M.5.2 LONG-TERM STORAGE ALTERNATIVES

M.5.2.1 Consolidation of Plutonium Alternative

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for the consolidated Pu storage facility in the *Beyond Design Basis Accident Analysis*. The studies postulated a set of accident scenarios that were representative of the risks and consequences for workers and the public that can be expected if the facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risks that are expected to envelop the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7}/yr are anticipated that will have consequences and risks larger than those described in this section. [Text deleted.] This includes the potential impacts of an aircraft crash which has been considered and dismissed because the probability of crashing into a single facility and causing sufficient damage to release Pu is much lower than 10^{-7}/yr.

M.5.2.1.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the facility. Through a screening process, three evaluation basis accidents and seven beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.2.1.1–1 and M.5.2.1.1–2. Accident scenario descriptions are provided in Table M.5.2.1.1–3. Accident source term information is provided in Tables M.5.2.1.1–4 and M.5.2.1.1–5.

[Text deleted.]

Table M.5.2.1.1–1. Evaluation Basis Accident Scenarios for Consolidation Alternative

Accident Scenario	Accident Frequency (per year)	Source Term at Risk[a] (PCV)	Source Term Released to Environment (g Pu)
PCV puncture by forklift	6.0×10^{-4}	2	0.0387
PCV breach by firearms discharge	3.5×10^{-4}	1	3.87×10^{-3}
PCV penetration by corrosion	0.064	1	0.158

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
Source: DOE 1995mm.

Table M.5.2.1.1–2. Beyond Evaluation Basis Accident Scenarios for Consolidation Alternative

Accident Scenario	Accident Frequency (per year)	Source Term at Risk[a] (PCV)	Source Term Released to Environment
Vault fire	1.0×10^{-7}	120	81.3 g Pu
Truck bay fire	1.0×10^{-7}	12	5.40 g Pu
Spontaneous combustion	7.0×10^{-7}	2	7.75×10^{-3} g Pu
Explosion in the vault	1.0×10^{-7}	45	12.7 g Pu
Explosion outside of vault	1.0×10^{-7}	1	0.058 g Pu
Nuclear criticality	1.0×10^{-7}	[b]	1.0×10^{19} fissions[b]
Beyond evaluation basis earthquake	1.0×10^{-7}	194	146 g Pu

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
[b] See Table M.5.2.1.1–5.
Source: DOE 1995mm.

Table M.5.2.1.1–3. Accident Scenario Descriptions for Consolidation and Collocation Facilities

Accident Scenario	Accident Description
Evaluation Basis Accidents	
PCV puncture by forklift	A forklift driver attempting to pick up a pallet containing PCVs in the shipping/receiving area encounters a situation in which the fork is incorrectly positioned such that it contacts the PCVs. Before the operator responds to the contact, the forward motion of the forklift punctures two PCVs with the tines of the fork. The operator backs the forklift away from the structure, and the PCVs fall off the fork, spilling some of the contents on the floor.
PCV breach by firearms discharge	Because of the armed security guard force at the storage facility, it is necessary to consider possible breach of a PCV caused by a bullet from accidental discharge of the guard's firearm. The PCV is not designed to withstand such an impact, and its effect would be to potentially penetrate the container and cause some dispersal of the contents. This can occur only where the PCVs are above the operating floor, and would be most likely in the shipping/receiving area and possibly some material handling areas.
PCV penetration by corrosion	The PCV is presumed to fail because of long-term corrosion, gradual buildup of internal pressure, or other causes generally internal to the PCV itself, and probably related to its contents. These events would generally be the result of errors in packaging the contents or in sealing the PCV. The failure would take place over an extended period, and the initial progress of the failure would be undetectable through casual external observation. Eventually, the PCV closure seal would be breached and a small slit or crack would develop. The opening would be enlarged through continuation of the driving force and eventually some PCV contents would be expelled into the storage area or into one of the handling/inspection areas.
Beyond Evaluation Basis Accidents	
Vault fire	A large amount of jet fuel, gasoline, or some high energy density fuel is introduced into the vault through a ventilation duct and ignited.
Truck bay fire	A fire occurs following the rupture of the a truck's fuel tank and ignition of the spilled fuel. A single trailer is engulfed by flames and is heated to at least the ignition point of Pu.
Spontaneous combustion	Due to improper packaging, the contents of two PCVs ignite spontaneously after being punctured by a forklift accident.
Explosion in the vault	An explosion of undefined origin is assumed to occur below grade in the vault. The detonation is assumed to deform some storage tubes, which in turn crush and open some PCVs. There is no fire and other systems remain intact.
Explosion outside of vault	An explosion of undefined origin is assumed to occur in the repackaging area. The blast has sufficient force to breach the glovebox, exposing the contents to the room atmosphere and bypassing two levels of filtration material.
Nuclear criticality	The only way a criticality event could occur would be in the case of multiple operational errors or an accident scenario that breaches PCVs and the fissile material somehow collects in a criticality favorable geometry.
Beyond evaluation basis earthquake	The building collapses and some PCVs are crushed.

Note: PCV=primary containment vessel.

Source: DOE 1995mm.

Table M.5.2.1.1–4. Consolidation Alternative Evaluation Basis Accident Source Terms

Accident Parameter	Accident Scenario		
	PCV Puncture by Forklift	PCV Breach by Firearms Discharge	PCV Penetration by Corrosion
Frequency of occurrence (per year)	6.0×10^{-4}	3.5×10^{-4}	6.4×10^{-2}
Pu released to environment (g)	0.0387	3.87×10^{-3}	0.158
Isotope Released to Environment (Ci)			
Pu-238	6.11×10^{-5}	6.11×10^{-6}	2.50×10^{-4}
Pu-239	2.21×10^{-3}	2.21×10^{-4}	9.04×10^{-3}
Pu-240	5.88×10^{-4}	5.88×10^{-5}	2.40×10^{-3}
Pu-241	2.09×10^{-3}	2.09×10^{-4}	8.52×10^{-3}
Pu-242	8.63×10^{-8}	8.63×10^{-9}	3.52×10^{-7}
Am-241	1.10×10^{-5}	1.10×10^{-6}	4.49×10^{-5}

Note: Am=Americium; PCV=primary containment vessel.
Source: Derived from Tables M.5.1.3.4–1 and M.5.2.1.1–1.

Table M.5.2.1.1–5. Consolidation Alternative Beyond Evaluation Basis Accident Source Terms

Accident Parameter	Accident Scenario						
	Vault Fire	Truck Bay Fire	Spontaneous Combustion	Explosion in the Vault	Explosion Outside of Vault	Nuclear Criticality	Beyond Evaluation Basis Earthquake
Frequency of occurrence (per year)	1.0×10^{-7}	1.0×10^{-7}	7.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}
Pu released to environment (g)	81.3	5.40	7.75×10^{-3}	12.69	0.058	NA	146.39
Fission	NA	NA	NA	NA	NA	1.0×10^{19}	NA
Isotope Released to Environment (Ci)							
Pu-238	0.128	8.53×10^{-3}	1.22×10^{-5}	0.020	9.16×10^{-5}	0	0.231
Pu-239	4.65	0.309	4.43×10^{-4}	0.726	3.32×10^{-3}	0	8.37
Pu-240	1.24	0.082	1.18×10^{-4}	0.193	8.82×10^{-4}	0	2.23
Pu-241	4.38	0.291	4.18×10^{-4}	0.684	3.13×10^{-3}	0	7.89
Pu-242	1.81×10^{-4}	1.20×10^{-5}	1.73×10^{-8}	2.83×10^{-5}	1.29×10^{-7}	0	3.26×10^{-4}
Am-241	0.023	1.53×10^{-3}	2.20×10^{-6}	3.60×10^{-3}	1.65×10^{-5}	0	4.16×10^{-2}
Kr-83m	0	0	0	0	0	55.0	0
Kr-85m	0	0	0	0	0	35.5	0
Kr-85	0	0	0	0	0	4.05×10^{-3}	0
Kr-87	0	0	0	0	0	215	0
Kr-88	0	0	0	0	0	115	0
Kr-89	0	0	0	0	0	6.5×10^{3}	0
Xe-131	0	0	0	0	0	0.05	0
Xe-133m	0	0	0	0	0	1.10	0
Xe-133	0	0	0	0	0	13.5	0
Xe-135m	0	0	0	0	0	1.65×10^{3}	0
Xe-135	0	0	0	0	0	205	0
Xe-137	0	0	0	0	0	2.45×10^{4}	0
Xe-138	0	0	0	0	0	5.5×10^{3}	0
I-131	0	0	0	0	0	0.55	0
I-132	0	0	0	0	0	60.0	0
I-133	0	0	0	0	0	8.0	0
I-134	0	0	0	0	0	215	0
I-135	0	0	0	0	0	22.5	0

[Text deleted.]

Note: NA=not applicable.

Source: Derived from Tables M.5.1.3.4–1 and M.5.2.1.1–2.

M.5.2.1.2 *Accident Impacts*

The estimated impacts of the postulated accidents at each site are provided in Tables M.5.2.1.2–1 through M.5.2.1.2–5. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.2.1.1–4 and M.5.2.1.1–5 using the MACCS computer code. [Text deleted].

