCFs), for the period of time 1943 through 2048 (106 years). Total general population collective doses were also estimated to be 330,000 person-rem (170 LCFs). The majority of the collective dose for workers and the general population was because of general transportation of radioactive material. The total number of LCFs over the time period 1943 through 2048 was estimated to be 310. Over this same period of time (106 years), about 54,060,000 people would die from cancer, based on 510,000 LCFs per year (DOC 1993). It should be noted that the estimated number of transportation-related LCFs would be indistinguishable from other LCFs, and the transportation-related LCFs would be 0.0000057 percent of the total number of expected LCFs during this timeframe.

L.7.2 Accident Impacts

For transportation accidents involving radioactive material, the dominant risk is from accidents that are unrelated to the cargo (i.e., traffic or vehicular accidents). Fatalities involving the shipment of radioactive materials were surveyed for 1971 through 1993 using the Radioactive Material Incident Report database. For 1971 through 1993, 21 vehicular accidents involving 36 fatalities occurred. These fatalities resulted from vehicular accidents

and were not associated with the radioactive nature of the cargo; no radiological fatalities because of transportation accidents have ever occurred in the United States. During the same period of time, over 1,100,000 persons were killed in vehicular accidents in the United States (National Safety Council 1994). About 100 additional vehicular accident fatalities were estimated to result from the transportation of radioactive material (i.e., the transportation associated with reasonably foreseeable actions and general radioactive materials transportation). During the 39-year time period from 2010 through 2048, approximately 1,600,000 people would be expected to be killed in vehicular accidents in the United States. The vehicular accident fatalities associated with radioactive materials transportation would be expected to be 0.006 percent of the total number of vehicular accident fatalities.

L.8 UNCERTAINTY AND CONSERVATISM IN ESTIMATED IMPACTS

The sequence of analyses performed to generate the estimates of radiological risk for the transportation includes: (1) determination of the inventory and characteristics, (2) estimation of shipment requirements, (3) determination of route characteristics, (4) calculation of radiation doses to exposed individuals (including estimation of environmental transport and uptake of radionuclides), and (5) estimation of health effects. Uncertainties are associated with each of these steps. Uncertainties exist in the way that the physical systems being analyzed are represented by the computational models, in the data required to exercise the models (due to measurement errors, sampling errors, natural variability, or unknowns simply caused by the future nature of the actions being analyzed), and in the calculations themselves (e.g., approximate algorithms used by the computers).

In principle, the uncertainty associated with each input or computational source can be estimated and the resultant uncertainty in each set of calculations can be predicted. Thus, the uncertainties from one set of calculations to the next can be propagated and the uncertainty in the final or absolute result can be estimated; however, conducting such a full-scale quantitative uncertainty analysis is often impractical and sometimes impossible, especially for actions to be initiated at an unspecified time in the future. Instead, the risk analysis is designed to ensure, through uniform and judicious selection of scenarios, models, and input parameters, that relative comparisons of risk among the various alternatives are meaningful. In the transportation risk assessment, this design is accomplished by uniformly applying common input parameters and assumptions to each alternative. Therefore, although considerable uncertainty is inherent in the absolute magnitude of the transportation risk for each alternative, much less uncertainty is associated with the relative differences among the alternatives in a given measure of risk.

In the following sections, areas of uncertainty are discussed for the assessment steps enumerated above. Special emphasis is placed on identifying whether the uncertainties affect relative or absolute measures of risk. The degree of conservatism of the assumption is addressed. Where practical, the parameters that most significantly affect the risk assessment results are identified.

L.8.1 Uncertainties in Material Inventory and Characterization

The inventories and the physical and radiological characteristics are important input parameters to the transportation risk assessment. The potential amount of transportation for any alternative is determined primarily by the projected nuclear material inventory and assumptions concerning shipment capacities. The physical and radiological characteristics are important in determining the amount of material released during accidents and the subsequent doses to exposed individuals through multiple environmental exposure pathways.

Uncertainties in the inventory and characterization will be reflected to some degree in the transportation risk results. If the inventory is overestimated (or underestimated), the resulting transportation risk estimates also will be overestimated (or underestimated) by roughly the same factor. However, the same inventory estimates are used to analyze the transportation impacts of each of the SPD EIS alternatives. Therefore, for comparative

purposes, the observed differences in transportation risks among alternatives are believed to represent unbiased, reasonably accurate estimates from current information in terms of relative risk comparisons.

No detailed characterization of surplus nonpit plutonium was included in the evaluation of each shipment of this material. Such information typically would not be compiled until actual shipments were being planned. Only global, conservative assumptions were used in the impact analysis. For the purpose of analysis, DOE assumed a maximum of 4.5 kg (9.9 lb) of plutonium per package, and 40 packages per SST/SGT. Actual SST/SGT shipments could handle more material. This leads to a conservative estimate of radiological accident risks for shipment of surplus nonpit plutonium for each alternative. However, since such shipments have been shown to have lower radiological accident risks than shipments of either plutonium dioxides from pits or lead assembly spent fuel, the overall effect would be very small.

L.8.2 Uncertainties in Containers, Shipment Capacities, and Number of Shipments

The amount of transportation required for each alternative is based, in part, on assumptions concerning the packaging characteristics and shipment capacities for commercial trucks and safe, secure transports. Changes in loading, tiedown, or packaging practices could affect estimates. Representative shipment capacities were defined for assessment purposes based on probable future shipment capacities. In reality, the actual shipment capacities may differ from the predicted capacities, so the projected number of shipments, and consequently the total transportation risk, would change. However, although the predicted transportation risks would increase or decrease accordingly, the relative differences in risks among alternatives would remain about the same. The maximum amount of material allowed in Type B containers is set by conservative safety analyses.

L.8.3 Uncertainties in Route Determination

Representative routes were determined between all origin and destination sites considered in the SPD EIS. The routes were determined consistent with current guidelines, regulations, and practices, but may not be the actual routes that would be used in the future. In reality, the actual routes could differ from the representative ones in terms of distances and total population along the routes. Moreover, since radioactive materials could be transported over an extended period of time starting at some time in the future, the highway infrastructures and the demographics along routes could change. These effects were not accounted for in the transportation assessment; however, it is not anticipated that these changes would significantly affect relative comparisons of risk among the alternatives considered in the SPD EIS. The dates and times that specific transportation routes would be used are classified.

L.8.4 Uncertainties in the Calculation of Radiation Doses

The models used to calculate radiation doses from transportation activities introduce a further uncertainty in the risk assessment process. It is generally difficult to estimate the accuracy or absolute uncertainty of the risk assessment results. The accuracy of the calculated results is closely related to the limitations of the computational models and the uncertainties in each of the input parameters that the model requires. The single greatest limitation facing users of RADTRAN, or any computer code of this type, is the scarcity of data for certain input parameters.

Uncertainties associated with the computational models are minimized by using state-of-the-art computer codes that have undergone extensive review. Because there are numerous uncertainties that are recognized but difficult to quantify, assumptions are made at each step of the risk assessment process that are intended to produce conservative results (i.e., overestimate the calculated dose and radiological risk). Because parameters and assumptions are applied to all alternatives, this model bias is not expected to affect the meaningfulness of relative comparisons of risk; however, the results may not represent risks in an absolute sense.

The single largest contributor to the collective population doses calculated with RADTRAN was found to be the dose to members of the public at truck stops. Currently, RADTRAN uses a simple point-source approximation for truck-stop exposures and assumes that the total stop time for a shipment is proportional to the shipment distance. The parameters used in the stop model were based on a survey of a very limited number of radioactive material shipments that examined a variety of shipment types in different areas of the country. It was assumed that stops occur as a function of distance, with a stop rate of 0.011 hr/km (0.018 hr/mi). For non-SST/SGT shipments, it was further assumed that an average of 50 people at each stop are exposed at a distance of 20 m (66 ft). In RADTRAN, the population dose is directly proportional to the external shipment dose rate and the number of people exposed, and inversely proportional to the square of the distance. For this assessment, it was assumed that many shipments (nonpit plutonium and spent nuclear fuel) would have external dose rates at the regulatory limit of 10 mrem/hr at 2 m (6.6 ft). In practice, the external dose rates would vary from shipment to shipment. The stop rate assumed results in an hour of stop time per 100 km (62 mi) of travel.

Based on the qualitative discussion with shippers, the parameter values used in the assessment appear to be conservative. However, data do not exist to quantitatively assess the degree of control, location, frequency, and duration of truck stops. However, based on the regulatory requirements of 10 CFR 73 for continuous escort of the material and the requirement for two drivers, it is clear that the trucks would be on the move much of the time until arrival at the destination. Therefore, the calculated impacts are extremely conservative. By using these conservative parameters, the calculations in the SPD EIS are consistent with the RADTRAN published values.

Shielding exposed populations is not considered. For all incident-free exposure scenarios, no credit has been taken for shielding exposed individuals. In reality, shielding would be afforded by trucks and cars sharing the transport routes, rural topography, and the houses and buildings in which people reside. Incident-free exposure to external radiation could be reduced significantly depending on the type of shielding present. For residential houses, shielding factors (i.e., the ratio of shielded to unshielded exposure rates) were estimated to range from 0.02 to 0.7, with a recommended value of 0.33. If shielding were to be considered for the maximally exposed resident living near a transport route, the calculated doses and risks would be reduced by approximately 70 percent. Similar levels of shielding may be provided to individuals exposed in vehicles.

Postaccident mitigative actions were not considered for dispersal accidents. For severe accidents involving the release and dispersal of radioactive materials in the environment, no postaccident mitigative actions, such as interdiction of crops or evacuation of the accident vicinity, were considered in this risk assessment. Postaccident mitigative measures to reduce groundshine doses (evacuation and/or decontamination) are assumed to occur 24 hours after the accident in RADTRAN analyses. Additionally, RADTRAN assumes that highly contaminated crops are not ingested (Neuhauser and Knipe 1995). Since RISKIND is modeling the worst credible accident, these measures were not considered. In reality, mitigative actions would take place following an accident in accordance with U.S. Environmental Protection Agency radiation protection guides for nuclear incidents (EPA 1992). The effects of mitigative actions on population accident doses are highly dependent on the severity, location, and timing of the accident. For this risk assessment, ingestion doses were only calculated for accidents occurring in rural areas (the calculated ingestion doses; however, it assumed, all food grown on contaminated ground is consumed and is not limited to the rural population). Interdiction of foodstuffs would act to reduce, but not eliminate, this contribution.

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Appendix M
Analysis of Environmental Justice

M.1 INTRODUCTION

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low-Income Populations*, directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse health or environmental effects of their programs, policies, and activities on minority and low-income populations.

The Council on Environmental Quality (CEQ) has oversight responsibility for documentation prepared in compliance with the National Environmental Policy Act (NEPA). In December 1997, the CEQ released guidance on environmental justice (CEQ 1997). The CEQ's guidance was adopted as the basis for the analysis of environmental justice contained in the *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS).

M.2 DEFINITIONS AND APPROACH

The following definitions were used in the analysis of environmental justice (CEQ 1997):

> **Low-income population:** Low-income populations in an affected area should be identified with the annual statistical poverty thresholds from the U.S. Bureau of the Census' Current Population Reports, Series P-60 on Income and Poverty. In identifying low-income populations, agencies may consider as a community either a group of individuals living in geographic proximity to one another, or a set of individuals (such as migrant workers or Native Americans), where either type of group experiences common conditions of environmental exposure or effect.
>
> **Minority:** Individual(s) who are members of the following population groups: American Indian or Alaskan Native; Asian or Pacific Islander; Black, not of Hispanic origin; or Hispanic.
>
> **Minority population:** Minority populations should be identified where either: (a) the minority population of the affected area exceeds 50 percent or (b) the minority population percentage of the affected area is meaningfully greater than the minority population percentage in the general population or other appropriate unit of geographic analysis. In identifying minority communities, agencies may consider as a community either a group of individuals living in geographic proximity to one another, or a geographically dispersed/transient set of individuals (such as migrant workers or American Indians), where either type of group experiences common conditions of environmental exposure or effect. The selection of the appropriate unit of geographic analysis may be a governing body's jurisdiction, a neighborhood, census tract, or other similar unit that is to be chosen so as to not artificially dilute or inflate the affected minority population. A minority population also exists if there is more than one minority group present and the minority percentage, as calculated by aggregating all minority persons, meets one of the above-stated thresholds.
>
> **Disproportionately high and adverse human health effects:** When determining whether human health effects are disproportionately high and adverse, agencies are to consider the following three factors to the extent practical:

a. Whether the health effects, which may be measured in risks and rate, are significant (as employed by NEPA), or above generally accepted norms. Adverse health effects may include bodily impairment, infirmity, illness, or death;

b. Whether the risk or rate of hazard exposure by a minority population or low-income population to an environmental hazard is significant (as employed by NEPA) and appreciably exceeds, or is likely to appreciably exceed, the risk or rate to the general population or other appropriate comparison group; and

c. Whether health effects occur in a minority or low-income population affected by cumulative or multiple adverse exposures from environmental hazards.

Disproportionately high and adverse environmental effects: When determining whether environmental effects are disproportionately high and adverse, agencies are to consider the following three factors to the extent practical:

a. Whether there is, or will be, an impact on the natural or physical environment that significantly (as employed by NEPA) and adversely affects a minority or low-income population. Such effects may include ecological, cultural, human health, economic, or social impacts on minority communities or low-income communities, when those impacts are interrelated to impacts on the natural or physical environment;

b. Whether environmental effects are significant (as employed by NEPA) and are or may be having an adverse impact on minority populations or low-income populations that appreciably exceeds, or is likely to appreciably exceed, those on the general population or other appropriate comparison group; and

c. Whether the environmental effects occur, or would occur, in a minority population or low-income population affected by cumulative or multiple adverse exposures from environmental hazards.

Data for the analysis of minorities were extracted from Table P12 of Summary Tape File 3A published on CD ROM by the Census Bureau (DOC 1992). Data for the analysis of low-income populations were extracted from Table P121 of Standard Tape File 3A.

Potentially affected areas examined in the SPD EIS include the areas surrounding proposed facilities for plutonium disposition located at four candidate DOE sites: the Hanford Site (Hanford), Idaho National Engineering and Environmental Laboratory (INEEL), the Pantex Plant (Pantex), and the Savannah River Site (SRS). Other potentially affected areas examined include the areas surrounding proposed reactor sites for mixed oxide (MOX) fuel irradiation: Catawba Nuclear Station, McGuire Nuclear Station, and North Anna Power Station. Minority and low-income populations residing within a 1.6-km (1-mi) corridor centered on representative transportation routes were also included in the evaluation of environmental justice.

M.3 SPATIAL RESOLUTION

For the purposes of enumeration and analysis, the Census Bureau has defined a variety of areal units (DOC 1992). Areal units of concern in this document include (in order of increasing spatial resolution): States, counties, census tracts, block groups, and blocks. The "block" is generally the smallest of these entities and offers the finest spatial resolution. This term refers to a relatively small geographical area bounded on all sides by visible features such as streets and streams, or by invisible boundaries such as city limits or property lines. During the 1990 census, the Census Bureau subdivided the United States and its territories into 7,017,425 blocks. For comparison, the number of counties, census tracts, and block groups used in the 1990 census were 3,248; 62,276; and 229,192; respectively. While blocks offer the finest spatial resolution, economic data required for identification of low-income populations are not available at the block-level of spatial resolution. In the analysis

below, block groups are used throughout as the areal unit. Block groups generally contain between 250 and 500 housing units (DOC 1992:A-4).

During the decennial census, the Census Bureau collects data from individuals and then aggregates the data according to residence in geographical areas such as counties or block groups. Boundaries of the areal units are selected to coincide with geographical features, such as streams and roads, or political boundaries, such as county and city borders. Boundaries used for aggregation of the census data usually do not coincide with boundaries used in the calculation of health effects. As discussed in Chapter 4 of the SPD EIS, radiological health effects due to an accident at one of the disposition facilities or reactor sites are evaluated for persons residing within a distance of 80 km (50 mi) of the accident site. In general, the boundary of the circle with an 80-km (50-mi) radius centered at the accident site will not coincide with boundaries used by the Census Bureau for enumeration of the population in the potentially affected area. Some block groups lie completely inside or outside the area included in the calculation of health effects. However, block groups intersecting the boundary of the potentially affected area are only partly included. Partial inclusion of block groups is illustrated in Figure M–1. This figure shows the block group structure near Idaho Falls, Idaho. The 80-km (50-mi) radius shown in this figure denotes the boundary used for calculation of health effects in the event of a radiological release at the Fuel and Materials Examination Facility (FMEF) at INEEL. Block groups that are unshaded in Figure M–1 lie within an 80-km (50-mi) radius centered at FMEF, and the total population of these block groups is included in the population count. Block groups shaded in gray lie outside of the circle, and the population of the shaded block groups is excluded from the population count. However, block groups such as those that are cross-hatched in Figure M–1 lie only partly within the circle. Because the geographical distribution of persons residing within a block group is not available from the census data, partial inclusions introduce uncertainties into the estimate of the population at risk.

In order to evaluate populations at risk in partially included block groups, it was assumed that residents are uniformly distributed throughout the area of each block group. For example, if 85 percent of the area of a block group lies within 80 km (50 mi) of the accident site, then it was assumed that 85 percent of the population residing in that block group would be at risk. An upper bound for the population at risk was obtained by including the total population of partially included block groups in the population at risk. Similarly, a lower bound for the population at risk was obtained by excluding the population of partially included blocks from the population at risk. As a general rule, if the areas of geographic units defined by the Census Bureau are small in comparison with the potentially affected area, then the uncertainties due to partial inclusions will be relatively small. Uncertainties in the estimates of populations surrounding disposition facilities and reactor sites are described in Appendixes M.5.1 and M.7.1, respectively.

M.4 POPULATION PROJECTIONS

In Chapter 4 and Appendixes J, K, and L of the SPD EIS, health effects were calculated for populations projected to reside in potentially affected areas during 2010 and 2015. Extrapolations of the total population for individual States are available from both the Census Bureau and various State agencies (Campbell 1996). The Census Bureau also projects populations by ethnic and racial classification in 1-year intervals for the years from 1995 to 2025. Data used to project minority populations in the SPD EIS were extracted from the Census Bureau's Web site (www.census.gov/population/www/projections/stproj.html). Minority populations determined from the 1990 census data were taken as a baseline. It was then assumed that percentage changes in the minority and majority populations of each block group for a given year (compared with the 1990 baseline data) would be the same as percentage changes in the State minority and majority populations projected for the same year. An advantage to this assumption is that the projected populations are obtained with consistent methodology regardless of the State and associated block group involved in the calculation. A disadvantage is that the methodology is insensitive to localized demographic changes that could alter the projection for a specific area.