Table M.5.2.1.2–1. Consolidation Alternative Accident Impacts at Hanford Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
PCV puncture by forklift	0.011	4.4×10^{-6}	8.8×10^{-5}	4.4×10^{-8}	0.64	3.2×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	1.1×10^{-3}	4.4×10^{-7}	8.8×10^{-6}	4.4×10^{-9}	0.064	3.2×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.045	1.8×10^{-5}	3.6×10^{-4}	1.8×10^{-7}	2.6	1.3×10^{-3}	6.4×10^{-2}
Vault fire	23.1	0.012	0.18	9.2×10^{-5}	1,340	0.67	1.0×10^{-7}
Truck bay fire	1.5	6.1×10^{-4}	0.012	6.1×10^{-6}	89	0.045	1.0×10^{-7}
Spontaneous combustion	2.2×10^{-3}	8.8×10^{-7}	1.8×10^{-5}	8.8×10^{-9}	0.13	6.4×10^{-5}	7.0×10^{-7}
Explosion in the vault	3.6	1.4×10^{-3}	0.029	1.4×10^{-5}	209	0.11	1.0×10^{-7}
Explosion outside of vault	0.016	6.6×10^{-6}	1.3×10^{-4}	6.6×10^{-8}	0.96	4.8×10^{-4}	1.0×10^{-7}
Nuclear criticality	0.010	4.2×10^{-6}	6.5×10^{-5}	3.3×10^{-8}	0.07	3.5×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	41.6	0.022	0.33	1.7×10^{-4}	2,410	1.2	1.0×10^{-7}
[Text deleted]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.1.1–4 and M.5.2.1.1–5 and the MACCS computer code.

Table M.5.2.1.2–2. Consolidation Alternative Accident Impacts at Nevada Test Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
PCV puncture by forklift	7.5×10^{-3}	3.0×10^{-6}	1.4×10^{-4}	7.0×10^{-8}	0.014	7.2×10^{-6}	6.0×10^{-4}
PCV breach by firearms discharge	7.5×10^{-4}	3.0×10^{-7}	1.4×10^{-5}	7.0×10^{-9}	1.4×10^{-3}	7.2×10^{-7}	3.5×10^{-4}
PCV penetration by corrosion	0.031	1.2×10^{-5}	5.7×10^{-4}	2.9×10^{-7}	0.059	3.0×10^{-5}	6.4×10^{-2}
Vault fire	15.8	7.6×10^{-3}	0.29	1.5×10^{-4}	30.3	0.015	1.0×10^{-7}
Truck bay fire	1.0	4.2×10^{-4}	0.019	9.7×10^{-5}	2.0	1.0×10^{-3}	1.0×10^{-7}
Spontaneous combustion	1.5×10^{-3}	6.0×10^{-7}	2.8×10^{-5}	1.4×10^{-8}	2.9×10^{-3}	1.5×10^{-6}	7.0×10^{-7}
Explosion in the vault	2.5	9.9×10^{-4}	0.046	2.3×10^{-5}	4.7	2.4×10^{-3}	1.0×10^{-7}
Explosion outside of vault	0.011	4.5×10^{-6}	2.1×10^{-4}	1.0×10^{-7}	0.021	1.1×10^{-5}	1.0×10^{-7}
Nuclear criticality	7.7×10^{-3}	3.1×10^{-6}	1.3×10^{-4}	6.5×10^{-8}	1.4×10^{-3}	6.9×10^{-7}	1.0×10^{-7}
Beyond evaluation basis earthquake	28.4	0.015	0.53	2.6×10^{-4}	55	0.027	1.0×10^{-7}
[Text deleted]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.1.1–4 and M.5.2.1.1–5 and the MACCS computer code.

Table M.5.2.1.2–3. Consolidation Alternative Accident Impacts at Idaho National Engineering Laboratory

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
PCV puncture by forklift	0.010	4.1×10^{-6}	8.8×10^{-5}	4.4×10^{-8}	0.19	9.6×10^{-5}	6.0×10^{-4}
PCV breach by firearms discharge	1.0×10^{-3}	4.1×10^{-7}	8.8×10^{-6}	4.4×10^{-9}	0.19	9.6×10^{-6}	3.5×10^{-4}
PCV penetration by corrosion	0.042	1.7×10^{-5}	3.6×10^{-4}	1.8×10^{-7}	0.78	3.9×10^{-4}	6.4×10^{-2}
Vault fire	21.6	0.011	0.19	9.3×10^{-5}	402	0.20	1.0×10^{-7}
Truck bay fire	1.4	5.7×10^{-4}	0.012	6.2×10^{-6}	26.7	0.013	1.0×10^{-7}
Spontaneous combustion	2.1×10^{-3}	8.2×10^{-7}	1.8×10^{-5}	8.9×10^{-9}	0.038	1.9×10^{-5}	7.0×10^{-7}
Explosion in the vault	3.4	1.3×10^{-3}	0.029	1.5×10^{-5}	62.7	0.031	1.0×10^{-7}
Explosion outside of vault	0.015	6.2×10^{-6}	1.3×10^{-4}	6.7×10^{-8}	0.29	1.4×10^{-4}	1.0×10^{-7}
Nuclear criticality	0.010	4.0×10^{-6}	7.7×10^{-5}	3.9×10^{-8}	0.018	9.0×10^{-6}	1.0×10^{-7}
Beyond evaluation basis earthquake	38.9	0.021	0.34	1.7×10^{-4}	723	0.36	1.0×10^{-7}
[Text deleted]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.1.1–4 and M.5.2.1.1–5 and the MACCS computer code.

Table M.5.2.1.2–4. Consolidation Alternative Accident Impacts at Pantex Plant

Accident Scenario	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		Accident Frequency (per year)
	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	
PCV puncture by forklift	4.4×10^{-3}	1.8×10^{-6}	1.4×10^{-3}	7.1×10^{-7}	0.22	1.1×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	4.4×10^{-4}	1.8×10^{-7}	1.4×10^{-4}	7.1×10^{-8}	0.022	1.1×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.018	7.2×10^{-6}	5.8×10^{-3}	2.9×10^{-6}	0.89	4.4×10^{-4}	6.4×10^{-2}
Vault fire	9.3	3.8×10^{-3}	3.0	1.5×10^{-3}	456	0.23	1.0×10^{-7}
Truck bay fire	0.62	2.5×10^{-4}	0.20	9.9×10^{-5}	303	0.015	1.0×10^{-7}
Spontaneous combustion	8.9×10^{-4}	3.5×10^{-7}	2.8×10^{-4}	1.4×10^{-7}	0.044	2.2×10^{-5}	7.0×10^{-7}
Explosion in the vault	1.5	5.8×10^{-4}	0.46	2.3×10^{-4}	71.2	0.036	1.0×10^{-7}
Explosion outside of vault	6.6×10^{-3}	2.7×10^{-6}	2.1×10^{-3}	1.1×10^{-6}	0.33	1.6×10^{-4}	1.0×10^{-7}
Nuclear criticality	4.8×10^{-3}	1.9×10^{-6}	1.9×10^{-3}	9.3×10^{-7}	0.046	2.3×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	16.7	7.5×10^{-3}	5.34	2.7×10^{-3}	821	0.41	1.0×10^{-7}
[Text deleted]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.1.1–4 and M.5.2.1.1–5 and the MACCS computer code.

Table M.5.2.1.2–5. Consolidation Alternative Accident Impacts at Savannah River Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
PCV puncture by forklift	7.2×10^{-3}	2.9×10^{-6}	1.4×10^{-4}	7.1×10^{-8}	0.068	3.4×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	7.2×10^{-4}	2.9×10^{-7}	1.4×10^{-5}	7.1×10^{-9}	0.068	3.4×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.029	1.2×10^{-5}	5.8×10^{-4}	2.9×10^{-7}	2.8	1.4×10^{-3}	6.4×10^{-2}
Vault fire	15.2	6.9×10^{-3}	0.3	1.5×10^{-4}	1,440	0.72	1.0×10^{-7}
Truck bay fire	1.0	4.0×10^{-4}	0.020	9.9×10^{-6}	95.5	0.048	1.0×10^{-7}
Spontaneous combustion	1.4×10^{-3}	5.8×10^{-7}	2.8×10^{-5}	1.4×10^{-8}	0.14	6.9×10^{-5}	7.0×10^{-7}
Explosion in the vault	2.4	9.4×10^{-4}	0.046	2.3×10^{-5}	224	0.11	1.0×10^{-7}
Explosion outside of vault	0.011	4.3×10^{-6}	2.1×10^{-4}	1.1×10^{-7}	1.0	5.1×10^{-4}	1.0×10^{-7}
Nuclear criticality	6.9×10^{-2}	2.8×10^{-6}	1.1×10^{-4}	5.7×10^{-8}	0.094	4.7×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	27.3	0.013	0.53	2.7×10^{-4}	2,590	1.3	1.0×10^{-7}
[Text deleted]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.1.1–4 and M.5.2.1.1–5 and the MACCS computer code.

M.5.2.2 Collocation of Plutonium and Highly Enriched Uranium Alternatives

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for the consolidated special nuclear storage plant in the *Beyond Design Basis Accident Analysis*. The study results are directly applicable for the evaluation of the collocation of Pu and HEU storage facilities because consequences of Pu-based accidents bound the consequences of similar uranium-based accidents. The consolidated special nuclear storage plant studies postulated a set of Pu-based accident scenarios that were representative of the risks and consequences for workers and the public that can be expected if the consolidation of Pu and collocation with HEU storage facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risks that are expected to envelop the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7}/yr are anticipated that will have consequences and risks larger than those described in this section.

M.5.2.2.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the facility. Through a screening process, three evaluation basis accidents and seven beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.2.2.1–1 and M.5.2.2.1–2. Accident source term information is provided in Tables M.5.2.2.1–3 and M.5.2.2.1–4.

Table M.5.2.2.1–1. Evaluation Basis Accident Scenarios for Collocation Alternative

Accident Scenario	Accident Frequency (per year)	Source Term at Risk (PCV)[a]	Source Term Released to Environment
PCV puncture by forklift	6.0×10^{-4}	2	0.0387 g Pu
PCV breach by firearms discharge	3.5×10^{-4}	1	3.87×10^{-3} g Pu
PCV penetration by corrosion	0.064	1	0.158 g Pu

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
Source: DOE 1995mm.

Table M.5.2.2.1–2. Beyond Evaluation Basis Accident Scenarios for Collocation Alternative

Accident Scenario	Accident Frequency (per yr)	Source Term at Risk (PCV)[a]	Source Term Released to Environment
Vault fire	1.0×10^{-7}	120	81.3 g Pu
Truck bay fire	1.0×10^{-7}	12	5.40 g Pu
Spontaneous combustion	7.0×10^{-7}	2	7.75×10^{-3} g Pu
Explosion in the vault	1.0×10^{-7}	45	12.7 g Pu
Explosion outside of vault	1.0×10^{-7}	1	0.058 g Pu
Nuclear criticality	1.0×10^{-7}	[b]	1.0×10^{19} fissions[b]
Beyond evaluation basis earthquake	1.0×10^{-7}	194	146

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
[b] See Table M.5.2.1.1–5.
Source: DOE 1995mm.

Table M.5.2.2.1–3. Collocation Alternative Evaluation Basis Accident Source Terms

Accident Parameter	Accident Scenario		
	PCV Puncture by Forklift	PCV Breach by Firearms Discharge	PCV Penetration by Corrosion
Frequency of occurrence (per yr)	6.0×10^{-4}	3.5×10^{-4}	0.064
Pu released to environment (g)	0.0387	3.87×10^{-3}	0.158
Isotope released to environment (Ci)			
Pu-238	6.11×10^{-5}	6.11×10^{-6}	2.50×10^{-4}
Pu-239	2.21×10^{-3}	2.21×10^{-4}	9.04×10^{-3}
Pu-240	5.88×10^{-4}	5.88×10^{-5}	2.40×10^{-3}
Pu-241	2.09×10^{-3}	2.09×10^{-4}	8.52×10^{-3}
Pu-242	8.63×10^{-8}	8.63×10^{-9}	3.52×10^{-7}
Am-241	1.10×10^{-5}	1.10×10^{-6}	4.49×10^{-5}

Note: PCV=primary containment vessel.
Source: Derived from Tables M.5.1.3.4–1 and M.5.2.2.1–1.

Table M.5.2.2.1–4. Collocation Alternative Beyond Evaluation Basis Accident Source Terms

Accident Parameter	Accident Scenario						
	Vault Fire	Truck Bay Fire	Spontaneous Combustion	Explosion in the Vault	Explosion Outside of Vault	Nuclear Criticality	Beyond Evaluation Basis Earthquake
Frequency of occurrence[a] (per year)	1.0×10^{-7}	1.0×10^{-7}	7.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}
Pu Released to environment (g)	81.3	5.40	7.75×10^{-3}	12.69	0.058	NA	146
Fissions	NA	NA	NA	NA	NA	1.0×10^{19}	NA
Isotope released to environment (Ci)							
Pu-238	0.128	8.53×10^{-3}	1.22×10^{-5}	0.0201	9.16×10^{-5}	0	0.231
Pu-239	4.65	0.309	4.43×10^{-4}	0.726	3.32×10^{-3}	0	8.37
Pu-240	1.24	0.082	1.18×10^{-4}	0.193	8.82×10^{-4}	0	2.23
Pu-241	4.38	0.291	4.18×10^{-4}	0.684	3.13×10^{-3}	0	7.89
Pu-242	1.81×10^{-4}	1.20×10^{-5}	1.73×10^{-8}	2.83×10^{-5}	1.29×10^{-7}	0	3.26×10^{-4}
Am-241	0.023	1.53×10^{-3}	2.20×10^{-6}	3.60×10^{-3}	1.65×10^{-5}	0	4.16×10^{-2}
Kr-83m	0	0	0	0	0	55.0	0
Kr-85m	0	0	0	0	0	35.5	0
Kr-85	0	0	0	0	0	4.05×10^{-3}	0
Kr-87	0	0	0	0	0	215	0
Kr-88	0	0	0	0	0	115	0
Kr-89	0	0	0	0	0	6.5×10^{3}	0
Xe-131m	0	0	0	0	0	0.05	0
Xe-133m	0	0	0	0	0	1.10	0
Xe-133	0	0	0	0	0	13.5	0
Xe-135m	0	0	0	0	0	1.65×10^{3}	0
Xe-135	0	0	0	0	0	205	0
Xe-137	0	0	0	0	0	2.45×10^{4}	0
Xe-138	0	0	0	0	0	5.5×10^{3}	0
I-131	0	0	0	0	0	0.55	0
I-132	0	0	0	0	0	60.0	0
I-133	0	0	0	0	0	8.0	0
I-134	0	0	0	0	0	215	0
I-135	0	0	0	0	0	22.5	0

[a] Midpoint of the estimated frequency range.

Note: NA=not applicable.

Source: Derived from Tables M.5.1.3.4–1 and M.5.2.2.1–1.

M.5.2.2.2 *Accident Impacts*

The estimated impacts of the postulated accidents at each site are provided in Tables M.5.2.2.2–1 through M.5.2.2.2–6. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.2.2.1–3 and M.5.2.2.1–4 using the MACCS computer code. [Text deleted.]

Table M.5.2.2.2–1. Collocation Alternative Accident Impacts at Hanford Site

Accident Scenario	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		Accident Frequency (per year)
	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	
PCV puncture by forklift	0.011	4.4×10^{-6}	8.8×10^{-5}	4.4×10^{-8}	0.64	3.2×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	1.1×10^{-3}	4.4×10^{-7}	8.8×10^{-6}	4.4×10^{-9}	0.064	3.2×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.045	1.8×10^{-5}	3.6×10^{-4}	1.8×10^{-7}	2.6	1.3×10^{-3}	0.064
Vault fire	23.1	0.012	0.18	9.2×10^{-5}	1,340	0.67	1.0×10^{-7}
Truck bay fire	1.5	6.1×10^{-4}	0.012	6.1×10^{-6}	89	0.045	1.0×10^{-7}
Spontaneous combustion	2.2×10^{-3}	8.8×10^{-7}	1.8×10^{-5}	8.8×10^{-9}	0.13	6.4×10^{-5}	7.0×10^{-7}
Explosion in the vault	3.6	1.4×10^{-3}	0.029	1.4×10^{-5}	209	0.11	1.0×10^{-7}
Explosion outside the vault	0.016	6.6×10^{-6}	1.3×10^{-4}	6.6×10^{-8}	0.96	4.8×10^{-4}	1.0×10^{-7}
Nuclear criticality	0.010	4.2×10^{-6}	6.5×10^{-5}	3.3×10^{-8}	0.07	3.5×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	41.6	0.022	0.33	1.7×10^{-4}	2,410	1.2	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.2.1–3 and M.5.2.2.1–4 and the MACCS computer code.

Table M.5.2.2.2–2. Collocation Alternative Accident Impacts at Nevada Test Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
PCV puncture by forklift	7.5×10^{-3}	3.0×10^{-6}	1.4×10^{-4}	7.0×10^{-8}	0.014	7.2×10^{-6}	6.0×10^{-4}
PCV breach by firearms discharge	7.5×10^{-4}	3.0×10^{-7}	1.4×10^{-5}	7.0×10^{-9}	1.4×10^{-3}	7.2×10^{-7}	3.5×10^{-4}
PCV penetration by corrosion	0.031	1.2×10^{-5}	5.7×10^{-4}	2.9×10^{-7}	0.059	3.0×10^{-5}	0.064
Vault fire	15.8	7.6×10^{-3}	0.29	1.5×10^{-4}	30.3	0.015	1.0×10^{-7}
Truck bay fire	1.0	4.2×10^{-4}	0.019	9.7×10^{-6}	2.0	1.0×10^{-3}	1.0×10^{-7}
Spontaneous combustion	1.5×10^{-3}	6.0×10^{-7}	2.8×10^{-5}	1.4×10^{-8}	2.9×10^{-3}	1.5×10^{-6}	7.0×10^{-7}
Explosion in the vault	2.5	9.9×10^{-4}	0.046	2.3×10^{-5}	4.7	2.4×10^{-3}	1.0×10^{-7}
Explosion outside the vault	0.011	4.5×10^{-6}	2.1×10^{-4}	1.0×10^{-7}	0.022	1.1×10^{-5}	1.0×10^{-7}
Nuclear criticality	7.7×10^{-3}	3.1×10^{-6}	1.3×10^{-4}	6.5×10^{-8}	1.4×10^{-3}	6.9×10^{-7}	1.0×10^{-7}
Beyond evaluation basis earthquake	28.4	0.015	0.53	2.6×10^{-4}	55	0.027	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.2.1–3 and M.5.2.2.1–4 and the MACCS computer code.

Table M.5.2.2.2–3. Collocation Alternative Accident Impacts at Idaho National Engineering Laboratory

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
PCV puncture by forklift	0.010	4.1×10^{-6}	8.8×10^{-5}	4.4×10^{-8}	0.19	9.6×10^{-5}	6.0×10^{-4}
PCV breach by firearms discharge	1.0×10^{-3}	4.1×10^{-7}	8.8×10^{-6}	4.4×10^{-9}	0.019	9.6×10^{-6}	3.5×10^{-4}
PCV penetration by corrosion	0.042	1.7×10^{-5}	3.6×10^{-4}	1.8×10^{-7}	0.78	3.9×10^{-4}	0.064
Vault fire	21.6	0.011	0.19	9.3×10^{-5}	402	0.20	1.0×10^{-7}
Truck bay fire	1.4	5.7×10^{-4}	0.012	6.2×10^{-6}	26.7	0.013	1.0×10^{-7}
Spontaneous combustion	2.1×10^{-3}	8.2×10^{-7}	1.8×10^{-5}	8.9×10^{-9}	0.038	1.9×10^{-5}	7.0×10^{-7}
Explosion in the vault	3.4	1.3×10^{-3}	0.029	1.5×10^{-5}	62.7	0.031	1.0×10^{-7}
Explosion outside the vault	0.015	6.2×10^{-6}	1.3×10^{-4}	6.7×10^{-8}	0.29	1.4×10^{-4}	1.0×10^{-7}
Nuclear criticality	0.010	4.0×10^{-6}	7.7×10^{-5}	3.9×10^{-8}	0.018	9.0×10^{-6}	1.0×10^{-7}
Beyond evaluation basis earthquake	38.9	0.021	0.34	1.7×10^{-4}	723	0.36	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.2.1–3 and M.5.2.2.1–4 and the MACCS computer code.