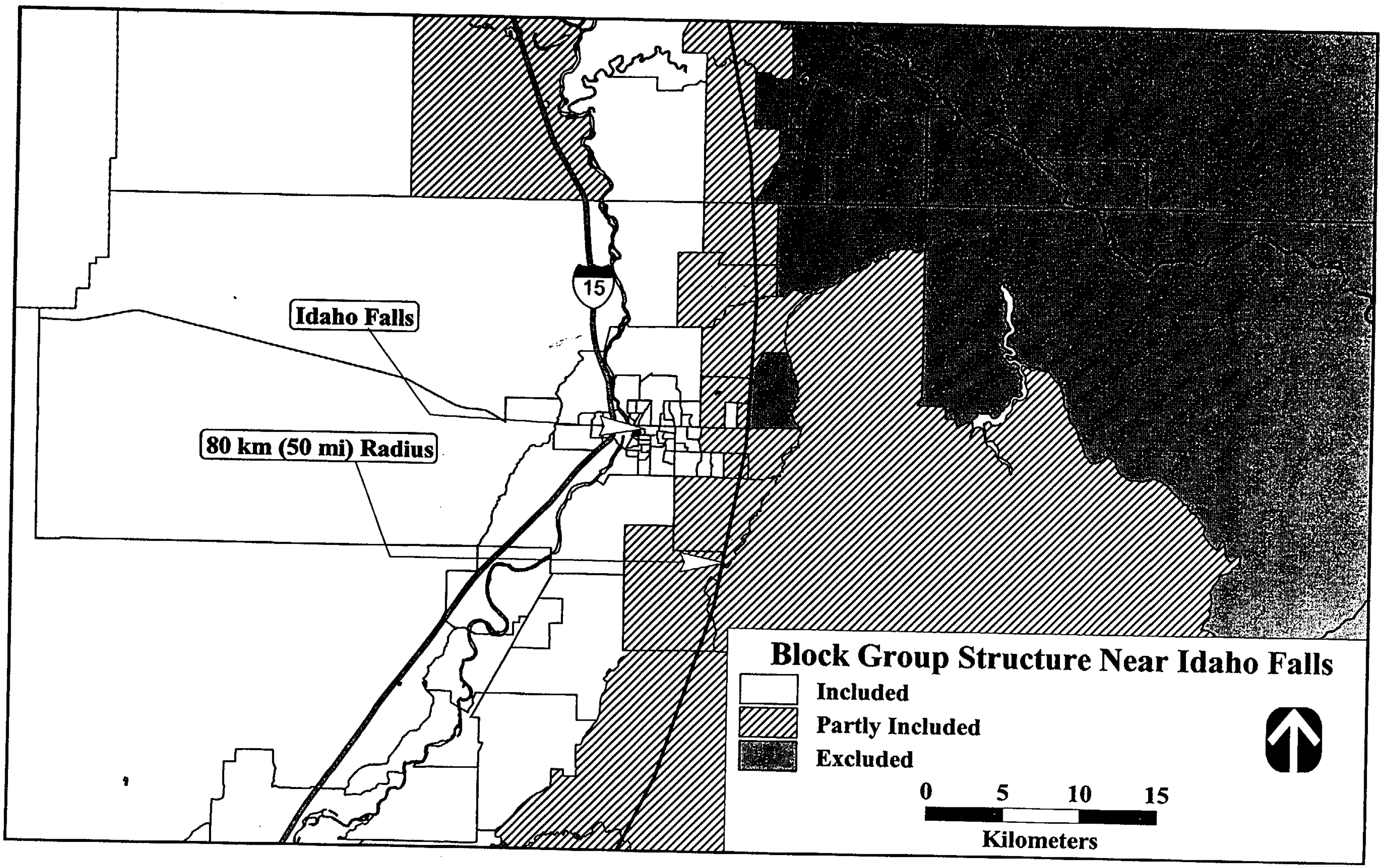

Figure M–1. Block Group Structure Near Idaho Falls, Idaho

The Census Bureau uses the cohort-component method to estimate future populations for each State (Campbell 1996). The set of cohorts is composed of: (1) age groups from 1 year or less to 85 years or more (in 1-year intervals), (2) male and female populations in each age group, and (3) the following racial and ethnic groups in each age group—Hispanic, non-Hispanic Asian, non-Hispanic Black, non-Hispanic Native American, and non-Hispanic White. Components of the population change used in the demographic accounting system are births, deaths, net State-to-State migration, and net international migration. If *P(t)* denotes the number of individuals in a given cohort at time *t*, then:

$$P(t) = P(t_0) + B - D + DIM - DOM + IIM - IOM$$

where:

$P(t_0)$	=	cohort population at time $t_0 \leq t$, where t_0 denotes the year 1990.
B	=	births expected during the period from t_0 to t.
D	=	deaths expected during the period from t_0 to t.
DIM	=	domestic migration expected into the State during the period from t_0 to t.
DOM	=	domestic migration expected out of the State during the period from t_0 to t.
IIM	=	international migration expected into the State during the period from t_0 to t.
IOM	=	international migration expected out of the State during the period from t_0 to t.

Estimated values for the components shown on the right side of the equation are based on past data and various assumptions regarding changes in the rates for birth, mortality, and migration (Campbell 1996). The Census Bureau does not project populations of individuals who identified themselves as "Other Race" during the 1990 census. This population group is less than 2 percent of the total population in each of the States. In order to project total populations in the environmental justice analysis, population projections for the "Other Race" group were made under the assumption that the growth rate for the "Other Race" population will be identical to the growth rate for the combined minority and White (non-Hispanic) populations.

M.5 RESULTS FOR THE CANDIDATE DOE SITES

M.5.1 Population Estimates

Table M–1 shows total populations, minority populations, and percentage minority populations that resided within 80 km (50 mi) of the various sites at the time of the 1990 census. The 80-km (50-mi) distance defines the radius of potential radiological effects for calculations of radiation dose to the general population (see Chapter 4 of the SPD EIS). Tables M–2 and M–3 show similar data for projected populations in 1997 and 2010. As discussed above, minority populations residing in potentially affected areas in 1990 were adopted as a baseline. Populations in 1997 and 2010 were then projected from the baseline data under the assumption that percentage changes in the majority and minority populations residing in the affected areas will be identical to those projected for State populations. The Census Bureau estimates that the national minority percentage will increase from approximately 24 percent in 1990 to 27 percent in 1997, and nearly 33 percent by 2010 (Campbell 1996). Percentage minority populations residing within 80 km (50 mi) of facilities at Hanford and SRS are projected to exceed the national percentage by year 2010. Percentage minority populations surrounding facilities at INEEL and Pantex were less than the national minority percentage in 1990 and are projected to remain so through the year 2010. In Tables M–1 through M–3, the sum of percentages shown in even-numbered columns beginning in column 6 may total slightly more or less than 100 percent due to roundoff.

Table M–4 illustrates the uncertainties in the population estimates for the year 2010 due to the partial inclusion of block groups within the boundaries of potentially affected areas. Column 2 of the table lists the number of

Table M–1. Racial and Ethnic Composition of Minority Populations Residing Within 80 km of Candidate DOE Sites in 1990

Candidate Site	Total Pop.	Minority Pop.	Percent Minority Pop.	Asian or Pacific Islander Pop.	Percent Asian or Pacific Islander Pop.	Black Pop.	Percent Black Pop.	Hispanic Pop.	Percent Hispanic Pop.	Native American Pop.	Percent Native American Pop.	Other Race	Percent Other Race Pop.	White Pop.	Percent White Pop.
Hanford 400 Area	277,515	70,493	25.4	3,989	1.4	2,788	1.0	59,736	21.5	3,981	1.4	372	0.1	206,651	74.5
Hanford 200 East	346,031	90,526	26.2	4,852	1.4	4,144	1.2	74,490	21.5	7,040	2.0	556	0.2	254,949	73.7
INEEL	119,138	11,757	9.9	1,166	1.0	385	0.3	7,154	6.0	3,052	2.6	135	0.1	107,246	90.0
Pantex	266,004	50,778	19.1	3,450	1.3	11,130	4.2	33,977	12.8	2,220	0.8	363	0.1	214,864	80.7
[Text deleted.]															
SRS APSF, if built	614,095	232,781	37.9	5,888	1.0	219,136	35.7	6,456	1.1	1,300	0.2	175	0.0	381,139	62.1
SRS DWPF	626,317	241,168	38.5	5,951	1.0	227,378	36.3	6,521	1.0	1,319	0.2	175	0.0	384,974	61.5

Key: APSF, Actinide Packaging and Storage Facility; DWPF, Defense Waste Processing Facility.

Table M–2. Projected Racial and Ethnic Composition of Minority Populations Residing Within 80 km of Candidate DOE Sites in 1997

Candidate Site	Total Pop.	Minority Pop.	Percent Minority Pop.	Asian or Pacific Islander Pop.	Percent Asian or Pacific Islander Pop.	Black Pop.	Percent Black Pop.	Hispanic Pop.	Percent Hispanic Pop.	Native American Pop.	Percent Native American Pop.	Other Race	Percent Other Race Pop.	White Pop.	Percent White Pop.
Hanford 400 Area	324,640	98,586	30.4	5,640	1.7	3,153	1.0	85,642	26.4	4,151	1.3	418	0.1	225,636	69.5
Hanford 200 East	396,420	126,166	31.8	6,885	1.7	4,666	1.2	106,551	26.9	8,064	2.0	631	0.2	269,623	68.0
INEEL	145,117	16,785	11.6	1,627	1.1	590	0.4	10,793	7.4	3,775	2.6	166	0.1	128,166	88.3
Pantex	292,004	62,845	21.5	5,107	1.7	12,801	4.4	42,490	14.6	2,447	0.8	414	0.1	228,745	78.3
[Text deleted.]															
SRS APSF, if built	694,891	274,985	39.6	9,276	1.3	254,807	36.7	9,456	1.4	1,447	0.2	201	0.0	419,704	60.4
SRS DWPF	688,352	275,654	40.0	9,332	1.4	255,459	37.1	9,422	1.4	1,441	0.2	201	0.0	412,497	59.9

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INEEL	145,117	16,785	11.6	1,627	1.1	590	0.4	10,793	7.4	3,775	2.6	166	0.1	128,166	88.3
Pantex	292,004	62,845	21.5	5,107	1.7	12,801	4.4	42,490	14.6	2,447	0.8	414	0.1	228,745	78.3
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SRS DWPF	688,352	275,654	40.0	9,332	1.4	255,459	37.1	9,422	1.4	1,441	0.2	201	0.0	412,497	59.9

block groups that are partly within the circle of 80-km (50-mi) radius centered at the various facilities. Column 3 shows the number of block groups that lie completely within the circle. Potentially affected areas surrounding Hanford and SRS include two States. Columns 2 and 3 show the number of partial or total inclusions for the affected States. Column 4 of the table, denoted as "T/P," shows the number of totally included block groups divided by the number of partially included block groups. In order to minimize the uncertainties in the population estimate, it is desirable that this ratio be as large as possible. Column 5 shows upper bounds for the estimates of the total population listed in column 6. As discussed above, upper bounds were obtained by including the total population of all block groups that lie at least partially within the affected area. Lower bounds for the estimate of total population shown in column 7 were obtained by including only the populations of totally included block groups. Analogous statements apply to columns 8 through 10.

As would be expected from the value of T/P shown in column 4, uncertainties in the total population estimate for Pantex were the smallest among the four sites (+2.4 percent and -2.7 percent), as were the uncertainties in the estimate of the minority population at risk near Pantex (+1.9 percent and -1.9 percent). Uncertainties in the population estimates for INEEL were the largest among the four sites (+17.2 percent and -15.2 percent for total population; +17.3 percent and -15.0 percent for minority population). None of the uncertainties shown in Table M–4 are large enough to noticeably affect the conclusions regarding radiological health effects or environmental justice.

M.5.2 Geographical Dispersion of Minority and Low-Income Populations

Figures M–2 through M–9 show the geographical distributions of minority and low-income populations at risk in the vicinity of the candidate DOE sites. Distributions shown in these figures are based on baseline population data for 1990. Even-numbered figures show the geographical distribution of minority populations in potentially affected areas within a distance of 80 km (50 mi) of candidate facilities. Block groups are shaded to indicate the percentage of the total population comprised of minorities. According to the decennial census of 1990, minorities comprised 24.2 percent of the total population of the contiguous United States. Block groups unshaded in the even-numbered figures are those for which the percentage of minority residents is less than the national percentage minority population. Areas shaded in gray show block groups for which the percentage of minority residents exceeds the national minority percentage by less than a factor of two. Diagonally hatched block groups shown in the even-numbered figures are those for which the percentage of minority residents exceeds the national minority percentage by a factor of two or more.

Odd-numbered figures show the geographical distribution of low-income populations potentially at risk from implementation of the proposed action or alternatives. According to the decennial census of 1990, 13.4 percent of the population of the contiguous United States reported incomes less than the poverty threshold. Block groups unshaded in Figures M–1, M–5, M–7, and M–9 are those for which the percentage of low-income residents is less than the national percentage of persons reporting an income less than the poverty threshold. Areas shaded in gray show block groups for which the percentage of low-income residents exceeds the national low-income percentage by less than a factor of two. Diagonally hatched block groups shown in the odd-numbered figures are those for which the percentage of low-income residents exceeds the national low-income percentage by a factor of two or more.

M.5.3 Environmental Effects on Minority and Low-Income Populations Residing Near Candidate DOE Sites

The analysis of environmental effects on populations residing within 80 km (50 mi) of proposed facilities is presented in Chapter 4 of the SPD EIS. This analysis shows that no radiological fatalities are likely to result from implementation of the proposed action or alternatives. Radiological risks to the public are small regardless of the racial and ethnic composition of the population, and regardless of the economic status of

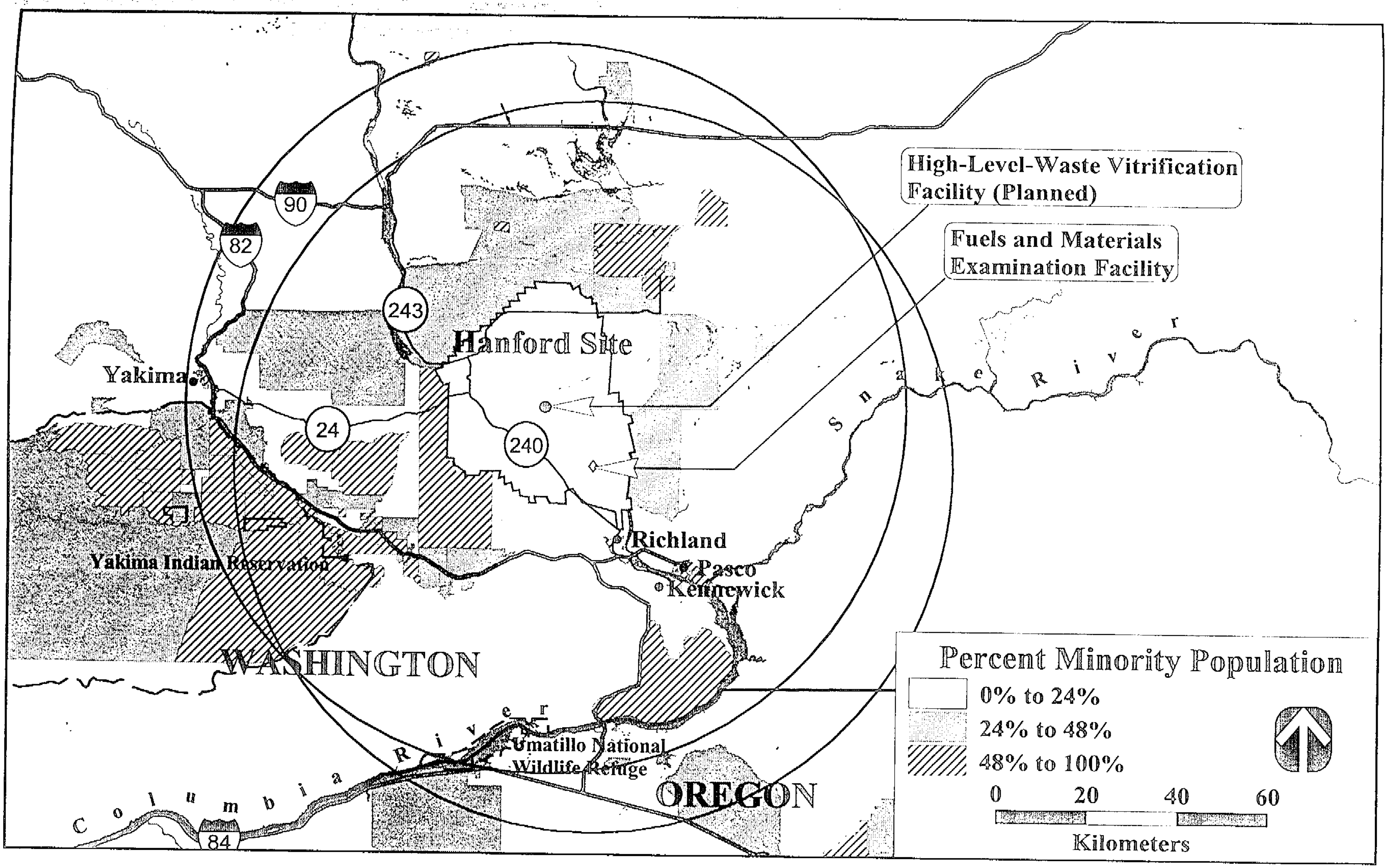

Figure M–2. Geographical Distribution of the Minority Population Residing Within 80 km (50 mi) of Proposed Facilities at Hanford

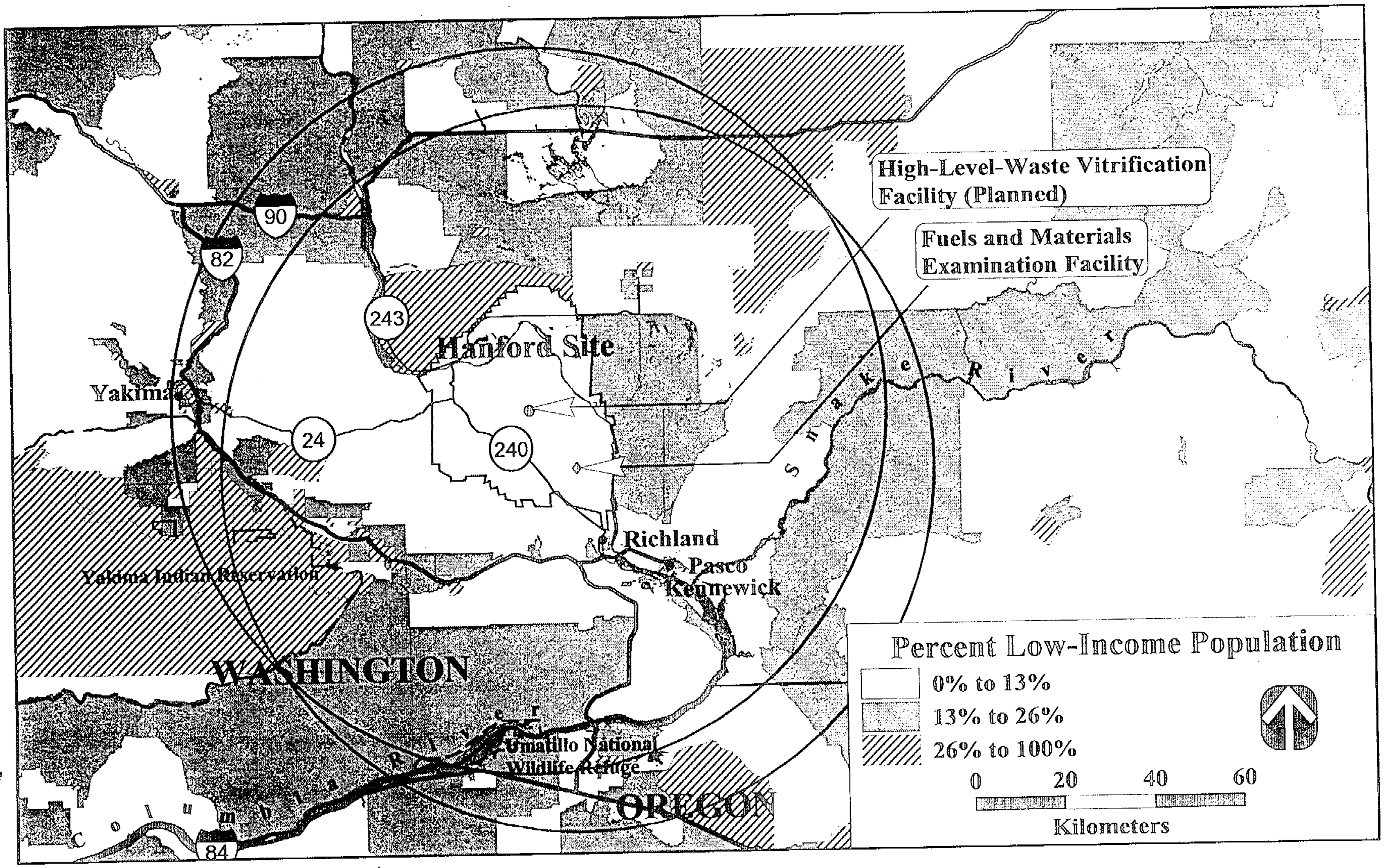

Figure M–3. Geographical Distribution of the Low-Income Population Residing Within 80 km (50 mi) of Proposed Facilities at Hanford