Table M.5.2.2.2–4. Collocation Alternative Accident Impacts at Pantex Plant

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
PCV puncture by forklift	4.4×10^{-3}	1.8×10^{-6}	1.4×10^{-3}	7.1×10^{-7}	0.22	1.1×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	4.4×10^{-4}	1.8×10^{-7}	1.4×10^{-4}	7.1×10^{-8}	0.022	1.1×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.018	7.2×10^{-6}	5.8×10^{-3}	2.9×10^{-6}	0.89	4.4×10^{-4}	0.064
Vault fire	9.3	3.8×10^{-3}	3.0	1.5×10^{-3}	456	0.23	1.0×10^{-7}
Truck bay fire	0.62	2.5×10^{-4}	0.20	9.9×10^{-5}	30.3	0.015	1.0×10^{-7}
Spontaneous combustion	8.9×10^{-4}	3.5×10^{-7}	2.8×10^{-4}	1.4×10^{-7}	0.044	2.2×10^{-5}	7.0×10^{-7}
Explosion in the vault	1.5	5.8×10^{-4}	0.46	2.3×10^{-4}	71.2	0.036	1.0×10^{-7}
Explosion outside the vault	6.6×10^{-3}	2.7×10^{-6}	2.1×10^{-3}	1.1×10^{-6}	0.33	1.6×10^{-4}	1.0×10^{-7}
Nuclear criticality	4.8×10^{-3}	1.9×10^{-6}	1.9×10^{-3}	9.3×10^{-7}	0.046	2.3×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	16.7	7.5×10^{-3}	5.3	2.7×10^{-3}	821	0.41	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.2.1–3 and M.5.2.2.1–4 and the MACCS computer code.

Table M.5.2.2.2–5. Collocation Alternative Accident Impacts at Oak Ridge Reservation

Accident Scenario	Worker at 619 m		Maximum Offsite Individual		Population to 80 km		Accident Frequency (per year)
	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality	Dose (person-rem)	Number of Cancer Fatalities[b]	
PCV puncture by forklift	0.015	6.0×10^{-6}	0.015	7.5×10^{-6}	2.6	1.3×10^{-3}	6.0×10^{-4}
PCV breach by firearms discharge	1.5×10^{-3}	6.0×10^{-7}	1.5×10^{-3}	7.5×10^{-7}	0.26	1.3×10^{-4}	3.5×10^{-4}
PCV penetration by corrosion	0.061	2.5×10^{-5}	0.061	3.1×10^{-5}	10.6	5.3×10^{-3}	0.064
Vault fire	31.6	0.016	31.6	0.019	5,480	2.7	1.0×10^{-7}
Truck bay fire	2.1	8.4×10^{-4}	2.1	1.1×10^{-3}	364	0.18	1.0×10^{-7}
Spontaneous combustion	3.0×10^{-3}	1.2×10^{-6}	3.0×10^{-3}	1.5×10^{-6}	0.52	2.6×10^{-4}	7.0×10^{-7}
Explosion in the vault	4.9	2.0×10^{-3}	4.9	2.5×10^{-3}	856	0.43	1.0×10^{-7}
Explosion outside the vault	0.023	9.0×10^{-6}	0.023	1.1×10^{-5}	3.9	2.0×10^{-3}	1.0×10^{-7}
Nuclear criticality	0.014	5.5×10^{-6}	0.014	6.9×10^{-6}	0.83	4.1×10^{-4}	1.0×10^{-7}
Beyond evaluation basis earthquake	57	0.032	57	0.041	9,870	4.9	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m or the site boundary [619 m for this facility at ORR], 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.2.1–3 and M.5.2.2.1–4 and the MACCS computer code.

Table M.5.2.2.2–6. Collocation Alternative Accident Impacts at Savannah River Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
PCV puncture by forklift	7.2×10^{-3}	2.9×10^{-6}	9.7×10^{-5}	4.8×10^{-8}	0.70	3.5×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	7.2×10^{-4}	2.9×10^{-7}	9.7×10^{-5}	4.8×10^{-9}	0.070	3.5×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.029	1.2×10^{-5}	3.9×10^{-4}	2.0×10^{-7}	2.9	1.4×10^{-3}	0.064
Vault fire	15.2	6.9×10^{-3}	0.20	1.0×10^{-4}	1,470	0.73	1.0×10^{-7}
Truck bay fire	1.0	4.0×10^{-4}	0.013	6.7×10^{-6}	97.5	0.049	1.0×10^{-7}
Spontaneous combustion	1.4×10^{-3}	5.8×10^{-7}	1.9×10^{-5}	9.7×10^{-9}	0.14	7.0×10^{-5}	7.0×10^{-7}
Explosion in the vault	2.4	9.4×10^{-4}	0.032	1.6×10^{-5}	229	0.12	1.0×10^{-7}
Explosion outside the vault	0.011	4.3×10^{-6}	1.5×10^{-4}	7.3×10^{-8}	1.1	5.2×10^{-4}	1.0×10^{-7}
Nuclear criticality	6.9×10^{-3}	2.8×10^{-6}	7.0×10^{-5}	3.5×10^{-8}	0.088	4.4×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	27.2	0.013	0.37	1.8×10^{-4}	2,640	1.3	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.2.1–3 and M.5.2.2.1–4 and the MACCS computer code

M.5.2.3 Upgrade of Existing Interim Storage Facilities at Hanford Site

The Fuels Materials Examination Facility (FMEF) at Hanford is partially completed and can be upgraded for the long-term storage of Pu that is presently on the site. The FMEF has never operated and therefore safety documentation pertaining to the storage of Pu is not available. Under the Upgrade Alternative, the FMEF will be modified and improved to meet the requirements for long-term storage of Pu. A second option is a new long-term storage facility which could be constructed in the 200 Area West at Hanford.

The impacts associated with upgraded storage of Pu that already exists at Hanford are shown in Table M.5.2.3–1. The impacts apply to either the FMEF upgrade or a new storage facility in the 200 West Area. The impacts are derived from the impacts for consolidated storage at Hanford with an adjustment for smaller amounts of materials at risk for certain accident scenarios for which the amount of Pu available is a factor.

Table M.5.2.3–1. Upgrade Without Rocky Flats Environmental Technology Site Plutonium or Los Alamos National Laboratory Plutonium Subalternative Accident Impacts at Hanford Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
PCV puncture by forklift	0.011	4.4×10^{-6}	8.8×10^{-5}	4.4×10^{-8}	0.64	3.2×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	1.1×10^{-3}	4.4×10^{-7}	8.8×10^{-6}	4.4×10^{-9}	0.064	3.2×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.045	1.8×10^{-5}	3.6×10^{-4}	1.8×10^{-7}	2.6	1.3×10^{-3}	6.4×10^{-3}
Vault fire	2.3	1.2×10^{-3}	0.018	9.2×10^{-6}	134	0.067	1.0×10^{-7}
Truck bay fire	1.5	6.1×10^{-4}	0.012	6.1×10^{-6}	89	0.045	1.0×10^{-7}
Spontaneous combustion	2.2×10^{-3}	8.8×10^{-7}	1.8×10^{-5}	8.8×10^{-9}	0.13	6.4×10^{-5}	7.0×10^{-7}
Explosion in the vault	0.36	1.4×10^{-4}	2.9×10^{-3}	1.4×10^{-6}	20.9	0.011	1.0×10^{-7}
Explosion outside of vault	0.016	6.6×10^{-6}	1.3×10^{-4}	6.6×10^{-8}	0.96	4.8×10^{-4}	1.0×10^{-7}
Nuclear criticality	0.010	4.2×10^{-6}	6.5×10^{-5}	3.3×10^{-8}	0.07	3.5×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	4.2	2.2×10^{-3}	0.033	1.7×10^{-5}	241	0.12	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using Table M.5.2.1.2–1 and adjustments for smaller amounts of Pu to be stored.

The impacts associated with the upgraded storage of only the Pu from RFETS and LANL are shown in Table M.5.2.3–2. The impacts apply to either the FMEF or a new storage facility in the 200 West Area. The impacts are derived from the impacts for consolidated storage at Hanford with an adjustment for smaller amounts of materials at risk for certain accident scenarios for which the amount of Pu available is a factor.

[Text deleted.]