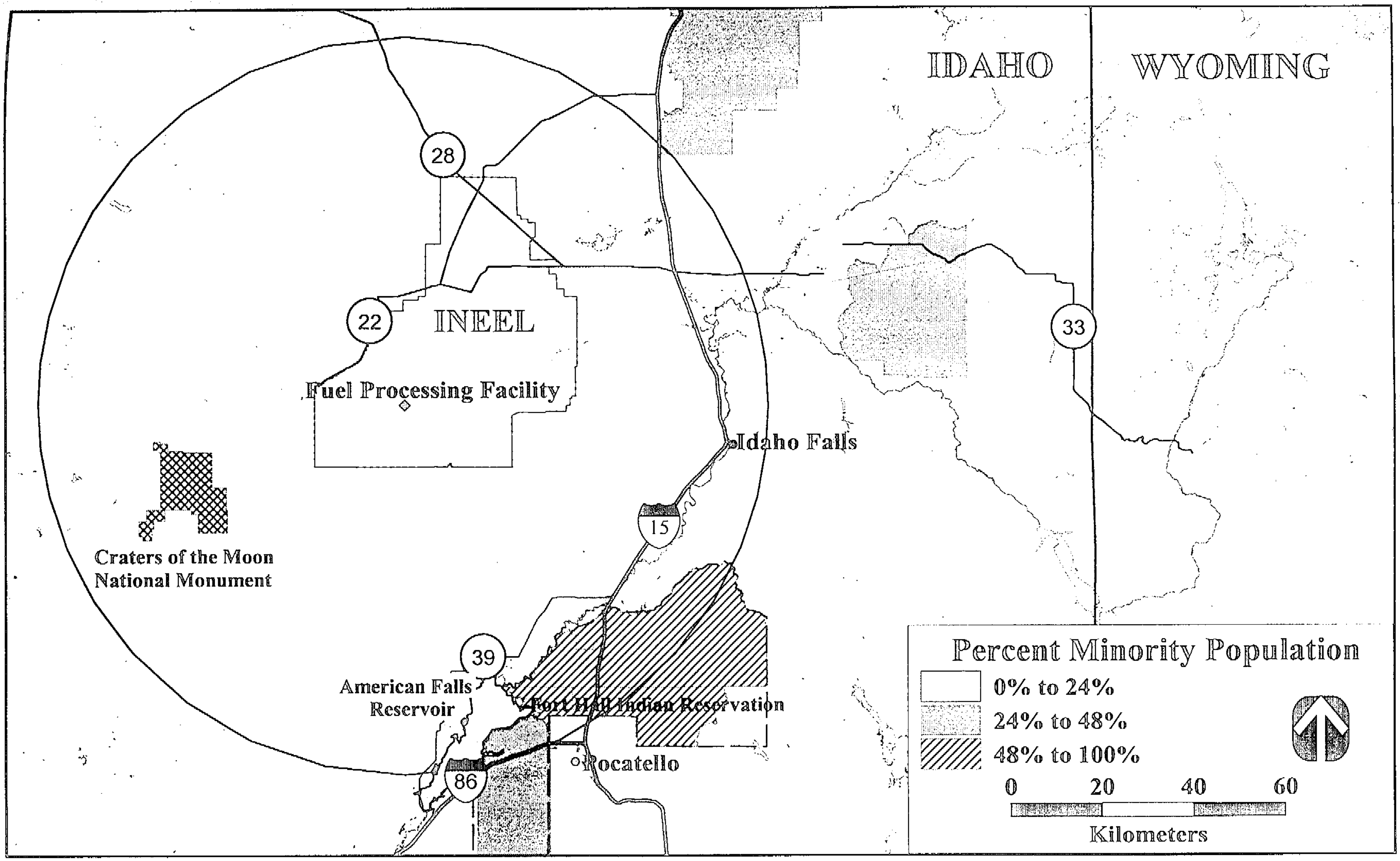

Figure M–4. Geographical Distribution of the Minority Population Residing Within 80 km (50 mi) of Fuel Processing Facility at INEEL

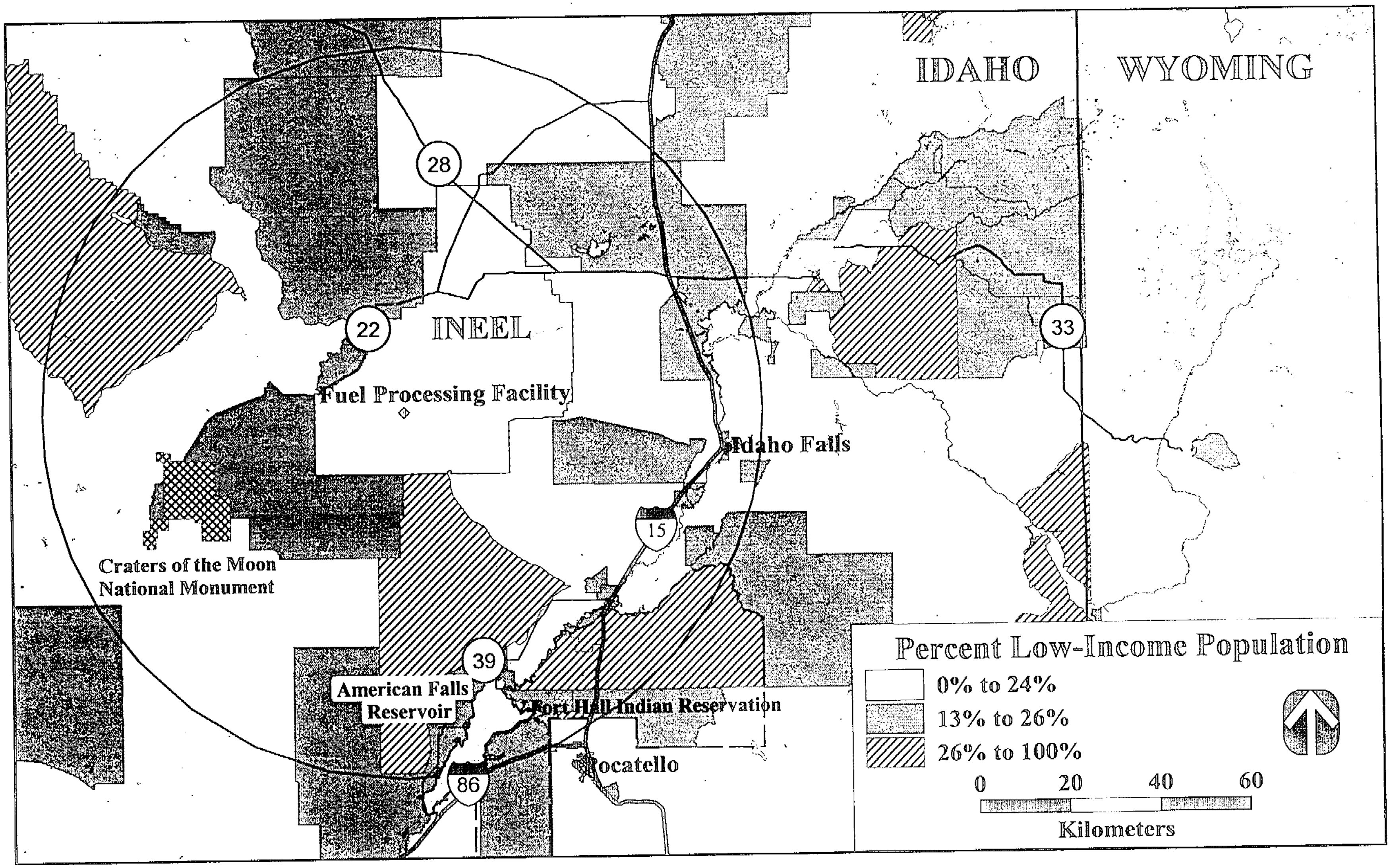

Figure M–5. Geographical Distribution of the Low-Income Population Residing Within 80 km (50 mi) of Fuel Processing Facility at INEEL

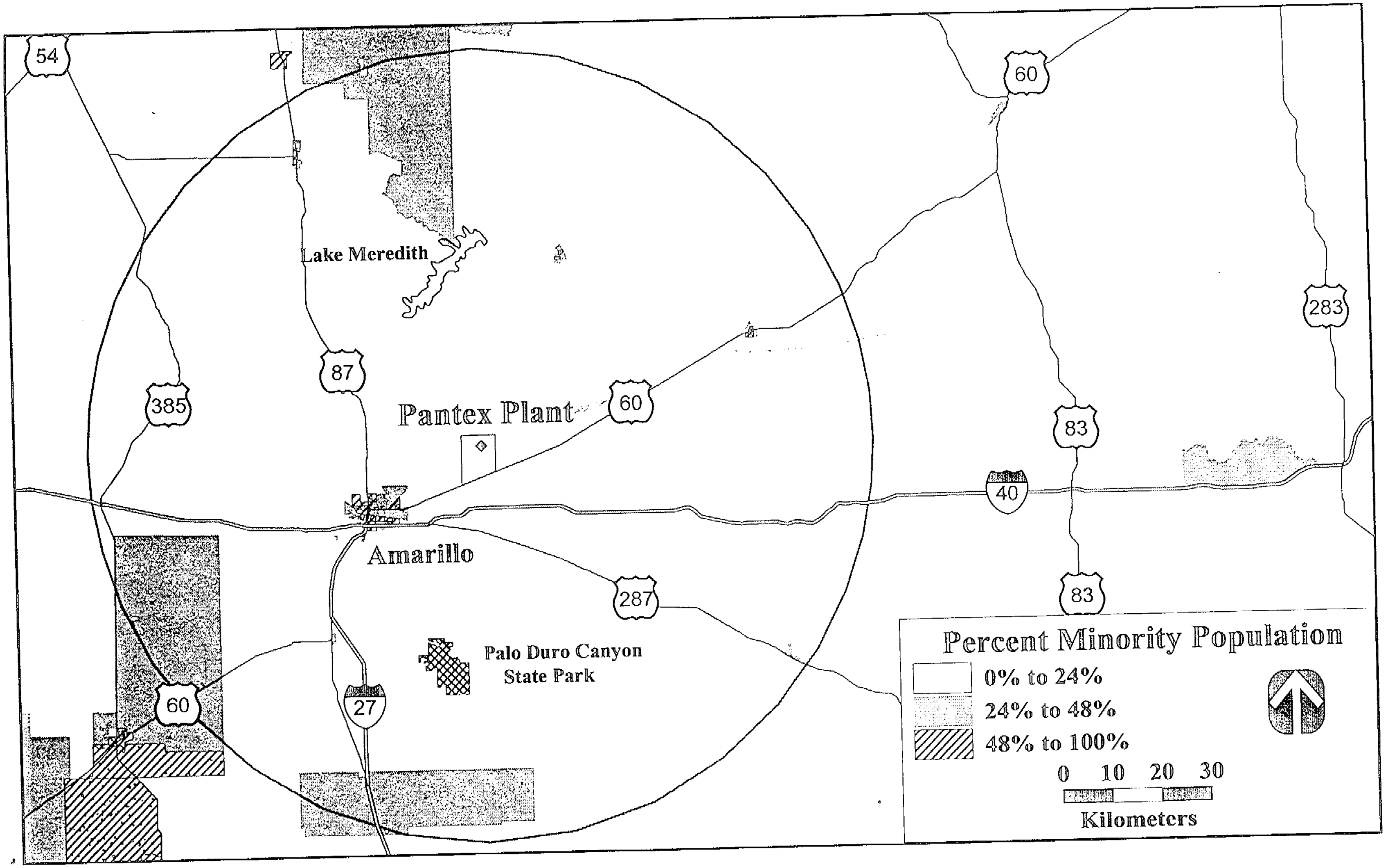

Figure M–6. Geographical Distribution of the Minority Population Residing Within 80 km (50 mi) of Potentially Affected Area at Pantex

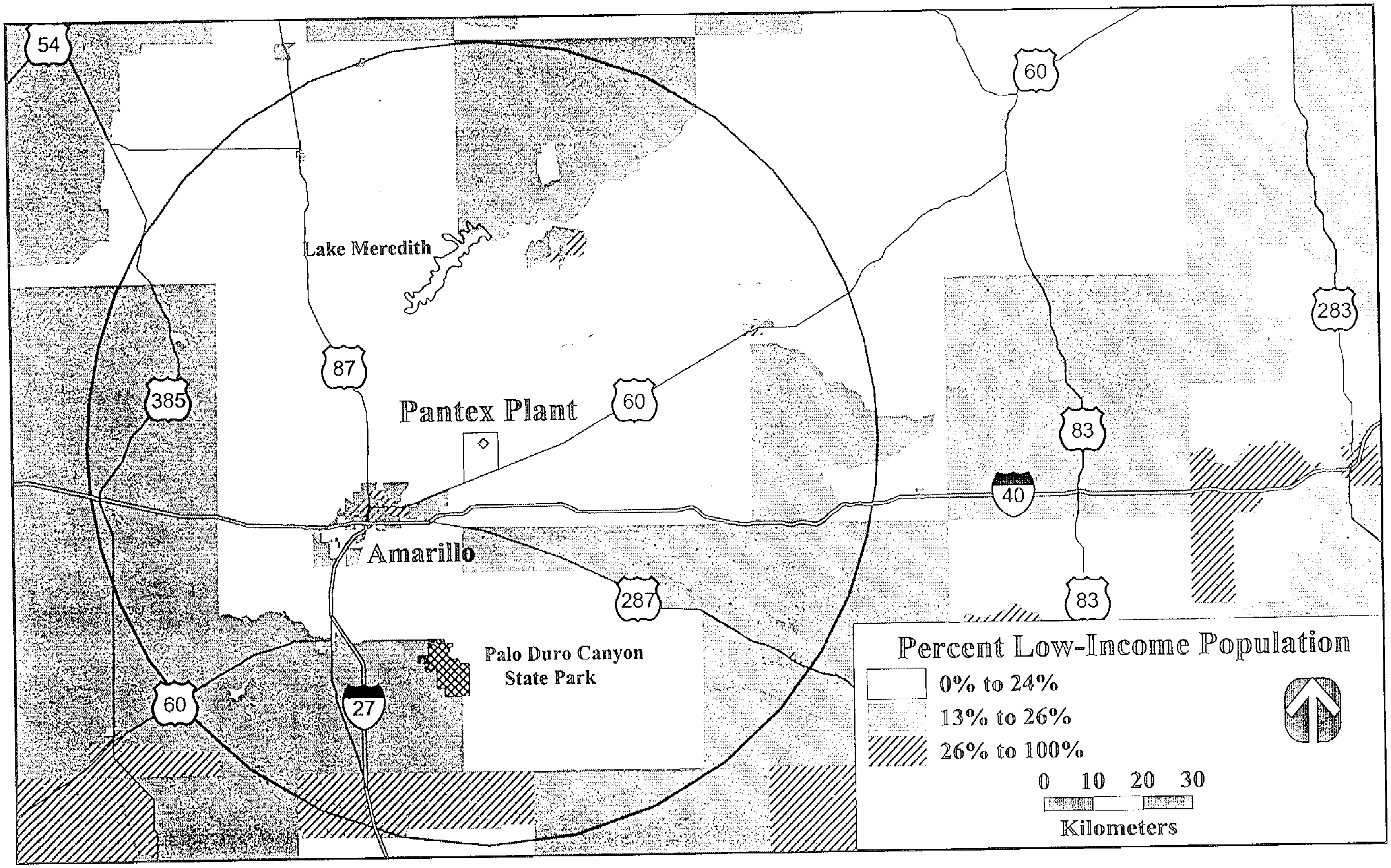

Figure M–7. Geographical Distribution of the Low-Income Population Residing Within 80 km (50 mi) of Potentially Affected Area at Pantex