Table M.5.2.3–2. Upgrade With Rocky Flats Environmental Technology Site Plutonium and Los Alamos National Laboratory Plutonium Subalternative Accident Impacts at Hanford Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
PCV puncture by forklift	0.011	4.4×10^{-6}	8.8×10^{-5}	4.4×10^{-8}	0.64	3.2×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	1.1×10^{-3}	4.4×10^{-7}	8.8×10^{-6}	4.4×10^{-9}	0.064	3.2×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.045	1.8×10^{-5}	3.6×10^{-4}	1.8×10^{-7}	2.6	1.3×10^{-3}	6.6×10^{-3}
Vault fire	2.4	1.2×10^{-3}	0.018	9.4×10^{-6}	137	0.069	1.0×10^{-7}
Truck bay fire	1.5	6.1×10^{-4}	0.012	6.1×10^{-6}	89	0.045	1.0×10^{-7}
Spontaneous combustion	2.2×10^{-3}	8.8×10^{-7}	1.8×10^{-5}	8.8×10^{-9}	0.13	6.4×10^{-5}	7.0×10^{-7}
Explosion in the vault	0.37	1.4×10^{-4}	3.0×10^{-3}	1.4×10^{-6}	21.4	0.011	1.0×10^{-7}
Explosion outside of vault	0.016	6.6×10^{-6}	1.3×10^{-4}	6.6×10^{-8}	0.96	4.8×10^{-4}	1.0×10^{-7}
Nuclear criticality	0.010	4.2×10^{-6}	6.5×10^{-5}	3.3×10^{-8}	0.07	3.5×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	4.3	2.3×10^{-3}	0.034	1.7×10^{-5}	247	0.12	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using Table M.5.2.1.2–1 and adjustments for smaller amounts of Pu to be stored.

M.5.2.4 Upgrade of Existing Interim Storage Facilities at Idaho National Engineering Laboratory

The Pu storage facilities at ANL-W at INEL are presently used for Pu storage and can be upgraded for the long-term storage of Pu that is presently on the site. Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed in the *Beyond Design Basis Accident Analysis*. The studies postulated a set of accident scenarios that were representative of the risks and consequences for workers. Although not all potential accidents were addressed, those that were postulated have consequences and risks that are expected to envelop the consequences and risks of the facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7}/yr are anticipated that will have consequences and risks larger than those described in this section.

The impacts associated with the upgraded storage of only the Pu from RFETS and LANL are shown in Table M.5.2.4–1. The impacts are derived from the impacts for consolidated storage at INEL with an adjustment for smaller amounts of material at risk for certain accident scenarios for which the amount of Pu available is a factor.

M.5.2.4.1 *Accident Scenarios and Source Terms*

A range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the upgraded Pu storage facility at ANL-W. Through a screening process,

Table M.5.2.4–1. Upgrade With Rocky Flats Environmental Technology Site Plutonium and Los Alamos National Laboratory Plutonium Subalternative Accident Impacts at Idaho National Engineering Laboratory

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
PCV puncture by forklift	0.010	4.1×10^{-7}	8.8×10^{-5}	4.4×10^{-8}	0.19	9.6×10^{-5}	6.0×10^{-4}
PCV breach by firearms discharge	1.0×10^{-3}	4.1×10^{-7}	8.8×10^{-6}	4.4×10^{-9}	0.19	9.6×10^{-6}	3.5×10^{-4}
PCV penetration by corrosion	0.042	1.7×10^{-5}	3.6×10^{-4}	1.8×10^{-7}	0.78	3.9×10^{-4}	6.6×10^{-3}
Vault fire	2.21	1.1×10^{-3}	0.019	9.5×10^{-5}	41.2	0.021	1.0×10^{-7}
Truck bay fire	1.4	5.7×10^{-4}	0.012	6.2×10^{-6}	26.7	0.013	1.0×10^{-7}
Spontaneous combustion	2.1×10^{-3}	8.2×10^{-7}	1.8×10^{-5}	8.9×10^{-9}	0.038	1.9×10^{-5}	1.0×10^{-7}
Explosion in the vault	0.35	1.3×10^{-4}	3.0×10^{-3}	1.5×10^{-6}	6.4	3.2×10^{-3}	1.0×10^{-7}
Explosion outside of vault	0.015	6.2×10^{-6}	1.3×10^{-4}	6.7×10^{-8}	0.29	1.4×10^{-4}	1.0×10^{-7}
Nuclear criticality	0.010	4.0×10^{-6}	7.7×10^{-5}	3.9×10^{-8}	0.018	9.0×10^{-6}	1.0×10^{-7}
Beyond evaluation basis earthquake	4.0	2.2×10^{-3}	0.035	1.7×10^{-5}	74.1	0.037	1.0×10^{-7}

[Text deleted.]

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using Table M.5.2.1.2–3 and adjustments for smaller amounts of Pu to be stored.

three evaluation basis accidents and seven beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.2.4.1–1 and M.5.2.4.1–2. Accident source term information is provided in Tables M.5.2.4.1–3 and M.5.2.4.1–4. Accident scenario descriptions are provided in Table M.5.2.4.1–5.

Table M.5.2.4.1–1. Evaluation Basis Accident Scenarios for Upgrade of Interim Storage Facilities at Idaho National Engineering Laboratory

Accident Scenario	Accident Frequency (per year)	Source Term at Risk (PCV)[a]	Source Term Released to Environment
PCV puncture by forklift	6.0×10^{-4}	2	0.038 g Pu
PCV breach by firearms discharge	3.5×10^{-4}	1	3.8×10^{-3} g Pu
PCV penetration by corrosion	0.064	1	0.157 g Pu

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
Source: DOE 1995mm.

Table M.5.2.4.1–2. Beyond Evaluation Basis Accident Scenarios for Upgrade of Interim Storage Facilities at Idaho National Engineering Laboratory

Accident Scenario	Accident Frequency (per year)	Source Term at Risk (PCV)[a]	Source Term Released to Environment
Vault fire	1.0×10^{-7}	264	66.3 g Pu
Truck bay fire	1.0×10^{-7}	12	5.42 g Pu
Spontaneous combustion	1.4×10^{-7}	2	7.6×10^{-3} g Pu
Explosion in the vault	1.0×10^{-7}	66	49.8 g Pu
Explosion outside of vault	1.0×10^{-7}	1	0.0542 g Pu
Nuclear criticality	1.0×10^{-7}	[b]	1.0×10^{19} fissions[b]
Beyond evaluation basis earthquake	1.0×10^{-7}	132	99.5 g Pu

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
[b] See Table M.5.2.1.1–5.
Source: DOE 1995mm.

Table M.5.2.4.1–3. Evaluation Basis Accident Source Terms for Upgrade of Interim Storage Facilities at Idaho National Engineering Laboratory

	Accident Scenario		
Accident Parameter	PCV Puncture by Forklift	PCV Breach by Firearms Discharge	PCV Penetration by Corrosion
Frequency of occurrence (per year)	6.0×10^{-4}	3.5×10^{-4}	0.064
Pu released to environment (g)	0.038	3.8×10^{-3}	0.157
Isotope released to environment (Ci)			
Pu-238	4.75×10^{-5}	4.75×10^{-6}	1.96×10^{-4}
Pu-239	2.20×10^{-3}	2.20×10^{-4}	9.09×10^{-3}
Pu-240	5.21×10^{-4}	5.21×10^{-5}	2.15×10^{-3}
Pu-241	4.29×10^{-3}	4.29×10^{-4}	0.0177
Pu-242	1.49×10^{-8}	1.49×10^{-9}	6.17×10^{-8}
Am-241	8.02×10^{-4}	8.02×10^{-5}	3.31×10^{-3}

Note: Am=Americium; PCV=primary containment vessel.
Source: Derived from Tables M.5.1.3.4–1 and M.5.2.4.1–1.

Table M.5.2.4.1–4. Beyond Evaluation Basis Accident Source Terms for Upgrade of Interim Storage Facilities at Idaho National Engineering Laboratory

	Accident Scenario						
Accident Parameter	**Vault Fire**	**Truck Bay Fire**	**Spontaneous Combustion**	**Explosion in the Vault**	**Explosion Outside of Vault**	**Nuclear Criticality**	**Beyond Evaluation Basis Earthquake**
Frequency of occurrence (per year)	$1.0x10^{-7}$	$1.0x10^{-7}$	$1.40x10^{-7}$	$1.0x10^{-7}$	$1.0x10^{-7}$	$1.0x10^{-7}$	$1.0x10^{-7}$
Pu released to environment (g)	66.3	5.42	$7.6x10^{-3}$	49.77	0.0542	NA	99.5
Fissions	NA	NA	NA	NA	NA	$1.0x10^{19}$	NA
Isotope released to environment (Ci)							
Pu-238	0.0829	$6.78x10^{-3}$	$9.50x10^{-6}$	0.0622	$6.78x10^{-5}$	0	0.124
Pu-239	3.84	0.314	$4.40x10^{-4}$	2.88	$3.14x10^{-3}$	0	5.76
Pu-240	0.908	0.0743	$1.04x10^{-4}$	0.682	$7.43x10^{-4}$	0	1.36
Pu-241	7.49	0.612	$8.59x10^{-4}$	5.62	$6.12x10^{-3}$	0	11.2
Pu-242	$2.61x10^{-5}$	$2.13x10^{-6}$	$2.99x10^{-9}$	$1.96x10^{-5}$	$2.13x10^{-8}$	0	$3.91x10^{-5}$
Am-241	1.4	0.114	$1.60x10^{4}$	1.05	$1.14x10^{-3}$	0	2.10
Kr-83m	0	0	0	0	0	55.0	0
Kr-85m	0	0	0	0	0	35.5	0
Kr-85	0	0	0	0	0	$4.05x10^{-3}$	0
Kr-87	0	0	0	0	0	215	0
Kr-88	0	0	0	0	0	115	0
Kr-89	0	0	0	0	0	$6.5x10^{3}$	0
Xe-131m	0	0	0	0	0	0.05	0
Xe-133m	0	0	0	0	0	1.10	0
Xe-133	0	0	0	0	0	13.5	0
Xe-135m	0	0	0	0	0	$1.65x10^{3}$	0
Xe-135	0	0	0	0	0	205	0
Xe-137	0	0	0	0	0	$2.45x10^{4}$	0
Xe-138	0	0	0	0	0	$5.5x10^{3}$	0
I-131	0	0	0	0	0	0.55	0
I-132	0	0	0	0	0	60.0	0
I-133	0	0	0	0	0	8.0	0
I-134	0	0	0	0	0	215	0
I-135	0	0	0	0	0	22.5	0

Note: NA=not applicable.