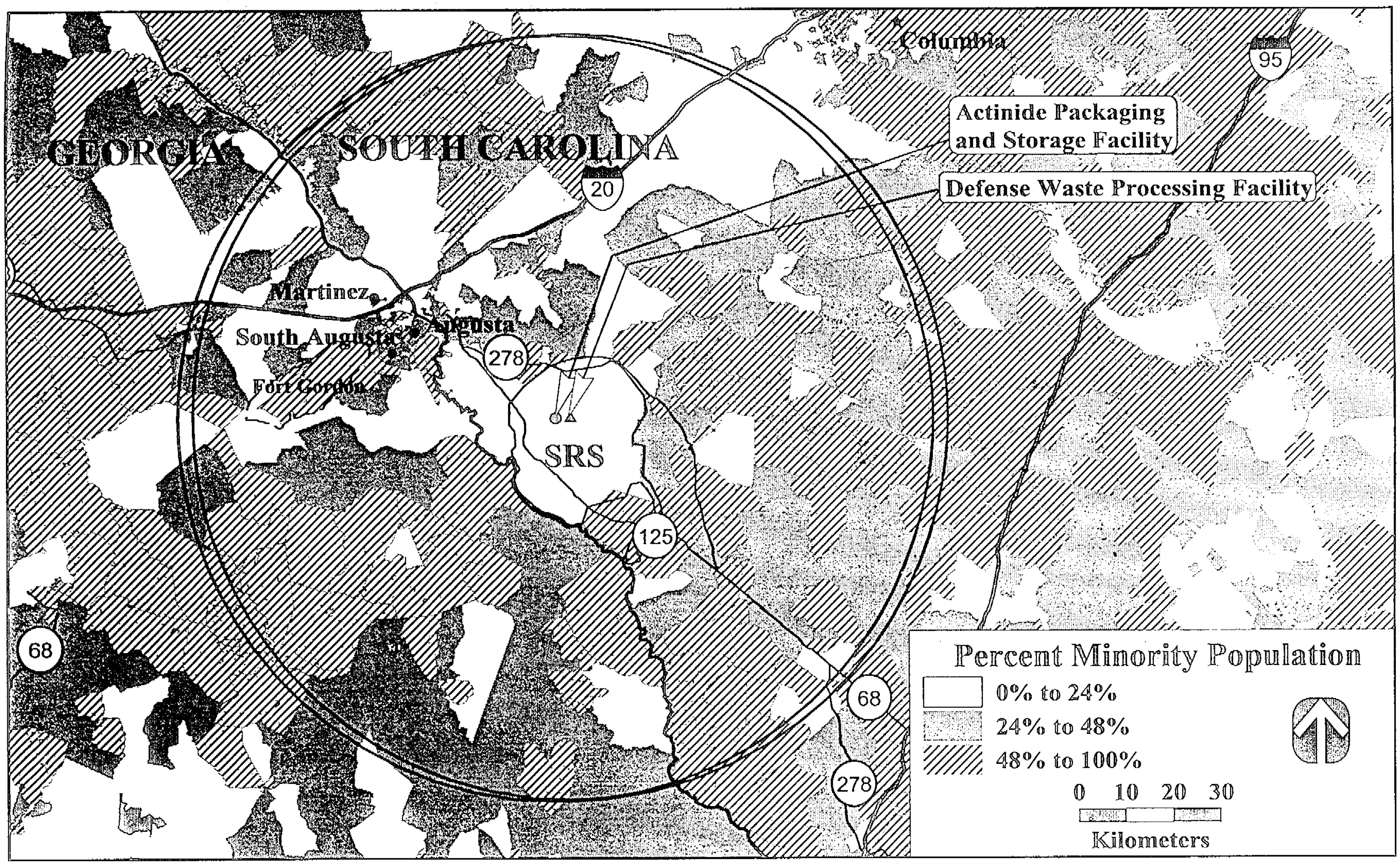

Figure M–8. Geographical Distribution of the Minority Population Residing Within 80 km (50 mi) of Proposed Facilities at SRS

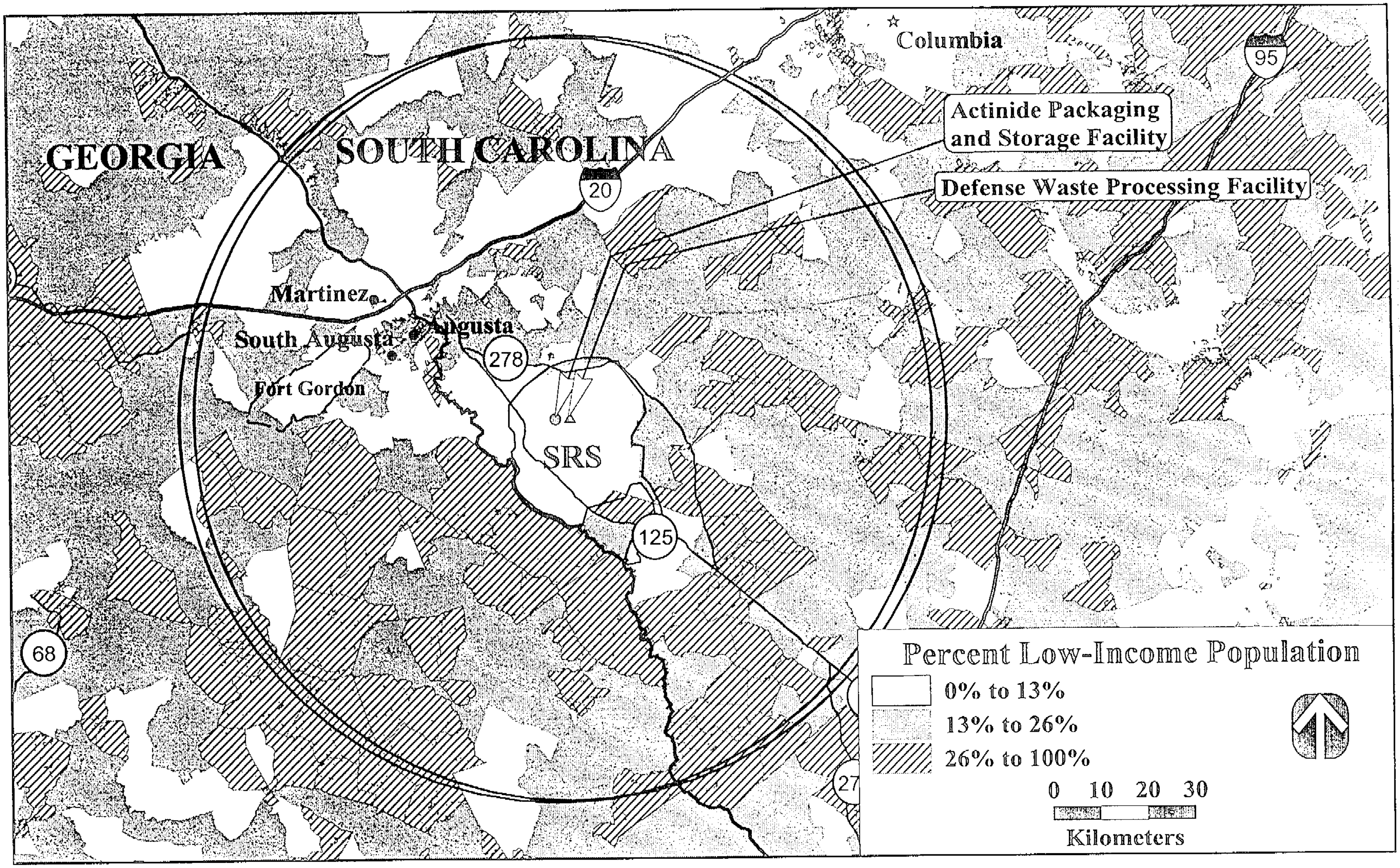

Figure M–9. Geographical Distribution of the Low-Income Population Residing Within 80 km (50 mi) of Proposed Facilities at SRS

individuals comprising the population. Nonradiological risks to the general population are also small regardless of the racial and ethnic composition or economic status of the population. Thus, disproportionately high and adverse impacts on minority and low-income populations residing near the various facilities are not likely to result from implementation of the proposed action or alternatives.

M.6 RESULTS FOR TRANSPORTATION ROUTES

Table M–5 shows minority populations residing along 1.6-km (1-mi) corridors centered on routes that are representative of those that could be used for the transportation of nuclear materials under the proposed action or alternatives. Table M–6 shows similar data for low-income populations. Population data for Tables M–5 and M–6 were extracted from Tables P–12 and P–121 of the STF–3A files (DOC 1992). Distances from a given origin to a given destination are similar but not identical to corresponding distances shown in Appendix L. This is because distances listed in Appendix L were calculated with the HIGHWAY computer code, while distances shown in Tables M–5 and M–6 were obtained from a Geographical Information System analysis using TigerLine data and STF3A files prepared by the Census Bureau. Both techniques use block group spatial resolution, and the differences are generally less than 5 percent.

Total and minority populations residing in the highway corridors are listed in Columns 4 and 5, respectively, of Table M–5. Column 6 shows minority populations residing within highway corridors as a percentage of the total population. Although total and minority populations residing within the corridors generally tend to increase with increasing distance, the relationship is clearly route dependent.

As discussed in Appendix L of the SPD EIS, implementation of the proposed action or alternatives would not result in significant radiological or nonradiological risks to populations residing along highway transportation routes. Although the percentage minority or low-income populations residing along highway routes can vary by as much as a factor of four, results of the analysis presented in Chapter 4 are independent of the racial and ethnic composition of populations within the corridors, as well as the economic status of populations at risk within the corridors. Implementation of the proposed action or alternatives is not likely to result in disproportionately high and adverse effects on minority or low-income populations residing within representative transportation corridors.

Table M–5. Minority Populations Residing Along Transportation Routes for Surplus Plutonium

Origin	Destination	Distance (km)	Total Population Along Route	Minority Population Along Route	Percentage Minority Population Along Route
ANL–W	Hanford	1,035	82,418	9,356	11.4
ANL–W	Pantex	2,395	281,386	82,566	29.3
ANL–W	SRS	3,756	580,985	122,415	21.1
Fuel fabrication	Hanford	4,760	601,233	95,417	15.9
Fuel fabrication	INEEL	4,092	556,388	88,331	15.9
Fuel fabrication	LANL	3,201	506,962	126,460	24.9
Fuel fabrication	Pantex	2,563	430,359	87,635	20.4
Fuel fabrication	SRS	578	75,050	30,702	40.9
Hanford	Geological repository	1,888	248,006	31,424	12.7
Hanford	INEEL	949	74,624	8,927	12.0
Hanford	LANL	2,515	276,768	71,860	26.0
Hanford	ORR	3,993	434,235	62,000	14.3
Hanford	Pantex	3,040	342,903	92,151	26.9
INEEL	ORR	3,316	389,496	59,174	15.2
INEEL	SRS	3,702	574,433	123,656	21.5
LANL	ANL–W	1,868	230,510	60,265	26.1
LANL	INEEL	1,840	227,759	65,563	28.8
LANL	LLNL	1,218	454,603	224,303	49.3
LANL	Pantex	647	85,252	35,326	41.4
LANL	SRS	2,779	521,907	163,376	31.3
LLNL	Fuel fabrication	4,838	771,701	257,880	33.4
LLNL	Geological repository	1,140	414,432	192,001	46.3
LLNL	Hanford	1,428	380,755	50,764	13.3
LLNL	INEEL	1,559	373,040	72,575	19.5
LLNL	Pantex	2,302	476,701	226,661	47.5
LLNL	SRS	4,395	856,464	403,622	47.1
Pantex	Geological repository	1,986	186,981	66,118	35.4
Pantex	INEEL	2,365	293,805	85,783	29.2
Pantex	ORR	1,753	245,038	59,671	24.4
Pantex	SRS	2,165	441,441	126,441	28.6
Pantex	WIPP	538	121,377	37,477	30.9
Portsmouth, OH	Fuel fabrication	977	239,221	40,636	17.0
RFETS	Hanford	1,848	141,585	23,178	16.4
RFETS	INEEL	1,170	104,960	17,791	17.0
RFETS	Pantex	1,252	252,177	81,450	32.3
RFETS	SRS	2,954	540,944	123,248	22.8
SRS	Hanford	4,377	615,204	126,016	20.5
SRS	ORR	568	109,074	15,614	14.3

Key: ANL–W, Argonne National Laboratory–West; LANL, Los Alamos National Laboratory; LLNL, Lawrence Livermore National Laboratory; ORR, Oak Ridge Reservation; RFETS, Rocky Flats Environmental Technology Site; WIPP, Waste Isolation Pilot Plant.

Table M–6. Low-Income Populations Residing Along Transportation Routes for Surplus Plutonium

Origin	Destination	Distance (km)	Total Population Along Route	Low-Income Population Along Route	Percentage Low-Income Population Along Route
ANL–W	Hanford	1,035	82,418	10,016	12.2
ANL–W	Pantex	2,395	281,386	44,102	15.7
ANL–W	SRS	3,756	580,985	60,473	10.4
Fuel fabrication	Hanford	4,760	601,233	61,518	10.2
Fuel fabrication	INEEL	4,092	556,388	55,229	9.9
Fuel fabrication	LANL	3,201	506,962	73,801	14.6
Fuel fabrication	Pantex	2,563	430,359	64,909	15.1
Fuel fabrication	SRS	578	75,050	10,673	14.2
Hanford	Geological repository	1,888	248,006	28,699	11.6
Hanford	INEEL	949	74,624	9,468	12.7
Hanford	LANL	2,515	276,768	42,384	15.3
Hanford	ORR	3,993	434,235	42,696	9.8
Hanford	Pantex	3,040	342,903	53,293	15.5
INEEL	ORR	3,316	389,496	39,171	10.1
INEEL	SRS	3,702	574,433	61,713	10.7
LANL	ANL–W	1,868	230,510	35,476	15.4
LANL	INEEL	1,840	227,759	35,984	15.8
LANL	LLNL	1,218	454,603	59,814	13.2
LANL	Pantex	647	85,252	12,635	14.8
LANL	SRS	2,779	521,907	80,398	15.4
LLNL	Fuel fabrication	4,838	771,701	103,519	13.4
LLNL	Geological repository	1,140	414,732	48,663	11.7
LLNL	Hanford	1,428	380,755	38,761	10.2
LLNL	INEEL	1,559	373,040	34,078	9.1
LLNL	Pantex	2,302	476,701	62,602	13.1
LLNL	SRS	4,395	856,464	136,322	15.9
Pantex	Geological repository	1,986	186,981	30,207	16.2
Pantex	INEEL	2,365	293,805	46,898	16.0
Pantex	ORR	1,753	245,038	44,137	18.0
Pantex	SRS	2,165	441,441	68,339	15.5
Pantex	WIPP	538	121,377	26,269	21.6
Portsmouth, OH	Fuel fabrication	977	239,221	33,268	13.9
RFETS	Hanford	1,848	141,585	15,985	11.3
RFETS	INEEL	1,170	104,960	10,424	9.9
RFETS	Pantex	1,252	252,177	41,478	16.4
RFETS	SRS	2,954	540,944	58,752	10.9
SRS	Hanford	4,377	615,204	65,311	10.6
SRS	ORR	568	109,074	13,061	12.0

ANL–W, Argonne National Laboratory–West; LANL, Los Alamos National Laboratory; LLNL, Lawrence Livermore National Laboratory; ORR, Oak Ridge Reservation; RFETS, Rocky Flats Environmental Technology Site; WIPP, Waste Isolation Pilot Plant.

M.7 RESULTS FOR THE REACTOR SITES

M.7.1 Minority and Low-Income Population Estimates

Table M–7 shows total populations, minority populations, and percentage minority populations that resided within 80 km (50 mi) of the various sites at the time of the 1990 census. The 80-km (50-mi) distance defines the radius of potential radiological effects for calculations of radiation dose to the general population. Table M–8 shows similar data for projected populations in 2015. As discussed in Appendix M.4, minority populations residing in potentially affected areas in 1990 were adopted as a baseline. Populations in 2015 were then projected from the baseline data under the assumption that percentage changes in the majority and minority populations residing in the affected areas will be identical to those projected for State populations. The Census Bureau estimates that the national minority percentage will increase from approximately 24 percent in 1990 to nearly 34 percent by 2015 (Census 1996). [Text deleted.] In Tables M–7 and M–8, the sum of percentages of the different populations may total slightly more or less than 100 percent due to roundoff.

Table M–9 illustrates the uncertainties in the population estimates for the year 2015 due to the partial inclusion of block groups within the boundaries of potentially affected areas. Column 2 of the table lists the number of block groups that are partly within the circle of 80-km (50-mi) radius centered at the various facilities. Column 3 shows the number of block groups that lie completely within the circle. Potentially affected areas surrounding all three of the proposed reactor sites include two States. Columns 2 and 3 show the number of partial or total inclusions for the affected States. Column 4 of the table, denoted as "T/P," shows the number of totally included block groups divided by the number of partially included block groups. In order to minimize the uncertainties in the population estimate, it is desirable that this ratio be as large as possible. Column 5 shows upper bounds for the estimates of the total population listed in column 6. As discussed above, upper bounds were obtained by including the total population of all block groups that lie at least partially within the affected area. Lower bounds for the estimate of total population shown in column 7 were obtained by including only the populations of totally included block groups. Analogous statements apply to columns 8 through 10.

As would be expected from the value of T/P shown in column 4, uncertainties in the total population estimate for McGuire were the smallest among the three proposed reactor sites (+3.7 percent and −2.4 percent), as were the uncertainties in the estimate of the minority population at risk near Catawba (+5.7 percent and −3.3 percent). Uncertainties in the population estimates for North Anna were the largest among the three sites (+6.5 percent and −4.5 percent for total population; +5.9 percent and −4.2 percent for minority population). None of the uncertainties shown in Table M–9 are large enough to noticeably affect the conclusions regarding radiological health effects or environmental justice.

An estimate of the percentage of low-income persons living within 80 km (50 mi) of the proposed reactor sites in 2015 was obtained using a linear projection of low-income data from the 1980 census and the 1990 census. In 1990, the percentage of low-income persons (i.e., those with reported incomes below the poverty threshold) residing in the contiguous United States was 13.1 percent. The percentage of low-income persons living within 80 km (50 mi) of the proposed reactor sites was lower than the national average in every case. Around Catawba, the percentage of low-income persons living within 80 km (50 mi), in 1990, was 10.5 percent. At McGuire, the percentage was 9.8 percent, and around North Anna, the percentage was 6.9 percent.

The estimated number of low-income persons living within 80 km (50 mi) of Catawba in 2015 is 157,477 or 7.0 percent of the projected population. The estimated number of low-income persons living within 80 km (50 mi) of McGuire in 2015 is 171,182 or 6.6 percent of the projected population. The estimated number of

Table M–7. Racial and Ethnic Composition of Minority Populations Residing Within 80 km of Proposed Reactor Sites in 1990

Reactor Site	Total Pop.	Minority Pop.	Percent Minority Pop.	Asian or Pacific Islander Pop.	Percent Asian or Pacific Islander Pop.	Black Pop.	Percent Black Pop	Hispanic Pop.	Percent Hispanic Pop.	Native American Pop.	Percent Native American Pop.	Other Race	Percent Other Race Pop.	White Pop.	Percent White Pop.
Catawba	1,519,392	315,089	20.7	10,942	0.7	288,382	19.0	10,666	0.7	5,098	0.3	442	0.0	1,203,861	79.2
McGuire	1,738,966	305,717	17.6	12,007	0.7	275,789	15.9	12,094	0.7	5,828	0.3	479	0.0	1,432,770	82.4
North Anna	1,286,156	281,652	21.9	18,783	1.5	241,619	18.8	17,550	1.4	3,686	0.3	947	0.1	1,003,557	78.0

Table M–8. Projected Racial and Ethnic Composition of Minority Populations Residing Within 80 km of Proposed Reactor Sites in 2015

Reactor Site	Total Pop.	Minority Pop.	Percent Minority Pop.	Asian or Pacific Islander Pop.	Percent Asian or Pacific Islander Pop.	Black Pop.	Percent Black Pop	Hispanic Pop.	Percent Hispanic Pop.	Native American Pop.	Percent Native American Pop.	Other Race	Percent Other Race Pop.	White Pop.	Percent White Pop.
Catawba	2,265,495	597,376	26.4	37,756	1.7	507,810	22.4	40,504	1.8	10,700	0.5	606	0.0	1,668,119	73.6
McGuire	2,575,369	620,701	24.1	43,333	1.7	517,577	20.1	46,486	1.8	12,635	0.5	670	0.0	1,954,668	75.9
North Anna	2,042,200	731,773	35.8	106,086	5.2	508,719	24.9	111,992	5.5	4,976	0.2	1,165	0.1	1,309,262	64.1

Table M–9. Uncertainties in Estimates of Total and Minority Populations for the Year 2015

Reactor Site	No. of Partially Included Block Groups	No. of Fully Included Block Groups	T/P	Upper Bound for Total Population	Estimate of Total Population	Lower Bound for Total Population	Upper Bound for Minority Population	Estimate of Minority Population	Lower Bound for Minority Population
Catawba	54 (NC) 52 (SC)	851 (NC) 314 (SC)	11.0	2,395,224	2,265,495	2,191,319	627,435	597,376	579,620
McGuire	64 (NC) 24 (SC)	1,190 (NC) 129 (SC)	15.0	2,672,795	2,575,369	2,513,292	636,842	620,701	611,521
North Anna	84 (VA) 10 (MD)	710 (VA) 5 (MD)	7.6	2,175,504	2,042,200	1,949,928	775,277	731,773	700,983

low-income persons living within 80 km (50 mi) of North Anna in 2015 is 110,531 or 5.4 percent of the projected population. [Text deleted.] Figures M–10 through M–15 show geographical distributions of minority and low-income populations residing with 80 km (50 mi) of the proposed reactor sites.