Source: Derived from Tables M.5.1.3.4–1, M.5.2.1.1–5, and M.5.2.4.1–2.

Table M.5.2.4.1–5. Accident Scenario Descriptions for Storage Facilities at Idaho National Engineering Laboratory

Accident Scenario	Accident Description
Evaluation Basis Accident	
PCV puncture by forklift	A forklift driver attempting to pick up a pallet containing PCVs in the shipping/receiving area encounters a situation in which the fork is positioned such that it contacts the PCVs. Before the operator responds to the contact, the forward motion of the forklift punctures two PCVs with the tines of the fork. The operator backs the forklift away from the structure, and the PCVs fall off the fork, spilling some of the contents on the floor.
PCV breach by firearms discharge	Because of the armed security guard force at the storage facility, it is necessary to consider possible breach of a PCV caused by a bullet from accidental discharge of the guard's firearm. The PCV is not designed to withstand such an impact, and its effect would be to potentially penetrate the container and cause some dispersal of the contents. This can occur only where the PCVs are above the operating floor, and would be most likely in the shipping/receiving area and possibly some material handling areas.
PCV penetration by corrosion	The PCV is presumed to fail because of long-term corrosion, gradual buildup of internal pressure, or other causes generally internal to the PCV itself, and probably related to its contents. These events would generally be the result of errors in packaging the contents or in sealing the PCV. The failure would take place over an extended period, and the initial progress of the failure would be undetectable through casual external observation. Eventually, the PCV closure seal would be breached and a small slit or crack would develop. The opening would be enlarged through continuation of the driving force and eventually some PCV contents would be expelled into the storage area or into the storage area or into one of the handling/inspection areas.
Beyond Evaluation Basis Accident	
Vault fire	A large amount of jet fuel, gasoline, or some other high energy density fuel is introduced into the vault through a ventilation duct and ignited.
Truck bay fire	A fire occurs following the rupture of a truck's fuel tank and ignition of the spilled fuel. A single trailer is engulfed by flames and is heated to at least the ignition point of Pu.
Spontaneous combustion	Due to improper packaging, the contents of two PCVs ignite spontaneously after being punctured by a forklift accident.
Explosion in the vault	An explosion of undefined origin is assumed to occur below grade in the vault. The detonation is assumed to deform some storage tubes, which in turn crush and open some PCVs. There is no fire, and other systems remain intact.
Explosion in the vault	An explosion of undefined origin is assumed to occur below grade in the vault. Portions of the ceiling spall or collapse. Falling debris punctures Beverly cans and the quart product can contained within.
Explosion outside of vault	An explosion of undefined origin is assumed to occur in the repackaging area. The blast has sufficient force to breach the glovebox, exposing the contents to the room atmosphere and bypassing two levels of filtration provided by the glovebox.
Nuclear criticality	The only way a criticality event could occur would be in the case of multiple operational errors or an accident scenario that breaches PCVs and the fissile material somehow collects in a criticality favorable geometry.

Table M.5.2.4.1–5. Accident Scenario Descriptions for Storage Facilities at Idaho National Engineering Laboratory—Continued

Accident Scenario	Accident Description
Beyond evaluation basis earthquake	The building collapses. The strength of the operating floor is such that it could withstand the weight and the impact. Differential motion between the operating floor and the vault floor could cause some of the storage tubes to buckle enough to crush some PCVs. Significant building damage occurs. Spalling of the roof and/or collapse of the Beverly can walls. Falling debris punctures Beverly cans and the quart product can contained within.

Note: PCV=primary containment vessel.
Source: DOE 1995mm.

M.5.2.4.2 *Accident Impacts*

The estimated impacts of the postulated accidents at INEL are provided in Table M.5.2.4.2–1. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.2.4.1–3 and M.5.2.4.1–4 using the MACCS computer code. [Text deleted.]

Table M.5.2.4.2–1. Accident Impacts for Upgrade of Existing Storage Facilities at Idaho National Engineering Laboratory

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
PCV puncture by forklift	0.014	5.6×10^{-6}	7.5×10^{-4}	3.8×10^{-7}	0.25	1.2×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	1.4×10^{-3}	5.6×10^{-7}	7.5×10^{-5}	3.8×10^{-8}	0.025	1.2×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.058	2.3×10^{-5}	3.1×10^{-3}	1.6×10^{-6}	1.0	5.1×10^{-4}	0.064
Vault fire	24.6	0.013	1.3	6.6×10^{-4}	433	0.22	1.0×10^{-7}
Truck bay fire	2.0	8.0×10^{-4}	0.11	5.4×10^{-5}	35.4	0.018	1.0×10^{-7}
Spontaneous combustion	2.8×10^{-3}	1.1×10^{-6}	1.5×10^{-4}	7.5×10^{-8}	0.05	2.5×10^{-5}	7.0×10^{-7}
Explosion in the vault	18.5	9.1×10^{-3}	0.98	4.9×10^{-4}	325	0.16	1.0×10^{-7}
Explosion outside the vault	0.020	8.0×10^{-6}	1.1×10^{-3}	5.4×10^{-7}	0.35	1.8×10^{-4}	1.0×10^{-7}
Nuclear criticality	0.010	4.0×10^{-6}	5.9×10^{-4}	3.0×10^{-7}	0.019	9.6×10^{-6}	1.0×10^{-7}
Beyond evaluation basis earthquake	36.9	0.020	2.0	9.8×10^{-4}	650	0.33	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.4.1–3 and M.5.2.4.1–4 and the MACCS computer code.

M.5.2.5 Upgrade or Consolidation of Existing Interim Storage Facilities at Pantex Plant

The accident analysis of the upgrade or consolidation of existing facilities at Pantex consists of two buildings: a Surplus Materials Storage Building (SM Building) and a Strategic Reserves Storage Building (SR Building). Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed in the *Beyond Design Basis Accident Analysis*. The studies postulated a set of accident scenarios that were representative of the risks and consequences for workers. Although not all potential accidents were addressed, those that were postulated have consequences and risks that are expected to envelop the consequences and risks of the facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7}/yr are anticipated that will have consequences and risks larger than those described in this section. [Text deleted.]

M.5.2.5.1 *Accident Scenarios and Source Terms*

A range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the modified Pu storage facilities at Pantex. Through a screening process, three evaluation basis accidents and seven beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.2.5.1–1 and M.5.2.5.1–2 for the SM Building and in Tables M.5.2.5.1–3 and M.5.2.5.1–4 for the SR Building. Accident source term information is provided in Tables M.5.2.5.1–5 and M.5.2.5.1–6 for the SM Building and in Tables M.5.2.5.1–7 and M.5.2.5.1–8 for the SR Building. Accident scenario descriptions are provided in Table M.5.2.5.1–9.

Table M.5.2.5.1–1. Evaluation Basis Accident Scenarios for Upgrade or Consolidation at Pantex Plant—Surplus Materials Storage Building

Accident Scenario	Accident Frequency (per year)	Source Term at Risk (PCV)[a]	Source Term Release to Environment
PCV puncture by forklift	6.0×10^{-4}	2	0.038 g Pu
PCV breach by firearms discharge	3.5×10^{-4}	1	3.8×10^{-3} g Pu
PCV penetration by corrosion	6.4×10^{-2}	1	0.157 g Pu

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
Source: DOE 1995mm.

Table M.5.2.5.1–2. Beyond Evaluation Basis Accident Scenarios for Upgrade or Consolidation at Pantex Plant—Surplus Materials Storage Building

Accident Scenario	Accident Frequency (per year)	Source Term at Risk (PCV)[a]	Source Term Release to Environment
Vault fire	1.0×10^{-7}	120	81 g Pu
Truck bay fire	1.0×10^{-7}	12	5.42 g Pu
Spontaneous combustion	7.0×10^{-7}	2	7.6×10^{-3} g Pu
Explosion in the vault	1.0×10^{-7}	45	12.6 g Pu
Explosion outside of vault	1.0×10^{-7}	1	0.054 g Pu
Nuclear criticality	1.0×10^{-7}	[b]	1.0×10^{19} fissions[b]
Beyond evaluation basis earthquake	1.0×10^{-7}	194	54.76 g Pu

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
[b] See Table M.5.2.5.1–6.
Source: DOE 1995mm.

Table M.5.2.5.1–3. Evaluation Basis Accident Scenarios for Upgrade or Consolidation at Pantex Plant—Strategic Reserves Storage Building

Accident Scenario	Accident Frequency (per year)	Source Term at Risk (PCV)[a]	Source Term Release to Environment
PCV puncture by forklift	6.0×10^{-4}	2	0.038 g Pu
PCV breach by firearms discharge	3.5×10^{-4}	1	3.8×10^{-3} g Pu
PCV penetration by corrosion	0.064	1	0.157 g Pu

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
Source: DOE 1995mm.

Table M.5.2.5.1–4. Beyond Evaluation Basis Accident Scenarios for Upgrade or Consolidation at Pantex Plant—Strategic Reserves Storage Building

Accident Scenario	Accident Frequency (per year)	Source Term at Risk (PCV)[a]	Source Term Release to Environment
Vault fire	1.0×10^{-7}	406	483 g Pu
Truck bay fire	1.0×10^{-7}	12	5.42 g Pu
Spontaneous combustion	7.0×10^{-7}	2	7.6×10^{-3} g Pu
Explosion in the vault	1.0×10^{-7}	120	16.3 g Pu
Explosion outside of vault	1.0×10^{-7}	1	0.054 g Pu
Nuclear criticality	1.0×10^{-7}	[b]	1.0×10^{19} fissions[b]
Beyond evaluation basis earthquake	1.0×10^{-7}	194	111.1 g Pu

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
[b] See Table M.5.2.5.1–6.
Source: DOE 1995mm.