M.7.2 Environmental Effects on Minority and Low-Income Populations Residing Near Proposed Reactor Sites

The analysis of environmental effects on populations residing within 80 km (50 mi) of the proposed reactor sites is presented in Chapter 4 of the SPD EIS. This analysis shows that no radiological fatalities are likely to result from implementation of the proposed action or alternatives. Radiological risks to the public are small regardless of the racial and ethnic composition of the population, and regardless of the economic status of individuals comprising the population. Nonradiological risks to the general population are also small regardless of the racial and ethnic composition or economic status of the population. Thus, disproportionately high and adverse impacts on minority and low-income populations residing near the various facilities are not likely to result from implementation of the proposed action or alternatives.

M.8 REFERENCES

Campbell, P., 1996, *Population Projections: 1995–2025*, U.S. Department of Commerce, Bureau of the Census, Washington, DC, October.

CEQ (Council on Environmental Quality), 1997, *Environmental Justice, Guidance Under the National Environmental Policy Act*, Executive Office of the President, Washington, DC, December 10.

DOC (U.S. Department of Commerce), 1992, *Census of Population and Housing, 1990: Summary Tape File 3 on CD-ROM*, Bureau of the Census, Washington, DC, May.

DOC (U.S. Department of Commerce), 1996, *Resident Population of the United States: Middle Series Projections, 2015–2030, by Sex, Race, and Hispanic Origin, with Median Age*, Bureau of the Census, Washington, DC, March.

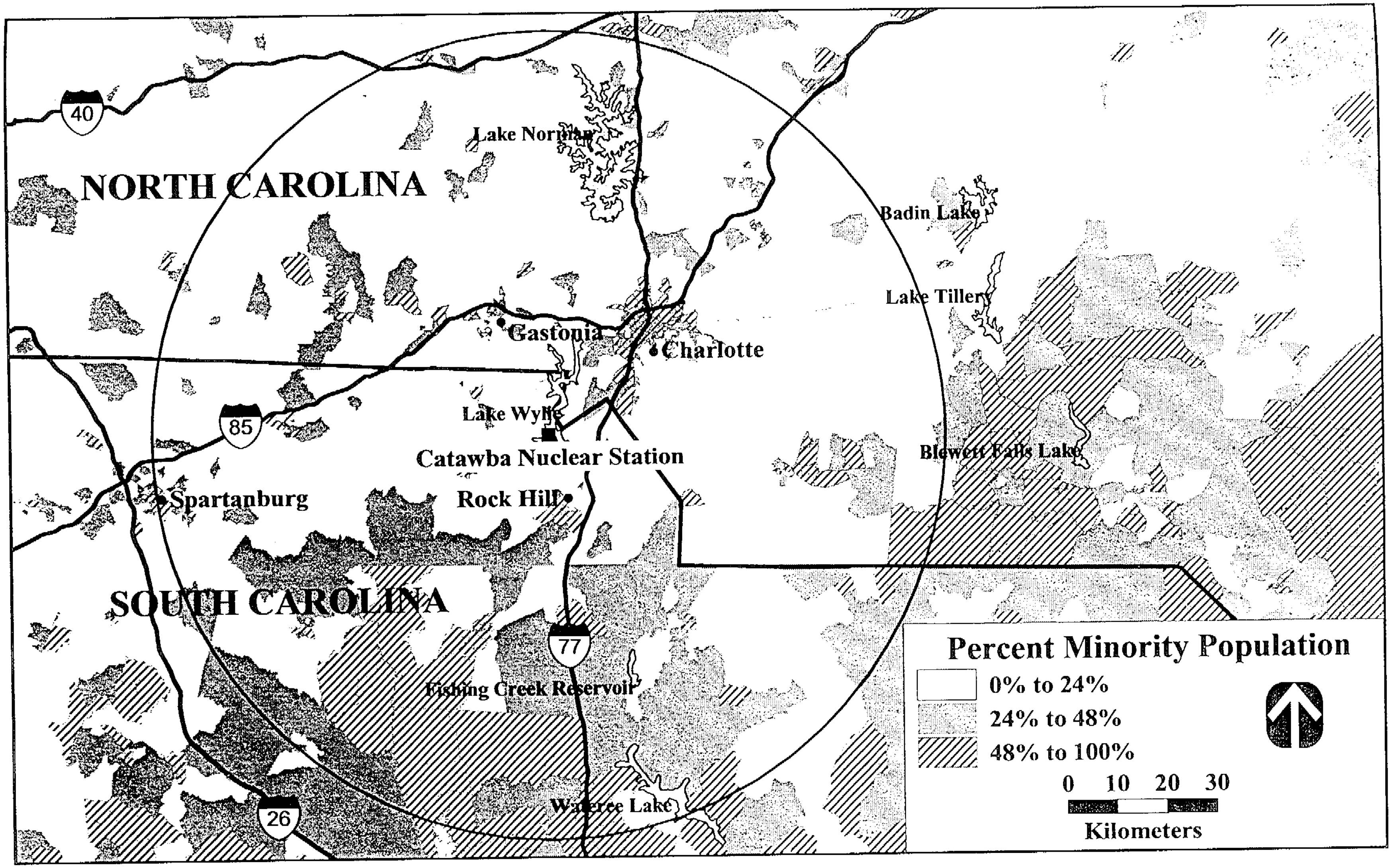

Figure M–10. Geographical Distribution of the Minority Population Residing Within 80 km (50 mi) of Catawba Nuclear Station

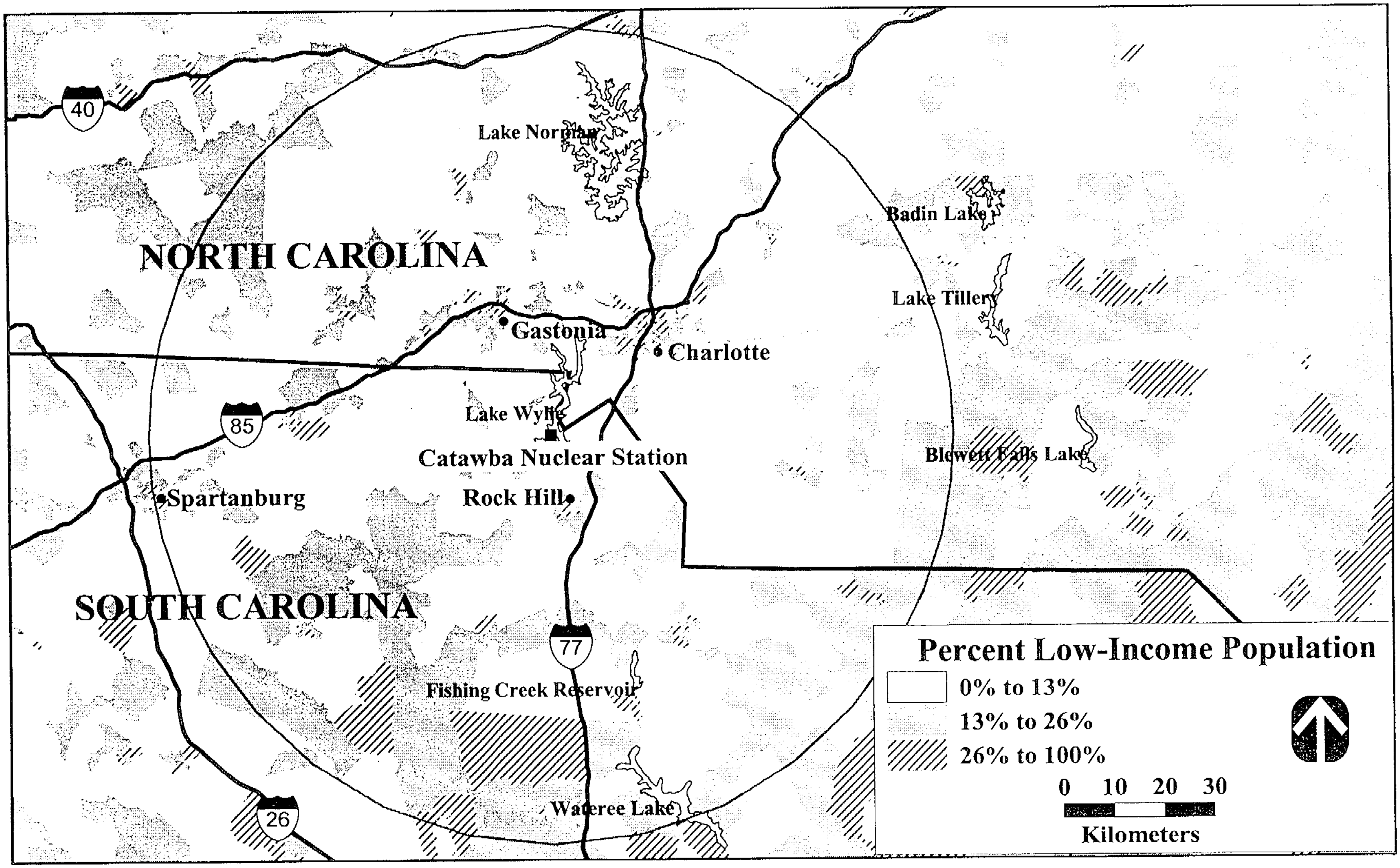

Figure M–11. Geographical Distribution of the Low-Income Population Residing Within 80 km (50 mi) of Catawba Nuclear Station

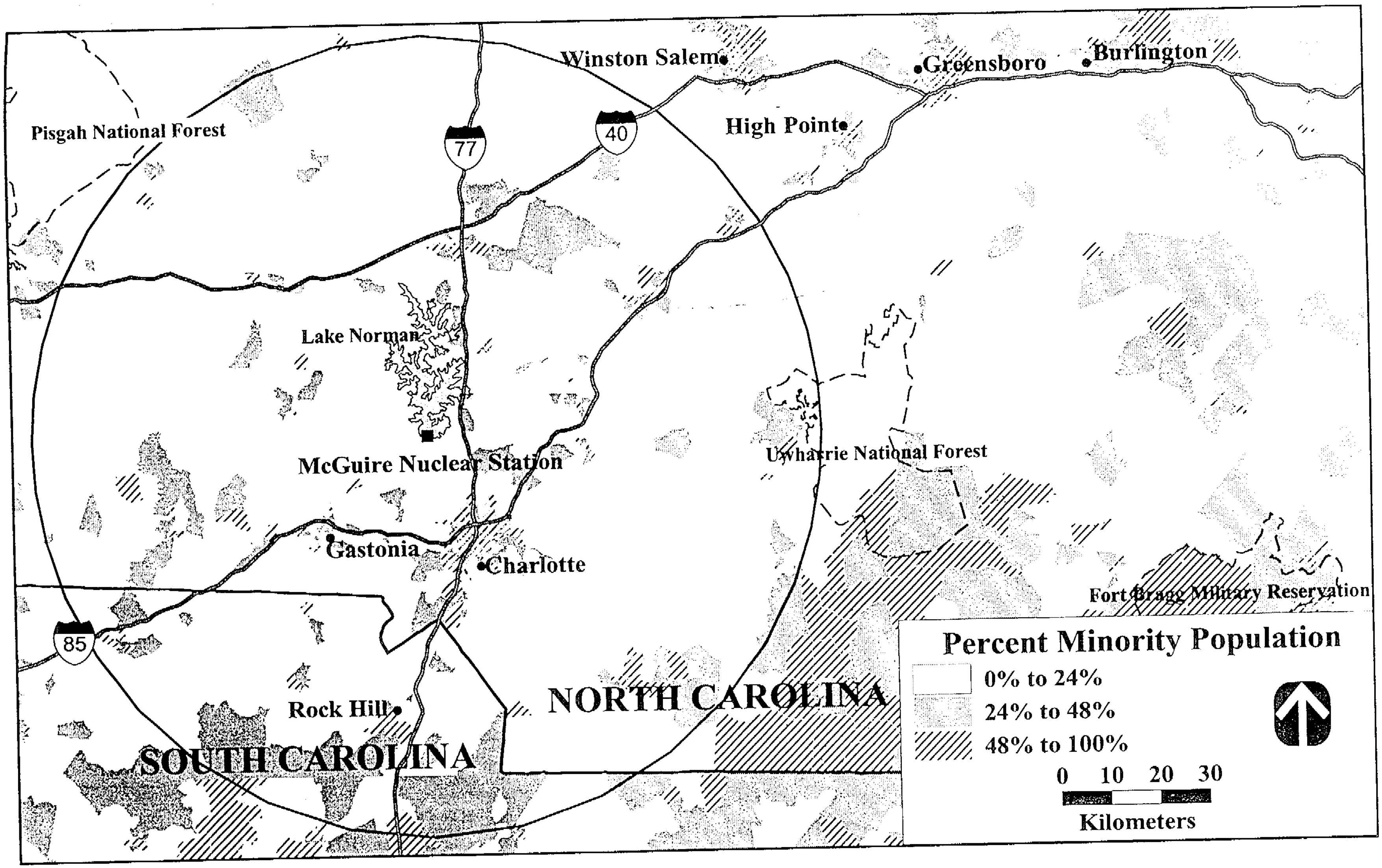

Figure M–12. Geographical Distribution of the Minority Population Residing Within 80 km (50 mi) of McGuire Nuclear Station

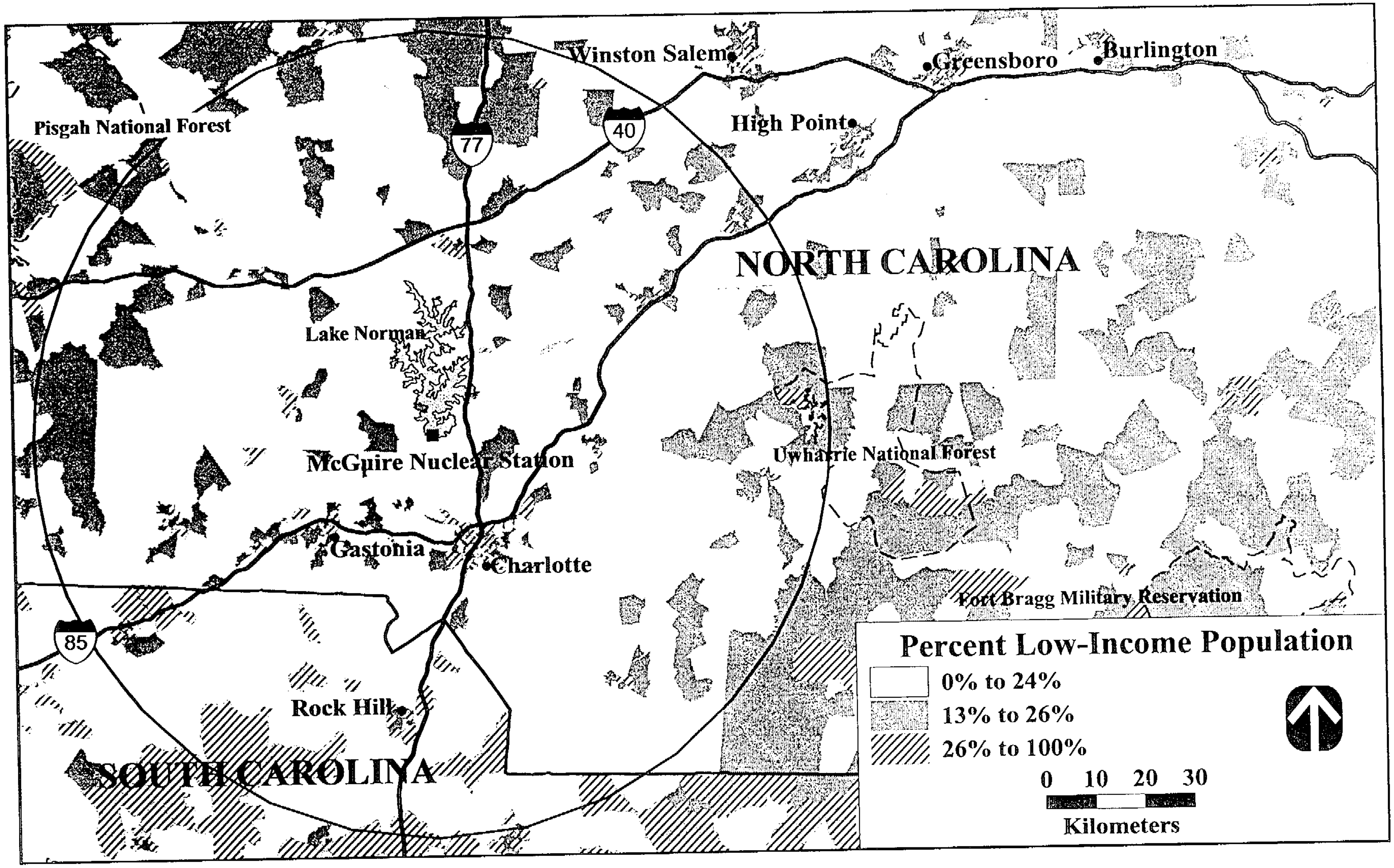

Figure M–13. Geographical Distribution of the Low-Income Population Residing Within 80 km (50 mi) of McGuire Nuclear Station

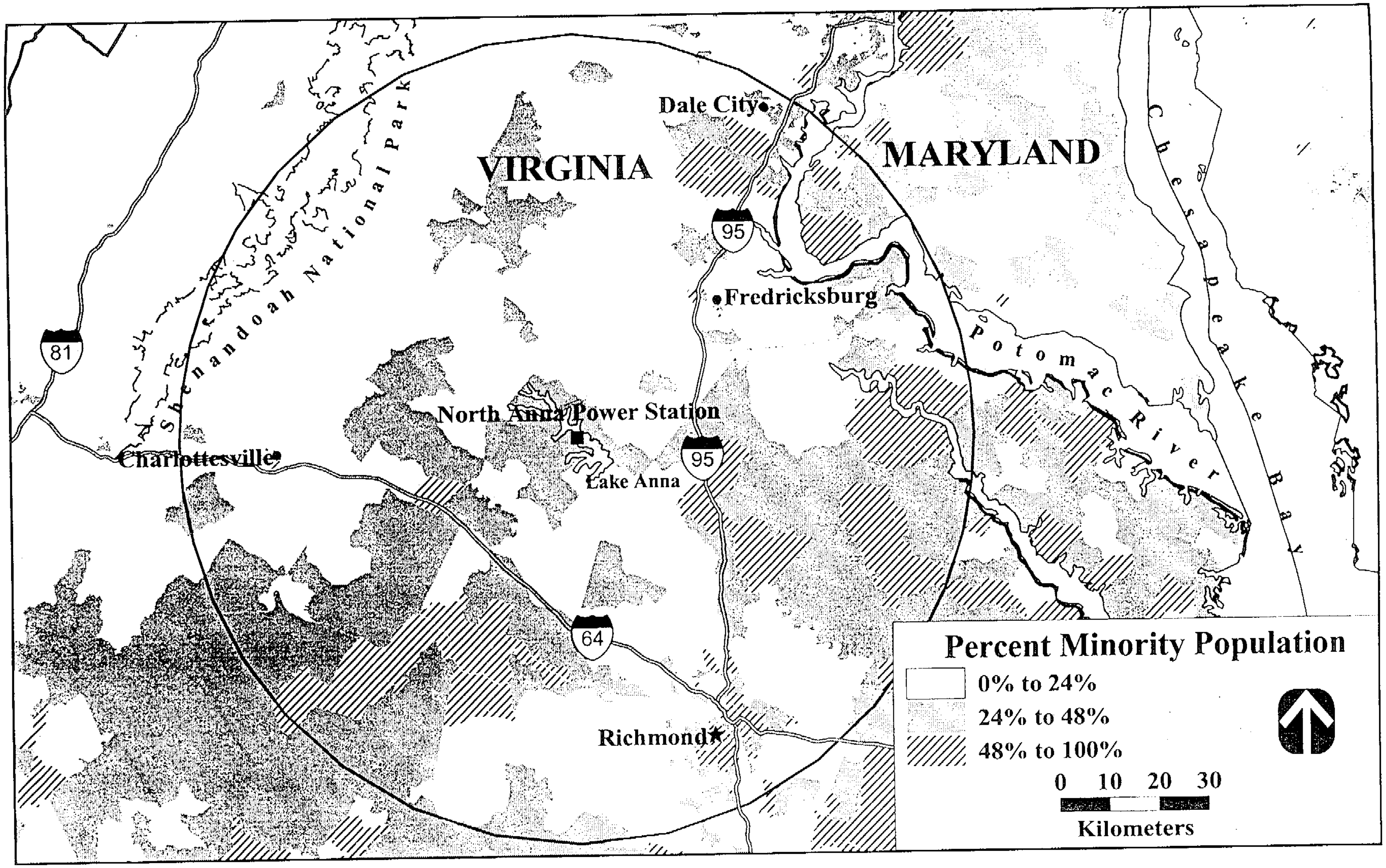

Figure M–14. Geographical Distribution of the Minority Population Residing Within 80 km (50 mi) of North Anna Power Station

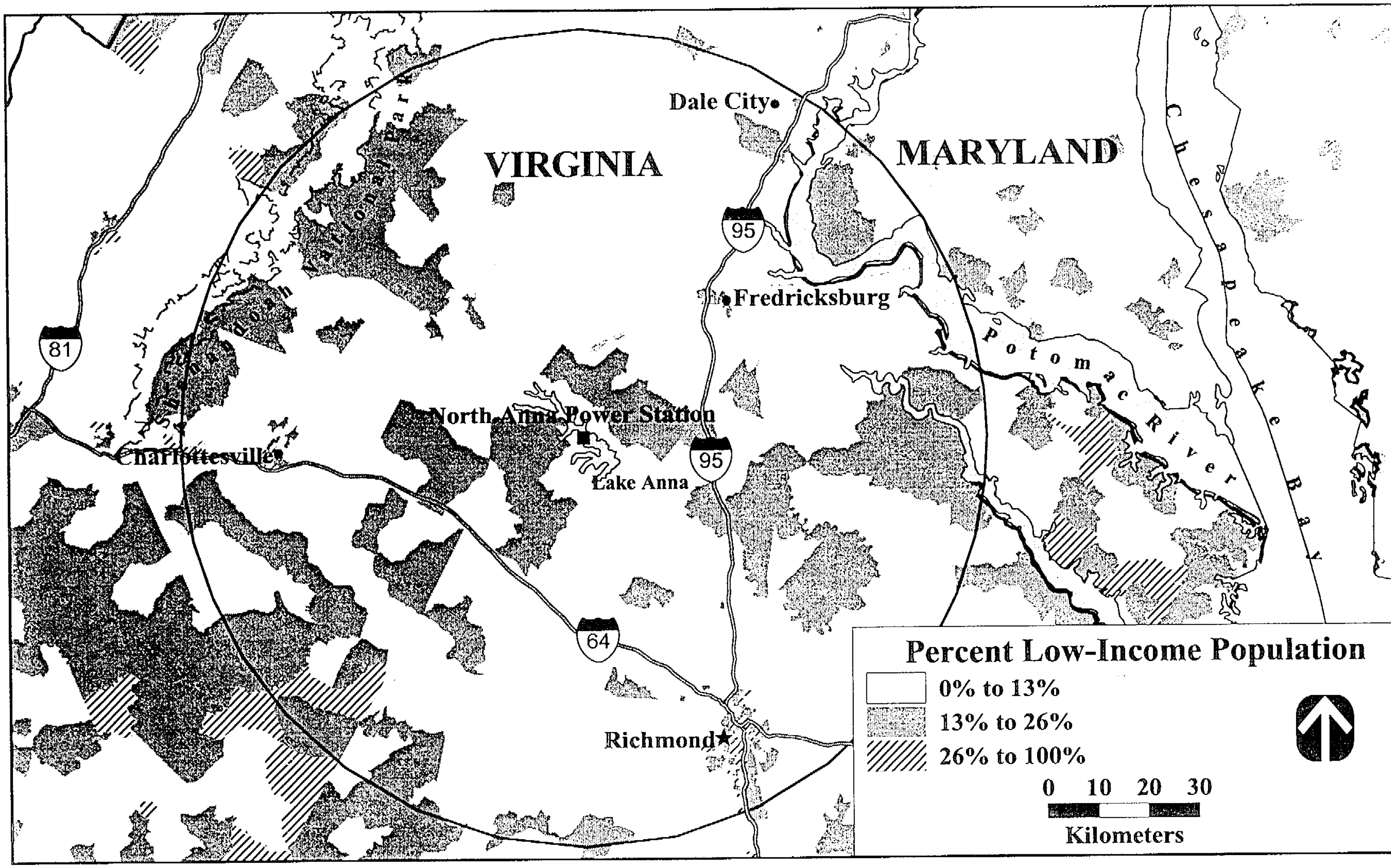

Figure M–15. Geographical Distribution of the Low-Income Population Residing Within 80 km (50 mi) of North Anna Power Station

Appendix O
Consultations

Certain statutes and regulations require the U.S. Department of Energy (DOE) to consider consultations with Federal, State, and local agencies and federally recognized Native American groups regarding the potential for alternatives for surplus plutonium disposition to disturb sensitive resources. These consultations are related to biotic, cultural, and Native American resources. DOE has initiated applicable consultations with Federal and State agencies and federally recognized Native American groups. Appendix O contains copies of the consultation letters sent by DOE to agencies and Native American groups, and any written responses provided by those agencies or groups. Attachments to responses are not included in Appendix O but are, nevertheless, part of the public record.

Department of Energy
Washington, DC 20585

October 30, 1998

David Hansen
State Historic Preservation Officer
Office of Archaeology & Historical Preservation
420 Golf Club Road SE, Suite 201
Lacey, Washington 98503

Subject: *Consultation for Surplus Plutonium Disposition Environmental Impact Analysis Process, Under Executive Memorandum Concerning Government-to-Government Relations*

Dear Mr. Hansen:

The purpose of this letter is to notify you that the United States Department of Energy (DOE) is in the process of conducting an Environmental Impact Analysis concerning the disposition of surplus plutonium.

With this letter we are soliciting specific concerns the Office of Archaeology and Historical Preservation may have about the proposal. This consultation is in accordance with the National Environmental Policy Act and Section 106 of the National Historic Preservation Act.

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. DOE is producing the SPD EIS in compliance with the National Environmental Policy Act (NEPA) and Council on Environmental Quality regulations implementing NEPA, DOE's NEPA Implementing Regulations (10 CFR 1021), and other applicable federal and state environmental legislation.

The purpose and need for the proposed action is to reduce the threat of nuclear weapons proliferation worldwide by disposing of surplus plutonium in the United States in an environmentally safe and timely manner. The SPD Draft EIS, a copy of which is attached for your review, examines the potential environmental impacts for 24 alternatives for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion; mixed oxide (MOX) fuel fabrication; and plutonium conversion and immobilization.

If an alternative is selected that includes siting of surplus plutonium disposition facilities at the Hanford site (e.g., Alternative 2), a maximum of about 15 hectares

David Hansen, Washington SHPO
10/30/98
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(37 acres) of land in the 400 Area would be impacted. No prehistoric or historic archaeological resources have been identified within the proposed construction areas, and no architectural resources in the 200 East of 400 Area. Preconstruction surveys (as required) and construction monitoring for previously unknown resources would be conducted within the framework of the *Hanford Cultural Resources Management Plan* (Battelle 1989; revised draft edition 1998).

If you have any specific concerns about the SPD EIS proposal, we would like to hear from you. Please contact me with your concerns or questions at:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
P.O. Box 23786
Washington, DC 20026-3786
(202) 586-0149.

You may also contact Dee Lloyd, Hanford Cultural Resources Program Manager, at (509) 372-2299.

Sincerely,

Marcus Jones
SPD EIS Document Manager

cc: Dee Lloyd, Cultural Resource Manager, Hanford
Lois Thompson, Federal Preservation Officer, DOE HQ

SPD EIS enclosure

Department of Energy
Washington, DC 20585

October 30, 1998

Mr. Russell Jim, Manager
Environmental Restoration/Waste Management Program
Confederated Tribes and Bands of the Yakama Indian Nation
2808 Main Street
Union Gap, Washington 98903

Subject: *Consultation for Surplus Plutonium Disposition Environmental Impact Analysis Process, Under Executive Memorandum Concerning Government-to-Government Relations*

Dear Mr. Jim:

The purpose of this letter is to notify you that the United States Department of Energy (DOE) is in the process of conducting an Environmental Impact Analysis concerning the disposition of surplus plutonium.

With this letter we are soliciting specific concerns the Confederated Tribes and Bands of the Yakama Indian Nation may have about the proposal. This consultation is in accordance with the Executive Memorandum (29 April 1994) entitled, "Government-to-Government Relations with Native American Tribal Governments", and DOE Order 1230.2. It also follows prior consultation initiated for compliance with the American Indian Religious Freedom Act (AIRFA) (PL 95-341) and the Native American Graves Protection and Repatriation Act (NAGPRA) (PL 101-601).

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. DOE is producing the SPD EIS in compliance with the National Environmental Policy Act (NEPA) and Council on Environmental Quality regulations implementing NEPA, DOE's NEPA Implementing Regulations (10 CFR 1021), and other applicable federal and state environmental legislation.

The purpose and need for the proposed action is to reduce the threat of nuclear weapons proliferation worldwide by disposing of surplus plutonium in the United States in an environmentally safe and timely manner. The SPD Draft EIS, a copy of which is attached for your review, examines the potential environmental impacts for 24 alternatives for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion; mixed oxide (MOX) fuel fabrication; and plutonium conversion and immobilization.

Mr. Russell Jim, Manager, Confederated Tribes and Bands of the Yakama Indian Nation
10/30/98
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If an alternative is selected that includes siting of surplus plutonium disposition facilities at the Hanford site (e.g., Alternative 2), a maximum of 15 hectares (37 acres) of land in previously disturbed portions of the 400 Area would be impacted. Based on previous investigations, no traditional cultural properties have been identified in the 400 Area or immediately adjacent areas.

If you have any specific concerns about the SPD EIS proposal, we would like to hear from you. Please contact me with your concerns or questions at:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
P.O. Box 23786
Washington, DC 20026-3786
(202) 586-0149.

You may also contact Kevin Clark, Hanford Indian Nation Program Manager, at (509) 376-6332.

Sincerely,

Marcus Jones
SPD EIS Document Manager

cc: Tom Woods, YIN
Nanci Peters, YIN
Kevin V. Clark, Indian Nation Program Manager, Hanford
Brandt Petrasek, EM-20, DOE HQ

SPD EIS enclosure

Department of Energy
Washington, DC 20585

October 30, 1998

Ms. Donna L. Powaukee, Director
Environmental Restoration/Waste Management Program
Nez Perce Tribe
P.O. Box 365
Lapwai, Idaho 83540

Subject: *Consultation for Surplus Plutonium Disposition Environmental Impact Analysis Process, Under Executive Memorandum Concerning Government-to-Government Relations*

Dear Ms. Powaukee:

The purpose of this letter is to notify you that the United States Department of Energy (DOE) is in the process of conducting an Environmental Impact Analysis concerning the disposition of surplus plutonium.

With this letter we are soliciting specific concerns the Nez Perce Tribe may have about the proposal. This consultation is in accordance with the Executive Memorandum (29 April 1994) entitled, "Government-to-Government Relations with Native American Tribal Governments", and DOE Order 1230.2. It also follows prior consultation initiated for compliance with the American Indian Religious Freedom Act (AIRFA) (PL 95-341) and the Native American Graves Protection and Repatriation Act (NAGPRA) (PL 101-601).

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. DOE is producing the SPD EIS in compliance with the National Environmental Policy Act (NEPA) and Council on Environmental Quality regulations implementing NEPA, DOE's NEPA Implementing Regulations (10 CFR 1021), and other applicable federal and state environmental legislation.

The purpose and need for the proposed action is to reduce the threat of nuclear weapons proliferation worldwide by disposing of surplus plutonium in the United States in an environmentally safe and timely manner. The SPD Draft EIS, a copy of which is attached for your review, examines the potential environmental impacts for 24 alternatives for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion; mixed oxide (MOX) fuel fabrication; and plutonium conversion and immobilization.

Ms. Donna L. Powaukee, Nez Perce Tribe
10/30/98
Page 2

If an alternative is selected that includes siting of surplus plutonium disposition facilities at the Hanford site (e.g., Alternative 2), a maximum of 15 hectares (37 acres) of land in previously disturbed portions of the 400 Area would be impacted. Based on previous investigations, no traditional cultural properties have been identified in the 400 Area or immediately adjacent areas.

If you have any specific concerns about the SPD EIS proposal, we would like to hear from you. Please contact me with your concerns or questions at:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
P.O. Box 23786
Washington, DC 20026-3786
(202) 586-0149.

You may also contact Kevin Clark, Hanford Indian Nation Program Manager, at (509) 376-6332.

Sincerely,

Marcus Jones
SPD EIS Document Manager

cc: Stan Sobczyk, NPT
Pat Sobotta, NPT
Kevin Clark, Indian Nations Program Manager, Hanford
Brandt Petrasek, EM-20, DOE HQ

SPD EIS enclosure

Department of Energy
Washington, DC 20585

October 30, 1998

Ms. Lenora Seelatsee
Wanapum Band
Grant County P.U.D
30 "C" Street, S.W.
P.O. Box 878
Ephrata, Washington 98823

Subject: *Consultation for Surplus Plutonium Disposition Environmental Impact Analysis Process, Under Executive Memorandum Concerning Government-to-Government Relations*

Dear Ms. Seelatsee:

The purpose of this letter is to notify you that the United States Department of Energy (DOE) is in the process of conducting an Environmental Impact Analysis concerning the disposition of surplus plutonium.

With this letter we are soliciting specific concerns the Wanapum Band may have about the proposal. This consultation is in accordance with the Executive Memorandum (29 April 1994) entitled, "Government-to-Government Relations with Native American Tribal Governments", and DOE Order 1230.2. It also follows prior consultation initiated for compliance with the American Indian Religious Freedom Act (AIRFA) (PL 95-341) and the Native American Graves Protection and Repatriation Act (NAGPRA) (PL 101-601).

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. DOE is producing the SPD EIS in compliance with the National Environmental Policy Act (NEPA) and Council on Environmental Quality regulations implementing NEPA, DOE's NEPA Implementing Regulations (10 CFR 1021), and other applicable federal and state environmental legislation.

The purpose and need for the proposed action is to reduce the threat of nuclear weapons proliferation worldwide by disposing of surplus plutonium in the United States in an environmentally safe and timely manner. The SPD Draft EIS, a copy of which is attached for your review, examines the potential environmental impacts for 24 alternatives for the proposed siting, construction, and operation of three types of

Ms. Lenora Seelatsee, Wanapum Band
10/30/98
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facilities: pit disassembly and conversion; mixed oxide (MOX) fuel fabrication; and plutonium conversion and immobilization.

If an alternative is selected that includes siting of surplus plutonium disposition facilities at the Hanford site (e.g., Alternative 2), a maximum of 15 hectares (37 acres) of land in previously disturbed portions of the 400 Area would be impacted. Based on previous investigations, no traditional cultural properties have been identified in the 400 Area or immediately adjacent areas.

If you have any specific concerns about the SPD EIS proposal, we would like to hear from you. Please contact me with your concerns or questions at:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
P.O. Box 23786
Washington, DC 20026-3786
(202) 586-0149.

You may also contact Kevin Clark, Hanford Indian Nation Program Manager, at (509) 376-6332.

Sincerely,

Marcus Jones
SPD EIS Document Manager

cc: Rex Buck, Jr., Wanapum
Robert Tomanawash, Wanapum
Kevin V. Clark, Indian Nation Program Manager, Hanford
Brandt Petrasek, EM-20, DOE HQ

SPD EIS enclosure

Department of Energy
Washington, DC 20585

October 30, 1998

Mr. J. R. Wilkinson, Manager
Special Sciences and Resources Program
Confederated Tribes of the Umatilla Indian Reservation
P.O. Box 638
Pendleton, Oregon 97801

Subject: *Consultation for Surplus Plutonium Disposition Environmental Impact Analysis Process, Under Executive Memorandum Concerning Government-to-Government Relations*

Dear Mr. Wilkinson:

The purpose of this letter is to notify you that the United States Department of Energy (DOE) is in the process of conducting an Environmental Impact Analysis concerning the disposition of surplus plutonium.

With this letter we are soliciting specific concerns the Confederated Tribes of the Umatilla Indian Reservation may have about the proposal. This consultation is in accordance with the Executive Memorandum (29 April 1994) entitled, "Government-to-Government Relations with Native American Tribal Governments", and DOE Order 1230.2. It also follows prior consultation initiated for compliance with the American Indian Religious Freedom Act (AIRFA) (PL 95-341) and the Native American Graves Protection and Repatriation Act (NAGPRA) (PL 101-601).

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. DOE is producing the SPD EIS in compliance with the National Environmental Policy Act (NEPA) and Council on Environmental Quality regulations implementing NEPA, DOE's NEPA Implementing Regulations (10 CFR 1021), and other applicable federal and state environmental legislation.