Table M.5.2.5.1–5. Evaluation Basis Accident Source Terms for Upgrade or Consolidation at Pantex Plant—Surplus Materials Storage Building

Accident Parameter	Accident Scenario		
	PCV Puncture by Forklift	PCV Breach by Firearms Discharge	PCV Penetration by Corrosion
Frequency of occurrence (per year)	6.0×10^{-4}	3.5×10^{-4}	0.064
Pu released to environment (g)	0.038	3.8×10^{-3}	0.157
Isotope Released to Environment (Ci)			
Pu-238	2.14×10^{-5}	2.14×10^{-6}	8.84×10^{-4}
Pu-239	2.20×10^{-3}	2.20×10^{-4}	9.07×10^{-3}
Pu-240	5.17×10^{-4}	5.17×10^{-5}	2.14×10^{-3}
Pu-241	1.63×10^{-3}	1.63×10^{-4}	6.75×10^{-3}
Pu-242	1.49×10^{-8}	1.49×10^{-9}	6.15×10^{-8}
Am-241	8.63×10^{-4}	8.63×10^{-5}	3.56×10^{-3}

Note: PCV=primary containment vessel.

Source: Derived from Tables M.5.1.3.4–2 and M.5.2.5.1–1.

Table M.5.2.5.1–6. Beyond Evaluation Basis Accident Source Terms for Upgrade or Consolidation at Pantex Plant—Surplus Materials Storage Building

	Accident Scenario						
Accident Parameter	**Vault Fire**	**Truck Bay Fire**	**Spontaneous Combustion**	**Explosion in the Vault**	**Explosion Outside of Vault**	**Nuclear Criticality**	**Beyond Evaluation Basis Earthquake**
Frequency of occurrence (per year)	1.0×10^{-7}	1.0×10^{-7}	1.40×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}
Pu released to environment (g)	81	5.42	7.6×10^{-3}	12.6	0.054	NA	54.76
Fissions	NA	NA	NA	NA	NA	1.0×10^{19}	NA
Isotope Released to Environment (Ci)							
Pu-238	0.0456	3.05×10^{-3}	4.28×10^{-6}	7.09×10^{-3}	3.04×10^{-5}	0	0.0308
Pu-239	4.68	0.313	4.39×10^{-4}	0.728	3.12×10^{-3}	0	3.17
Pu-240	1.10	0.0737	1.03×10^{-4}	0.171	7.34×10^{-4}	0	0.745
Pu-241	3.48	0.233	3.27×10^{-4}	0.542	2.32×10^{-3}	0	2.35
Pu-242	3.18×10^{-5}	2.12×10^{-6}	2.98×10^{-9}	4.94×10^{-6}	2.12×10^{-8}	0	2.15×10^{-5}
Am-241	1.84	0.123	1.73×10^{-4}	0.286	1.23×10^{-3}	0	1.24
Kr-83m	0	0	0	0	0	55.0	0
Kr-85m	0	0	0	0	0	35.5	0
Kr-85	0	0	0	0	0	4.05×10^{-3}	0
Kr-87	0	0	0	0	0	215	0
Kr-88	0	0	0	0	0	115	0
Kr-89	0	0	0	0	0	6.5×10^{3}	0
Xe-131m	0	0	0	0	0	0.05	0
Xe-133m	0	0	0	0	0	1.10	0
Xe-133	0	0	0	0	0	13.5	0
Xe-135m	0	0	0	0	0	1.65×10^{3}	0
Xe-135	0	0	0	0	0	205	0
Xe-137	0	0	0	0	0	2.45×10^{4}	0
Xe-138	0	0	0	0	0	5.5×10^{3}	0
I-131	0	0	0	0	0	0.55	0
I-132	0	0	0	0	0	60.0	0
I-133	0	0	0	0	0	8.0	0
I-134	0	0	0	0	0	215	0
I-135	0	0	0	0	0	22.5	0

Note: NA=not applicable.

Source: Derived from Tables M.5.1.3.4–2, M.5.2.1.1–3, and M.5.2.5.1–2.

Table M.5.2.5.1–7. Evaluation Basis Accident Source Terms for Upgrade or Consolidation at Pantex Plant—Strategic Reserves Storage Building

	Accident Scenario		
Accident Parameter	**PCV Puncture by Forklift**	**PCV Breach by Firearms Discharge**	**PCV Penetration by Corrosion**
Frequency of occurrence (per year)	6.0×10^{-4}	3.5×10^{-4}	0.064
Pu released to environment (g)	0.038	3.8×10^{-3}	0.157
Isotope Released to Environment (Ci)			
Pu-238	2.14×10^{-5}	2.14×10^{-6}	8.84×10^{-4}
Pu-239	2.20×10^{-3}	2.20×10^{-4}	9.07×10^{-3}
Pu-240	5.17×10^{-4}	5.17×10^{-5}	2.14×10^{-3}
Pu-241	1.63×10^{-3}	1.63×10^{-4}	6.75×10^{-3}
Pu-242	1.49×10^{-8}	1.49×10^{-9}	6.15×10^{-8}
Am-241	8.63×10^{-4}	8.63×10^{-5}	3.56×10^{-3}

Note: PCV=primary containment vessel.
Source: Derived from Tables M.5.1.3.4–2 and M.5.2.5.1–3.

Table M.5.2.5.1–8. Beyond Evaluation Basis Accident Source Terms for Upgrade or Consolidation at Pantex Plant—Strategic Reserves Storage Building

	Accident Scenario						
Accident Parameter	**Vault Fire**	**Truck Bay Fire**	**Spontaneous Combustion**	**Explosion in the Vault**	**Explosion Outside of Vault**	**Nuclear Criticality**	**Beyond Evaluation Basis Earthquake**
Frequency of occurrence (per year)	1.0×10^{-7}	1.0×10^{-7}	1.40×10^{-7}/yr	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}
Pu released to environment (g)	482.5	5.42	7.6×10^{-3}	16.27	0.054	NA	111.1
Fissions	NA	NA	NA	NA	NA	1.0×10^{19}	NA
Isotope Released to Environment (Ci)							
Pu-238	0.272	3.05×10^{-3}	428×10^{-6}	9.16x10-3	3.04×10^{-5}	0	0.0625
Pu-239	27.9	0.313	4.39×10^{-4}	0.940	3.12×10^{-3}	0	6.42
Pu-240	6.56	0.0737	1.04×10^{-4}	0.221	7.34×10^{-4}	0	1.51
Pu-241	20.7	0.233	3.27×10^{-4}	0.700	2.32×10^{-3}	0	4.78
Pu-242	1.89×10^{-4}	2.12×10^{-6}	2.98×10^{-9}	6.38×10^{-6}	2.12×10^{-8}	0	4.36×10^{-5}
Am-241	11.0	0.114	1.73×10^{-4}	0.369	1.23×10^{-3}	0	2.52
Kr-83m	0	0	0	0	0	55.0	0
Kr-85m	0	0	0	0	0	35.5	0
Kr-85	0	0	0	0	0	4.05×10^{-3}	0
Kr-87	0	0	0	0	0	215	0
Kr-88	0	0	0	0	0	115	0
Kr-89	0	0	0	0	0	6.5×10^{3}	0
Xe-131m	0	0	0	0	0	0.05	0
Xe-133m	0	0	0	0	0	1.10	0
Xe-133	0	0	0	0	0	13.5	0
Xe-135m	0	0	0	0	0	1.65×10^{3}	0
Xe-135	0	0	0	0	0	205	0
Xe-137	0	0	0	0	0	2.45×10^{4}	0
Xe-138	0	0	0	0	0	5.5×10^{3}	0
I-131	0	0	0	0	0	0.55	0
I-132	0	0	0	0	0	60.0	0
I-133	0	0	0	0	0	8.0	0
I-134	0	0	0	0	0	215	0
I-135	0	0	0	0	0	22.5	0

Note: NA=not applicable.

Source: Derived from Tables M.5.1.3.4–2 and M.5.2.5.1–4.

Table M.5.2.5.1–9. Accident Scenario Descriptions for Storage Facilities at Pantex Plant

Accident Scenario	Accident Description
Evaluation Basis Accident	
PCV puncture by forklift	A forklift driver attempting to pick up a pallet containing PCVs in the shipping/receiving area encounters a situation in which the fork is positioned such that it contacts the PCVs. Before the operator responds to the contact, the forward motion of the forklift punctures two PCVs with the tines of the fork. The operator backs the forklift away from the structure, and the PCVs fall off the fork, spilling some of the contents on the floor.
PCV breach by firearms discharge	Because of the armed security guard force at the storage facility, it is necessary to consider possible breach of a PCV caused by a bullet from accidental discharge of the guard's firearm. The PCV is not designed to withstand such an impact, and its effect would be to potentially penetrate the container and cause some dispersal of the contents. This can occur only where the PCVs are above the operating floor, and would be most likely in the shipping/receiving area and possibly some material handling areas.
PCV penetration by corrosion	The PCV is presumed to fail because of long-term corrosion, gradual buildup of internal pressure, or other causes generally internal to the PCV itself, and probably related to its contents. These events would generally be the result of errors in packaging the contents or in sealing the PCV. The failure would take place over an extended period, and the initial progress of the failure would be undetectable through casual external observation. Eventually, the PCV closure seal would be breached and a small slit or crack would develop. The opening would be enlarged through continuation of the driving force, and eventually some PCV contents would be expelled into the storage area or into one of the handling/inspection areas.
Beyond Evaluation Basis Accident	
Vault fire	A large amount of jet fuel, gasoline, or some other high energy density fuel is introduced into the vault through a ventilation duct and ignited.
Truck bay fire	A fire occurs following the rupture of a truck's fuel tank and ignition of the spilled fuel. A single trailer is engulfed by flames and is heated to at least the ignition point of Pu.
Spontaneous combustion	Due to improper packaging, the contents of two PCVs ignite spontaneously after being punctured by a forklift accident.
Explosion in the vault	An explosion of undefined origin is assumed to occur below grade in the vault. The detonation is assumed to deform some storage tubes, which in turn crush and open some PCVs. There is no fire, and other systems remain intact.
Explosion in the vault	An explosion of undefined origin is assumed to occur below grade in the vault. The explosion is confined to a single compartment. The effect will be to damage some of the racks and some of the PCVs. PCVs within 20 ft of the blast may be damaged and breached.
Explosion outside of vault	An explosion of undefined origin is assumed to occur in the repackaging area. The blast has sufficient force to breach the glovebox, exposing the contents to the room atmosphere and bypassing two levels of filtration provided by the glovebox.
Nuclear criticality	The only way a criticality event could occur would be in the case of multiple operational errors or an accident scenario that breaches PCVs and the fissile material somehow collects in a criticality favorable geometry.