The purpose and need for the proposed action is to reduce the threat of nuclear weapons proliferation worldwide by disposing of surplus plutonium in the United States in an environmentally safe and timely manner. The SPD Draft EIS, a copy of which is attached for your review, examines the potential environmental impacts for 24 alternatives for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion; mixed oxide (MOX) fuel fabrication; and plutonium conversion and immobilization.

Mr. J. R. Wilkinson, Manager, Confederated Tribes of the Umatilla Reservation
10/30/98
Page 2

If an alternative is selected that includes siting of surplus plutonium disposition facilities at the Hanford site (e.g., Alternative 2), a maximum of 15 hectares (37 acres) of land in previously disturbed portions of the 400 Area would be impacted. Based on previous investigations, no traditional cultural properties have been identified in the 400 Area or immediately adjacent areas.

If you have any specific concerns about the SPD EIS proposal, we would like to hear from you. Please contact me with your concerns or questions at:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
P.O. Box 23786
Washington, DC 20026-3786
(202) 586-0149.

You may also contact Kevin Clark, Hanford Indian Nation Program Manager, at (509) 376-6332.

Sincerely,

Marcus Jones
SPD EIS Document Manager

cc: Jo Marie Tessman, CTUIR
Kevin V. Clark, Indian Nation Program Manager, Hanford
Brandt Petrasek, EM-20, DOE HQ

SPD EIS enclosure

Department of Energy
Washington, DC 20585
July 28, 1998

Mr. Richard Roy
U.S. Department of Interior
Fish and Wildlife Service
Post Office Box 1157
Moses Lake, WA 98837

Dear Mr. Roy:

INFORMAL CONSULTATION UNDER SECTION 7 OF THE ENDANGERED SPECIES ACT FOR SURPLUS PLUTONIUM DISPOSITION

The Department of Energy (DOE) published its Notice of Intent to prepare the *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) in the Federal Register (Vol. 92, No. 99) on May 22, 1997. This SPD EIS is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. To summarize, the purpose of the proposed action is to reduce the threat of nuclear weapons proliferation worldwide in an environmentally safe and timely manner by conducting disposition of surplus plutonium in the United States, thus setting a nonproliferation example for other nations.

The SPD Draft EIS, a copy of which is attached for your review, examines twenty-four alternatives and analyzes the potential environmental impacts for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion, mixed oxide (MOX) fuel fabrication, and plutonium conversion and immobilization. The Hanford Site near Richland, Washington is a candidate site for all three facilities. The candidate sites and alternatives are shown in Table 2-1 of the SPD Draft EIS. Please note that where practical, the modification of existing buildings is being considered.

Alternative 2 proposes locating pit disassembly and conversion, and plutonium conversion and immobilization facilities in the Fuels and Materials Examination Facility (FMEF) and the MOX fuel fabrication facility in new construction adjacent to FMEF in the 400 Area. In addition, the planned high-level waste vitrification facility in the 200 East Area would be used to process the canisters from the plutonium conversion and immobilization facility. Although several alternatives include locating facilities at Hanford, Alternative 2 has the greatest potential for impacts on ecological resources.

Preliminary analyses suggest that overall impacts on ecological resources from constructing and operating the proposed surplus plutonium disposition facilities would be limited because the land area required (15 hectares [37 acres]) is relatively small in comparison to regionally available habitat; habitat disturbance would be minimized because construction would take place in previously disturbed or developed areas; and operational impacts would be minimized because

facility releases of airborne and aqueous effluents would be controlled and permitted. Section 4.26.1.3 of the SPD Draft EIS presents the ecological resources analysis for the Hanford Site.

Although sources indicate that no critical habitat for any threatened and endangered species exists near the proposed construction area, there may be Washington State-classified special status species associated with shrub-steppe habitat that could be affected due to land disturbance and noise. Animal species include burrowing owl, ferruginous hawk, golden eagle, long-billed curlew, sage thrasher, Swainson's hawk, pygmy rabbit, desert night snake, and striped whipsnake. It is doubtful the loggerhead shrike and sage sparrow would be affected because a fire in the 400 Area previously destroyed most of their habitat. Plant species include crouching milkvetch, piper's daisy, squill onion, and stalked-pod milkvetch.

Consistent with the Endangered Species Act, DOE requests that the Fish and Wildlife Service provide any additional information on the presence of threatened and endangered animal and plant species, both listed and proposed, in the vicinity of the 200 East and 400 Areas at Hanford. Information on the habitats of these species would also be appreciated. DOE also requests information on any other species of concern that are known to occur or potentially occur in the vicinity of the 200 East and 400 Areas.

As part of DOE's National Environmental Policy Act process, DOE encourages the Fish and Wildlife Service to identify any concerns or issues that it believes should be addressed in the SPD EIS. To facilitate incorporation of your input into the SPD Final EIS, please provide a written response by September 16, 1998.

Please mail your response to:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
1000 Independence Avenue, SW
Washington, DC 20585

If you have any questions, please contact me at (202) 586-0149.

Sincerely,

Marcus Jones

Marcus Jones
SPD EIS Document Manager

cc: Charles A. Brandt, PNNL
Dana Ward, DOE

United States Department of the Interior

FISH AND WILDLIFE SERVICE
517 South Buchanan
Moses Lake, Washington 98837
Phone: 509-765-6125 FAX: 509-765-9043

December 3, 1998

Department of Energy
Office of Fissile Materials Disposition
Attn: Marcus Jones
SPD EIS Document Manager
1000 Independence Avenue, SW
Washington, DC 20585

RE: Surplus Plutonium Disposition Environmental Impact Statement
FWS Reference: 1-9-99-SP-052

Dear Mr. Jones:

Thank you for your request of December 3, 1998. Enclosed is a list of threatened and endangered species, candidate species and species of concern (Enclosure A), that may be present at the Hanford Reservation. We are enclosing a list of the whole site, due to the limited site-specific information provided in your December 3, 1998 letter. This list fulfills the requirements of the U. S. Fish and Wildlife Service (Service) under Section 7(c) of the Endangered Species Act of 1973, as amended (Act).

The Service has included aquatic species due to the possibilities of groundwater transmission of radioactive materials. Thus, we are giving you the opportunity to make an initial evaluation of possible effects to each species, as provided in the Federal Register (Vol. 51, No. 106, pg. 19946) on June 3, 1986. We are enclosing a copy of the requirements for federal agency compliance under the Act (Enclosure B).

Should the biological assessment for the proposed project determine that a listed species is likely to be affected (adversely or beneficially) by the project, the federal agency should request Section 7 consultation through this office. If the biological assessment determines that the proposed action is "not likely to adversely affect" a listed species, the federal agency should request Service concurrence with that determination through the informal consultation process. If the biological assessment determines the project to have "no effect," we would appreciate receiving a copy for our information.

Candidate species and species of concern are included simply as advance notice to federal agencies of species which may be proposed and listed in the future. Protection provided to these species now may preclude possible listing in the future. If early evaluation of your project indicates that it is likely to adversely impact a candidate species, or species of concern, the federal agency may wish to request technical assistance from this office.

There are other species, including anadromous fishes that have been federally listed by the National Marine Fisheries Service (NMFS). Some of these species may occur in the vicinity of your project. Please contact NMFS in Lacy, WA at (360) 753-5828, or in Portland, OR at (503) 231-2319, to request a species list.

Thank you for your efforts to protect our nation's species and their habitats. If you have additional questions regarding your responsibilities under the Act, please contact Richard Smith of this office at (509) 765-6125.

Sincerely,

Donald C. Haley

Richard Roy
Acting Assistant Field Supervisor

ENCLOSURES

Department of Energy
Washington, DC 20585
July 28, 1998

Mr. Jay McConnaughey
Washington Department of Fish and Wildlife
1315 West 4th
Kennewick, WA 99336

Dear Mr. McConnaughey:

The Department of Energy (DOE) published its Notice of Intent to prepare the *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) in the Federal Register (Vol. 92, No. 99) on May 22, 1997. This SPD EIS is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. To summarize, the purpose of the proposed action is to reduce the threat of nuclear weapons proliferation worldwide in an environmentally safe and timely manner by conducting disposition of surplus plutonium in the United States, thus setting a nonproliferation example for other nations.

The SPD Draft EIS, a copy of which is attached for your review, examines twenty-four alternatives and analyzes the potential environmental impacts for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion, mixed oxide (MOX) fuel fabrication, and plutonium conversion and immobilization. The Hanford Site near Richland, Washington is a candidate site for all three facilities. The candidate sites and alternatives are shown in Table 2-1 of the SPD Draft EIS. Please note that where practical, the modification of existing buildings is being considered.

Alternative 2 proposes locating pit disassembly and conversion, and plutonium conversion and immobilization facilities in the Fuels and Materials Examination Facility (FMEF) and the MOX fuel fabrication facility in new construction adjacent to FMEF in the 400 Area. In addition, the planned high-level waste vitrification facility in the 200 East Area would be used to process the canisters from the plutonium conversion and immobilization facility. Although several alternatives include locating facilities at Hanford, Alternative 2 has the greatest potential for impacts on ecological resources.

Preliminary analyses suggest that overall impacts on ecological resources from constructing and operating the proposed surplus plutonium disposition facilities would be limited because the land area required (15 hectares [37 acres]) is relatively small in comparison to regionally available habitat; habitat disturbance would be minimized because construction would take place in previously disturbed or developed areas; and operational impacts would be minimized because

facility releases of airborne and aqueous effluents would be controlled and permitted. Section 4.26.1.3 of the SPD Draft EIS presents the ecological resources analysis for the Hanford Site.

Although sources indicate that no critical habitat for any threatened and endangered species exists near the proposed construction area, there may be Washington State-classified special status species associated with shrub-steppe habitat that could be affected due to land disturbance and noise. Animal species include burrowing owl, ferruginous hawk, golden eagle, long-billed curlew, sage thrasher, Swainson's hawk, pygmy rabbit, desert night snake, and striped whipsnake. It is doubtful the loggerhead shrike and sage sparrow would be affected because a fire in the 400 Area previously destroyed most of their habitat. Plant species include crouching milkvetch, piper's daisy, squill onion, and stalked-pod milkvetch.

As part of DOE's National Environmental Policy Act process, DOE encourages the Washington Department of Fish and Wildlife to identify any concerns or issues that it believes should be addressed in the SPD EIS. To facilitate incorporation of your input into the SPD Final EIS, please provide a written response by September 16, 1998.

Please mail your response to:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
1000 Independence Avenue, SW
Washington, DC 20585

If you have any questions, please contact me at (202) 586-0149.

Sincerely,

Marcus Jones

Marcus Jones
SPD EIS Document Manager

cc: Charles A. Brandt, PNNL
Dana Ward, DOE

STATE OF WASHINGTON
DEPARTMENT OF FISH AND WILDLIFE
1701 S 24th Avenue • Yakima, Washington 98902-5720 • (509) 575-2740 FAX (509) 575-2474
c/o Department of Ecology
1315 W 4th Ave, Kennewick, WA 99336

7 December, 1998

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
1000 Independence Ave. SW
Washington, DC 20585

Dear Mr. Jones:

Subject: Comments on the *Surplus Plutonium Disposition Draft Environmental Impact Statement, July 1998*, DOE/EIS-0283-D.

Upon a recent request for comments on the aforementioned document by U.S. Department of Energy (USDOE) Washington DC staff, the Washington Department of Fish and Wildlife (WDFW) is providing comments and greatly appreciates the invitation to submit comments even after the official closing of the comment period.

The WDFW supports the identified preferred alternatives in the draft EIS for siting plutonium disposition facilities (i.e. Immobilization at SRS, MOX Fuel Fabrication at SRS and Pit Disassembly and Conversion at SRS or Pantex). We concur with USDOE's determination as stated in the *Summary* "that Hanford's cleanup mission is critical, therefore … prefers that the cleanup mission remain the site's top priority…" It is important that cleanup continue to remain the focus of the Hanford Site to be protective of the Columbia River ecosystem.

The Hanford Site ecosystem contains biological resources of regional, national, and international significance. The Hanford Reach supports a healthy stock of upriver bright fall chinook salmon (*Oncorhynchus tshawytscha*) and provides essential habitat for the federally listed Upper Columbia River steelhead (*Oncorhynchus mykiss*) which has been listed as endangered. The Nature Conservancy of Washington findings from a multi-year biodiversity inventory confirm the importance of the Hanford Site, and the 1997 annual report states "Findings from the biodiversity inventory to date show that the Hanford Site,

Mr. Jones
7 December, 1998
Page 2 of 3

including the Hanford Reach, is home to an irreplaceable natural legacy[1]." Over the duration of the inventory, TNC scientists discovered 40 species new to science. Other biological studies support the significance of these resources as well. The significance of shrub steppe is accurately reflected in the *draft Hanford Site Biological Resource Management Plan* by the following: "...the percentage that Hanford contributes to the existence of shrub steppe within the ecoregion has increased by about 250% since European settlement". The WDFW has designated nearly 80% of the site as Priority Shrub Steppe Habitat including the post-fire habitat. Finally, the National Biological Service (currently known as the National Biological Division of the U.S. Geological Service) has listed native shrub and grassland steppe in Washington and Oregon as an endangered ecosystem[2].

The Hanford Site has been identified in several alternatives with alternative 2 having the greatest potential for impacts on ecological resources. Impacts would include the loss of 37 acres of habitat and effluent discharge to the Columbia River. The WDFW provides the following comments in the event that the facilities are actually sited at the Hanford Site.

The draft EIS mentions that effluent discharges would occur to the Columbia River. Given this information, the USDOE should enter into consultation with the National Marine Fisheries Service under Section 7 of the Endangered Species Act to ensure that the action is not likely to jeopardize the continued existence of any listed species (16 U.S.C. Sec.1536 (a)(2)) (i.e. Upper Columbia River steelhead). Consultation requirements of Section 7 are nondiscretionary and are effective at the time of species' listing regardless of whether critical habitat is designated. Our concerns are with the release of contaminants and thermal discharge that may adversely affect anadromous fish. Again, as in our comments on DOE/EA-1259, we would expect an aquatic biological review to occur given the evidence that suggest Upper Columbia River steelhead spawn where fall chinook salmon have been previously observed spawning in the Hanford Reach.

We commend USDOE for first looking at the modification of existing buildings before constructing new ones. This action is consistent with the mitigation hierarchy as defined in 40CFR§1508.20. As stated earlier, WDFW designated post-fire shrub steppe habitat located in the southeast portion of the Hanford Site as Priority Shrub Steppe Habitat. Our concerns with this habitat are captured in a letter dated 1 July, 1998 to Mr. Dana Ward,

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[1] The Nature Conservancy of Washington. Biodiversity Inventory and analysis of the Hanford Site, 1997 Annual report, May 1998.
[2] Noss, Reed F., E.T. Laroe III, and J.M. Scott. Endangered ecosystems of the United States: A preliminary assessment of loss and degradation. Biological Report 28, Feb. 1995, National Biological Service, U.S. Department of the Interior.

Mr. Jones
7 December, 1998
Page 3 of 3

USDOE-RL. We believe every effort should be made to protect this habitat from further fragmentation and degradation which would occur from habitat disturbances, and that any adverse impacts that could not be mitigated through minimization and rectification should be compensated for at a 3:1 ratio. This would be consistent with USDOE's steward role of sustaining the natural ecosystems as stated in the Land and Facility Use Policy. Also, a commitment to fully mitigate adverse impacts to Priority Shrub Steppe Habitat would be consistent with past actions, such as, the Safe Interim Storage EIS, Tank Waste Remediation System EIS, and Solid Waste Retrieval Complex, Enhanced Radioactive and Mixed Waste Storage Facility, Infrastructure Upgrades, and Central Waste, Support Complex EA where adverse impacts were compensated.

We would request language be included in the final EIS that states "The project will be reviewed with the Washington Department of Fish and Wildlife and a mitigation action plan be developed and implemented to compensate for the destruction of Priority Shrub Steppe habitat from this project".

Again, thank you for the opportunity to comment. If you have any questions on these comments, please contact me at (509) 736-3095.

Sincerely,

Jay McConnaughey
Habitat Biologist, Hanford Site

Enclosures (2)

cc w/o enc:
USDOE
  Paul Dunigan. Jr.
Washington Department of Ecology
  Rebecca Inman
  Ron Skinnarland
WDFW
  Ted Clausing
  Neil Rikard

Department of Energy
Washington, DC 20585

October 30, 1998

Robert Yohe
State Historic Preservation Officer
100 Main
Boise, Idaho 83702

Subject: *Consultation for Surplus Plutonium Disposition Environmental Impact Analysis Process*

Dear Mr Yohe:

The purpose of this letter is to notify you that the United States Department of Energy (DOE) is in the process of conducting an Environmental Impact Analysis concerning the disposition of surplus plutonium.

With this letter we are soliciting specific concerns the Idaho State Historic Preservation Office may have about the proposal. This consultation is in accordance with National Environmental Policy Act and Section 106 of the National Historic Preservation Act.

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. DOE is producing the SPD EIS in compliance with the National Environmental Policy Act (NEPA) and Council on Environmental Quality regulations implementing NEPA, DOE's NEPA Implementing Regulations (10 CFR 1021), and other applicable federal and state environmental legislation.

The purpose and need for the proposed action is to reduce the threat of nuclear weapons proliferation worldwide by disposing of surplus plutonium in the United States in an environmentally safe and timely manner. The SPD Draft EIS, a copy of which is attached for your review, examines the potential environmental impacts for 24 alternatives for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion; mixed oxide (MOX) fuel fabrication; and plutonium conversion and immobilization.

If an alternative is selected that includes siting of surplus plutonium disposition facilities at the Idaho National Environmental and Engineering Laboratory (INEEL) site (e.g., Alternative 7A), a maximum of about 13 hectares (32 acres) of land inside the Idaho Nuclear Technology and Engineering Center (INTEC) protected area adjacent to

Robert Yohe, State Historic Preservation Officer
10/30/98
Page 2

the Fuel Processing Facility (FPF) would be impacted. Six prehistoric resources within the vicinity of the proposed construction area have been identified, but none are eligible for nomination to the National Register. A homestead and a trash dump may be eligible for the National Register, and a historic building survey being conducted within INTEC is likely to identify structures potentially eligible for the National Register based on their Cold War associations. Direct impact of the proposed construction would be unlikely; however, consistent with the *INEL Management Plan for Cultural Resources*, surveys and monitoring would be conducted to ensure against impact to National Register-eligible resources.

If you have any specific concerns about the SPD EIS proposal, we would like to hear from you. Please contact me with your concerns or questions at:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
P.O. Box 23786
Washington, DC 20026-3786
(202) 586-0149.

You may also contact Bob Stark, the INEEL Technical Lead for Cultural Resources, at (208) 526-1122.

Sincerely,

Marcus Jones
SPD EIS Document Manager

cc: Bob Stark, Technical Lead for Cultural Resources, INEEL
Lois Thompson, Federal Preservation Officer, DOE HQ

SPD EIS enclosure

Department of Energy
Washington, DC 20585

October 30, 1998

Mr. Keith Tinno, Tribal Chairman
Fort Hall Reservation
P.O. Box 306
Fort Hall, Idaho 83203

Subject: *Consultation for Surplus Plutonium Disposition Environmental Impact Analysis Process, Under Executive Memorandum Concerning Government-to-Government Relations*

Dear Mr. Tinno:

The purpose of this letter is to notify you that the United States Department of Energy (DOE) is in the process of conducting an Environmental Impact Analysis concerning the disposition of surplus plutonium.

With this letter we are soliciting specific concerns the Shoshone and Bannock Tribes may have about the proposal. This consultation is in accordance with the Executive Memorandum (29 April 1994) entitled, "Government-to-Government Relations with Native American Tribal Governments", and DOE Order 1230.2. It also follows prior consultation initiated for compliance with the American Indian Religious Freedom Act (AIRFA) (PL 95-341) and the Native American Graves Protection and Repatriation Act (NAGPRA) (PL 101-601).

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. DOE is producing the SPD EIS in compliance with the National Environmental Policy Act (NEPA) and Council on Environmental Quality regulations implementing NEPA, DOE's NEPA Implementing Regulations (10 CFR 1021), and other applicable federal and state-delegated environmental legislation.

The purpose and need for the proposed action is to reduce the threat of nuclear weapons proliferation worldwide by disposing of surplus plutonium in the United States in an environmentally safe and timely manner. The SPD Draft EIS, a copy of which is attached for your review, examines the potential environmental impacts for 24 alternatives for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion; mixed oxide (MOX) fuel fabrication; and plutonium conversion and immobilization.

Mr. Keith Tinno, Tribal Chairman, Fort Hall Reservation
10/30/98
Page 2

If an alternative is selected that includes siting of surplus plutonium disposition facilities at the INEEL site (e.g., Alternative 7A), a maximum of about 13 hectares (32 acres) of land inside the Idaho nuclear Technology and Engineering Center (INTEC) protected area adjacent to the Fuel Processing Facility (FPF) would be impacted. Specific Native American resources have not been identified within the proposed construction area, but operations could result in indirect impacts, such as access restrictions. DOE would conduct direct consultation with the Shoshone and Bannock Tribes, consistent with a working agreement between DOE and the tribes, to ensure there are no direct construction-related impacts.

If you have any specific concerns about the SPD EIS proposal, we would like to hear from you. Please contact me with your concerns or questions at:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
P.O. Box 23786
Washington, DC 20026-3786
(202) 586-0149.

You may also contact Bob Pence, the INEEL American Indian Program Manager, at (208) 526-6518.

Sincerely,

Marcus Jones
SPD EIS Document Manager

cc: Diana Yupe, Fort Hall
Bob Pence, American Indian Program Manager, INEEL
Brandt Petrasek, EM-20, DOE HQ

SPD EIS enclosure

Department of Energy
Washington, DC 20585
July 28, 1998

Ms. Susan Burch
U.S. Department of Interior
Fish and Wildlife Service
Snake River Basin Office
Columbia River Basin Ecological Region
1387 South Vinnell Way
Room 368
Boise, ID 83709

Dear Ms. Burch:

INFORMAL CONSULTATION UNDER SECTION 7 OF THE ENDANGERED SPECIES ACT FOR SURPLUS PLUTONIUM DISPOSITION

The Department of Energy (DOE) published its Notice of Intent to prepare the *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) in the Federal Register (Vol. 92, No. 99) on May 22, 1997. This SPD EIS is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. To summarize, the purpose of the proposed action is to reduce the threat of nuclear weapons proliferation worldwide in an environmentally safe and timely manner by conducting disposition of surplus plutonium in the United States, thus setting a nonproliferation example for other nations.

The SPD Draft EIS, a copy of which is attached for your review, examines twenty-four alternatives and analyzes the potential environmental impacts for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion, mixed oxide (MOX) fuel fabrication, and plutonium conversion and immobilization. The Idaho National Engineering and Environmental Laboratory (INEEL) near Idaho Falls, Idaho is a candidate site for the pit disassembly and MOX facilities. Alternatives 7A, 7B, and 8 propose locating pit disassembly and conversion in the Fuel Processing Facility (FPF) and MOX fuel fabrication in new construction in the Idaho Nuclear Technology and Energy Center (INTEC) area. The candidate sites and alternatives are shown in Table 2-1 of the SPD Draft EIS. Please note that where practical, the modification of existing buildings is being considered.

Preliminary analyses suggest that overall impacts on ecological resources from constructing and operating the proposed surplus plutonium disposition facilities would be limited because the land area required (13 hectares [32 acres]) is relatively small in comparison to regionally available habitat; habitat disturbance would be minimized because construction would take place in previously disturbed or developed areas; and operational impacts would be minimized because facility releases of airborne and aqueous effluents would be controlled and permitted. Section

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4.26.2.3 of the SPD Draft EIS presents the ecological resources analysis for INEEL.

Although sources indicate that no critical habitat for any threatened and endangered species exists near the proposed construction area, there may be Federal or State-classified special status species in the area surrounding INTEC. These species include bald eagle, black tern, burrowing owl, ferruginous hawk, loggerhead shrike, long-eared and small-footed myotis, northern goshawk, northern sagebrush lizard, peregrine falcon, pygmy rabbit, Townsend's western big-eared bat, trumpeter swan, and white-faced ibis. Noise disturbance is probably the most important impact affecting local wildlife populations.

Consistent with the Endangered Species Act, DOE requests that the Fish and Wildlife Service provide any additional information on the presence of threatened and endangered animal and plant species, both listed and proposed, in the vicinity of the INTEC area at INEEL. Information on the habitats of these species would also be appreciated. DOE also requests information on any other species of concern that are known to occur or potentially occur in the vicinity of INTEC.

As part of DOE's National Environmental Policy Act process, DOE encourages the Fish and Wildlife Service to identify any concerns or issues it believes should be addressed in the SPD EIS. To facilitate incorporation of your input into the SPD Final EIS, please provide a written response by September 16, 1998.

Please mail your response to:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
1000 Independence Avenue, SW
Washington, DC 20585

If you have any questions, please contact me at (202) 586-0149.

Sincerely,

Marcus Jones

Marcus Jones
SPD EIS Document Manager

cc: Roger Twitchell, DOE
Tim Reynolds, ESRF

United States Department of the Interior

FISH AND WILDLIFE SERVICE

Snake River Basin Office, Columbia River Basin Ecoregion
1387 South Vinnell Way, Room 368
Boise, Idaho 83709

August 18, 1998

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
1000 Independence Avenue S.W.
Washington, D.C. 20585

Subject: Surplus Plutonium Disposition–Section 7 Consultation
File #506.0000 SP #1-4-98-SP-247

Dear Mr. Jones:

The U.S. Fish and Wildlife Service (Service) has received your letter announcing your Notice of Intent to prepare the Surplus Plutonium Disposition Environmental Impact Statement. Your letter to us, dated July 28 1998 and received here August 10, 1998 dealt specifically with issues related to species listed under the Endangered Species Act of 1973 (Act). Your letter noted a number of rare and sensitive species that could occur at the Idaho National Engineering and Environmental Laboratory site. Two listed species, the threatened bald eagle and peregrine falcon, are included on your list. The Service concurs that the list you developed is accurate, and we are providing you a reference number to document our concurrence with your list (SP #1-4-98-SP-247).

At this time, staffing and funding constraints will preclude our direct involvement with your analysis of this project. As you know, Idaho Department of Fish and Game's Conservation Data Center is the repository for information about status and distribution of species of concern, including those listed under the Act. We encourage you to work with them to obtain the most current information about the species that may occur at the site. If you determine that a listed species may be affected by the project, Section 7 of the Act requires that you consult with the Service. In that event, we will be available for informal consultation.

Thank you for providing the Service with the opportunity to comment on the proposed project. Contact Alison Beck Haas of my staff in Boise (208) 378-5384 or Mike Donahoo in Pocatello (208) 233-8550 if you have questions.

Sincerely,

Robert B. Ruesink

Supervisor, Snake River Basin Office

cc: FWS-CBE, Portland (Diggs)
FWS, Pocatello (Donahoo)

Department of Energy
Washington, DC 20585
July 28, 1998

Mr. George Stephens
Idaho Department of Fish and Game
Conservation Data Center
600 South Walnut
Boise, ID 83705

Dear Mr. Stephens:

The Department of Energy (DOE) published its Notice of Intent to prepare the *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) in the Federal Register (Vol. 92, No. 99) on May 22, 1997. This SPD EIS is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. To summarize, the purpose of the proposed action is to reduce the threat of nuclear weapons proliferation worldwide in an environmentally safe and timely manner by conducting disposition of surplus plutonium in the United States, thus setting a nonproliferation example for other nations.

The SPD Draft EIS, a copy of which is attached for your review, examines twenty-four alternatives and analyzes the potential environmental impacts for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion, mixed oxide (MOX) fuel fabrication, and plutonium conversion and immobilization. The Idaho National Engineering and Environmental Laboratory (INEEL) near Idaho Falls, Idaho is a candidate site for the pit disassembly and MOX facilities. Alternatives 7A, 7B, and 8 propose locating pit disassembly and conversion in the Fuel Processing Facility (FPF) and MOX fuel fabrication in new construction in the Idaho Nuclear Technology and Energy Center (INTEC) area. The candidate sites and alternatives are shown in Table 2-1 of the SPD Draft EIS. Please note that where practical, the modification of existing buildings is being considered.

Preliminary analyses suggest that overall impacts on ecological resources from constructing and operating the proposed surplus plutonium disposition facilities would be limited because the land area required (13 hectares [32 acres]) is relatively small in comparison to regionally available habitat; habitat disturbance would be minimized because construction would take place in previously disturbed or developed areas; and operational impacts would be minimized because facility releases of airborne and aqueous effluents would be controlled and permitted. Section 4.26.2.3 of the SPD Draft EIS presents the ecological resources analysis for INEEL.

Although sources indicate that no critical habitat for any threatened and endangered species exists near the proposed construction area, there may be Federal or State-classified special status species in the area surrounding INTEC. These species include bald eagle, black tern, burrowing owl, ferruginous hawk, loggerhead shrike, long-eared and small-footed myotis, northern goshawk,

northern sagebrush lizard, peregrine falcon, pygmy rabbit, Townsend's western big-eared bat, trumpeter swan, and white-faced ibis. Noise disturbance is probably the most important impact affecting local wildlife populations.

As part of DOE's National Environmental Policy Act process, DOE encourages the Idaho Department of Fish and Game to identify any concerns or issues it believes should be addressed in the SPD EIS. To facilitate incorporation of your input into the SPD Final EIS, please provide a written response by September 16, 1998.

Please mail your response to:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
1000 Independence Avenue, SW
Washington, DC 20585

If you have any questions, please contact me at (202) 586-0149.

Sincerely,

Marcus Jones

Marcus Jones
SPD EIS Document Manager

cc: Roger Twitchell, DOE
Tim Reynolds, ESRF

IDAHO CONSERVATION DATA CENTER
Idaho Department of Fish and Game • 600 South Walnut • P.O. Box 25 Boise, Idaho 83707 • (208) 334-3402 • FAX 334-2114

12 August 1998

Marcus Jones, SPD EIS Document Manager
Department of Energy
Washington, D. C. 20585

Dear Mr. Jones:

I am responding to your request for input relative to special status species associated with INEEL and construction at the Idaho Nuclear Technology and Energy Center (INTEC). Enclosed is a list of special status plants and animals known to occur at INEEL. These represent species for which the Conservation Data Center (CDC) has documentation of occurrence.

Within a 10-mile radius of INTEC, the only occurrences in the CDC database are ferruginous hawk nesting territories and Merriam's shrew capture sites. In the eastern part of Idaho, gray wolf is considered an experimental, nonessential population. With regard to the species listed in your letter, the Lower Snake River Basin office of the U. S. Fish and Wildlife Service does not consider northern sagebrush lizard to be a Species of Concern.

If you have questions regarding this response, please contact me.

Sincerely,

George Stephens

George Stephens
Fish and Game Data Coordinator

IDAHO CONSERVATION DATA CENTER

Idaho Department of Fish and Game • 600 South Walnut • P.O. Box 25, Boise, Idaho 83707 • (208) 334-3402 • FAX 334-2114

gstephen@idfg.state.id.us http://www.state.id.us/fishgame/cdchome.htm

MEMORANDUM

TO: Kevin Folk

FROM: George Stephens

DATE: 12 February 1999

RE: INTEC area at INEEL

I am responding to your phone call this morning. After reviewing the original request (28 Jul 1998, from Marcus Jones) and looking at my response (12 Aug 1998), I can provide an update to our phone conversation.

Jones' request was not clear. His letter refers to the INTEC "area," to multiple sites on INEEL, and to Idaho Fish and Game addressing any concerns it has with the EIS. With regard to special status species, I think my response to Jones' letter is in tune with his request. In the body of my (1998) letter, I addressed (1) the two known species occurrences in the INTEC "area" and (2) the known occurrences on the entirety of INEEL with regard to the multiple sites. If you check the species list accompanying my letter, you will note INEEL is indicated (at the top) of the list.

On the phone, I explained the basis for conducting a database search of a 10-mile radius around a project area. Primarily, it is to check whether a peregrine falcon eyrie or hack site is known from the area. That 10-mile guideline came from the U. S. Fish and Wildlife Service for the CDC to use when developing a Sec. 7 (ESA) species list. Many other species don't have well-defined guidelines, and I simply included other known occurrences found within the 10-mile radius. Animals generally tend to move around and are often found over a larger area than where an individual was observed or trapped.

The pages accompanying this memorandum contain printed database records for the known occurrences in the INTEC area. In addition to these species, pygmy rabbit should be considered as a probable occurrence in any area of big sagebrush habitat. The printout contains a rare plant not addressed in the 1998 response. The CDC only recently began to track nonvascular plants; this plant occurrence had not been processed at the time of Jones' request.

If you have additional questions, please contact me.

Department of Energy
Washington, DC 20585

October 30, 1998

Mr. Virgil Franklin Sr.
Cheyenne-Arapaho Tribe of Oklahoma
P.O. Box 38
Concho OK 73022

Subject: *Consultation for Surplus Plutonium Disposition Environmental Impact Analysis Process, Under Executive Memorandum Concerning Government-to-Government Relations with Native American Tribal Governments*

Dear Mr. Franklin:

The purpose of this letter is to notify you that the United States Department of Energy (DOE) is in the process of conducting an Environmental Impact Analysis concerning the disposition of surplus plutonium.

With this letter we are soliciting specific concerns the Cheyenne-Arapaho Tribe of Oklahoma may have about the proposal. This consultation is in accordance with the Executive Memorandum (29 April 1994) entitled, "Government-to-Government Relations with Native American Tribal Governments", and DOE Order 1230.2. It also follows prior consultation initiated for compliance with the American Indian Religious Freedom Act (AIRFA) (PL 95-341) and the Native American Graves Protection and Repatriation Act (NAGPRA) (PL 101-601).

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. DOE is producing the SPD EIS in compliance with the National Environmental Policy Act (NEPA) and Council on Environmental Quality regulations implementing NEPA, DOE's NEPA Implementing Regulations (10 CFR 1021), and other applicable federal and state environmental legislation.

The purpose and need for the proposed action is to reduce the threat of nuclear weapons proliferation worldwide by disposing of surplus plutonium in the United States in an environmentally safe and timely manner. The SPD Draft EIS, a copy of which is attached for your review, examines the potential environmental impacts for 24 alternatives for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion; mixed oxide (MOX) fuel fabrication; and plutonium conversion and immobilization.

Mr. Virgil Franklin Sr.
Cheyenne-Arapaho Tribe of Oklahoma
10/30/98
Page 2

If an alternative is selected that includes siting of surplus plutonium disposition facilities at the Pantex plant (e.g., Alternative 9A), a maximum of 16 hectares (39 acres) of land in or near Zone 4 would be impacted. Based on previous consultations, no traditional cultural properties have been identified in Zone 4 or immediately adjacent areas.

If you have any specific concerns about the SPD EIS proposal, we would like to hear from you. Please contact me with your concerns or questions at:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
P.O. Box 23786
Washington, DC 20026-3786
(202) 586-0149.

You may also contact Vicki Battley, Pantex Environmental Protection Team Leader, at (806) 477-3189.

Sincerely,

Marcus Jones
SPD EIS Document Manager

cc: Vicki Battley, DOE – Amarillo Area Office
Brandt Petrasek, EM-20, DOE HQ

SPD EIS enclosure

Department of Energy
Washington, DC 20585

October 30, 1998

Mr. Billy Evans Horse
Kiowa Tribe of Oklahoma
P.O. Box 369
Carnegie OK 73015

Subject: *Consultation for Surplus Plutonium Disposition Environmental Impact Analysis Process, Under Executive Memorandum Concerning Government-to-Government Relations with Native American Tribal Governments*

Dear Mr. Evans Horse:

The purpose of this letter is to notify you that the United States Department of Energy (DOE) is in the process of conducting an Environmental Impact Analysis concerning the disposition of surplus plutonium.

With this letter we are soliciting specific concerns the Kiowa Tribe of Oklahoma may have about the proposal. This consultation is in accordance with the Executive Memorandum (29 April 1994) entitled, "Government-to-Government Relations with Native American Tribal Governments", and DOE Order 1230.2. It also follows prior consultation initiated for compliance with the American Indian Religious Freedom Act (AIRFA) (PL 95-341) and the Native American Graves Protection and Repatriation Act (NAGPRA) (PL 101-601).

The *Surplus Plutonium Disposition Environmental Impact Statement* (SPD EIS) is tiered from the *Storage and Disposition of Weapons-Usable Fissile Materials Final Programmatic EIS* (DOE/EIS-0229), issued in December 1996, and the associated Record of Decision (62 FR 3014), issued on January 14, 1997. DOE is producing the SPD EIS in compliance with the National Environmental Policy Act (NEPA) and Council on Environmental Quality regulations implementing NEPA, DOE's NEPA Implementing Regulations (10 CFR 1021), and other applicable federal and state environmental legislation.

The purpose and need for the proposed action is to reduce the threat of nuclear weapons proliferation worldwide by disposing of surplus plutonium in the United States in an environmentally safe and timely manner. The SPD Draft EIS, a copy of which is attached for your review, examines the potential environmental impacts for 24 alternatives for the proposed siting, construction, and operation of three types of facilities: pit disassembly and conversion; mixed oxide (MOX) fuel fabrication; and plutonium conversion and immobilization.

Mr. Billy Evans Horse
Kiowa Tribe of Oklahoma
10/30/98
Page 2

If an alternative is selected that includes siting of surplus plutonium disposition facilities at the Pantex plant (e.g., Alternative 9A), a maximum of 16 hectares (39 acres) of land in or near Zone 4 would be impacted. Based on previous consultations, no traditional cultural properties have been identified in Zone 4 or immediately adjacent areas.

If you have any specific concerns about the SPD EIS proposal, we would like to hear from you. Please contact me with your concerns or questions at:

Marcus Jones
SPD EIS Document Manager
U.S. Department of Energy
Office of Fissile Materials Disposition
P.O. Box 23786
Washington, DC 20026-3786
(202) 586-0149.

You may also contact Vicki Battley, Pantex Environmental Protection Team Leader, at (806) 477-3189.

Sincerely,

Marcus Jones
SPD EIS Document Manager

cc: Vicki Battley, DOE – Amarillo Area Office
Brandt Petrasek, EM-20, DOE HQ

SPD EIS enclosure