Table M.5.2.5.1–9. Accident Scenario Descriptions for Storage Facilities at Pantex Plant—Continued

Accident Scenario	Accident Description
Beyond evaluation basis earthquake	The building collapses. The strength of the operating floor is such that it could withstand the weight and the impact. Differential motion between the operating floor and the vault floor could cause some of the storage tubes to buckle enough to crush some PCVs. The PCVs will fall onto the storage compartment floor. Some of the PCVs may be damaged by falling debris and breached.

Note: PCV=primary containment vessel.
Source: DOE 1995mm.

M.5.2.5.2 *Accident Impacts*

The estimated impacts of the postulated accidents for the interim storage facilities in the SM Building and SR Building are provided in Tables M.5.2.5.2–1 and M.5.2.5.2–2. [Text deleted.] The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.2.5.1–5 through M.5.2.5.1–8 using the MACCS computer code. Accident impacts for the Upgrade With RFETS Pu Pits Subalternative (Preferred Alternative) are shown in Table M.5.2.5.2–3. The impacts are based on the impacts estimated for the Consolidation Alternative in Table M.5.2.1.2–4 and adjustments to certain accidents to reflect upgrade storage of 25,000 positions instead of 40,000 positions as in consolidation.

M.5.2.5.3 *Aircraft Crash*

Pantex is located approximately 13.6 km (8.5 mi) from the northeast-southwest runway at Amarillo International Airport. Potential accident scenarios in which an aircraft crashes into one or more facilities at Pantex have been developed for the Pantex EIS. A discussion of aircraft accidents for this PEIS is contained in Appendix R.

Table M.5.2.5.2–1. Accident Impacts for Consolidation in the Surplus Materials Storage Building at Pantex Plant

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
PCV puncture by forklift	6.1×10^{-3}	2.4×10^{-6}	1.9×10^{-3}	9.7×10^{-7}	0.28	1.4×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	6.1×10^{-4}	2.4×10^{-7}	1.9×10^{-4}	9.7×10^{-8}	0.028	1.4×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.025	1.0×10^{-5}	8.0×10^{-3}	4.0×10^{-6}	1.1	5.7×10^{-4}	0.064
Vault fire	13	5.7×10^{-3}	4.1	2.1×10^{-3}	588	10.29	1.0×10^{-7}
Truck bay fire	0.87	3.5×10^{-4}	0.28	1.4×10^{-4}	39.3	0.020	1.0×10^{-7}
Spontaneous combustion	1.2×10^{-3}	4.9×10^{-7}	3.9×10^{-4}	1.9×10^{-7}	0.055	2.8×10^{-5}	7.0×10^{-7}
Explosion in the vault	2.0	8.1×10^{-4}	0.64	3.2×10^{-4}	91.5	0.046	1.0×10^{-7}
Explosion outside of vault	8.7×10^{-3}	3.5×10^{-6}	2.8×10^{-3}	1.4×10^{-6}	0.39	2.0×10^{-4}	1.0×10^{-7}
Nuclear criticality	4.8×10^{-3}	1.9×10^{-6}	1.9×10^{-3}	9.3×10^{-7}	0.041	2.1×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	8.8	3.6×10^{-3}	2.8	1.4×10^{-3}	398	0.20	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.5.1–5 and M.5.2.5.1–6 and the MACCS computer code.

Table M.5.2.5.2–2. Accident Impacts for Consolidation in the Strategic Reserves Storage Building at Pantex Plant

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
PCV puncture by forklift	6.1×10^{-3}	2.4×10^{-6}	2.0×10^{-3}	9.9×10^{-7}	0.28	1.4×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	6.1×10^{-4}	2.4×10^{-7}	2.0×10^{-4}	9.9×10^{-8}	0.028	1.4×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.025	1.0×10^{-5}	8.1×10^{-3}	4.1×10^{-6}	1.2	5.7×10^{-4}	0.064
Vault fire	78	0.043	25	0.015	3,500	1.8	1.0×10^{-7}
Truck bay fire	0.87	3.5×10^{-4}	0.28	1.4×10^{-4}	39.5	0.020	1.0×10^{-7}
Spontaneous combustion	1.2×10^{-3}	4.9×10^{-7}	4.0×10^{-4}	2.0×10^{-7}	0.056	2.8×10^{-5}	7.0×10^{-7}
Explosion in the vault	2.6	1.0×10^{-3}	0.84	4.2×10^{-4}	119	0.059	1.0×10^{-7}
Explosion outside of vault	8.7×10^{-3}	3.5×10^{-6}	2.8×10^{-3}	1.4×10^{-6}	0.39	2.0×10^{-4}	1.0×10^{-7}
Nuclear criticality	4.8×10^{-3}	1.9×10^{-6}	1.9×10^{-3}	9.4×10^{-7}	0.042	2.1×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	17.8	8.0×10^{-3}	5.8	3.0×10^{-3}	810	0.44	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.5.1–7 and M.5.2.5.1–8 and the MACCS computer code.

Table M.5.2.5.2–3. Upgrade With Rocky Flats Environmental Technology Site Plutonium Pits Subalternative (Preferred Alternative) Accident Impacts at Pantex Plant

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
PCV puncture by forklift	4.4×10^{-3}	1.8×10^{-6}	1.4×10^{-3}	7.1×10^{-7}	0.22	1.1×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	4.4×10^{-4}	1.8×10^{-7}	1.4×10^{-4}	7.1×10^{-8}	0.022	1.1×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.018	7.2×10^{-6}	5.8×10^{-3}	2.9×10^{-6}	0.89	4.4×10^{-4}	0.04
Vault fire	5.8	2.4×10^{-3}	1.9	9.4×10^{-4}	285	0.14	1.0×10^{-7}
Truck bay fire	0.62	2.5×10^{-4}	0.20	1.0×10^{-4}	303	0.015	1.0×10^{-7}
Spontaneous combustion	8.9×10^{-4}	3.5×10^{-7}	2.8×10^{-4}	1.4×10^{-7}	0.044	2.2×10^{-5}	7.0×10^{-7}
Explosion in the vault	0.94	3.6×10^{-4}	0.29	1.4×10^{-4}	44.5	0.023	1.0×10^{-7}
Explosion outside of vault	6.6×10^{-3}	2.7×10^{-6}	2.1×10^{-3}	1.1×10^{-6}	0.33	1.6×10^{-4}	1.0×10^{-7}
Nuclear criticality	4.8×10^{-3}	1.9×10^{-6}	1.9×10^{-3}	9.3×10^{-7}	0.046	2.3×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	10.4	4.7×10^{-3}	3.34	1.7×10^{-3}	513	0.26	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using M.5.2.1.2–4 and adjustments for smaller amounts of Pu to be stored.

M.5.2.6 Upgrade of Existing Storage Facilities at Oak Ridge Reservation

The HEU storage facilities at ORR are presently used for HEU storage and would be upgraded for the purpose of long-term storage of HEU that is presently on the site. Under the Upgrade Alternative, existing facilities at the Y–12 site would be modified and improved to meet the requirements for long-term storage of HEU. For upgraded conditions, potential accidents and their consequences have previously been addressed and documented according to requirements in DOE Orders.

Under the Preferred Alternative at ORR, nonsurplus HEU and surplus HEU pending disposition would remain in storage at Y–12 in existing and upgraded storage facilities. Upgrades for HEU storage in Building 9212, the building used in the Y–12 EA accident analysis, would include structural modifications to numerous columns, knee braces, and cross braces to provide proper stiffness and load distribution as documented in *Natural Phenomena Upgrade of the Downsized/Consolidated Oak Ridge Uranium/Lithium Plant Facilities* (Y/EN-5080, 1994). Appendix G of the Y–12 EA contains a list of buildings and the modifications required to bring the buildings into conformance with the target performance goal that is equivalent to the structural response of new facilities. The modifications made to these facilities are expected to result in a reduction in risk of accidents to workers and the public for equivalent quantities of stored HEU. Modification to these facilities would ensure that long-term storage would be in accordance with DOE Orders, and that the risks to the public of prompt fatalities due to accidents and of latent cancer fatalities due to normal operations would be minimized. These structural modifications would reduce the risk from seismic initiators such as a beyond design basis earthquake scenario.

Buildings included in the upgrade for long-term storage at Y–12, as described in Section 2.3.1, would be evaluated by analyses employing methodologies outlined in DOE Order 5480.21, Unreviewed Safety Questions; DOE Order 5480.22, Technical Safety Requirements; and DOE Order 5480.23, Nuclear Safety Analysis Reports. Facilities and buildings within Y–12 that contain substantial quantities of enriched uranium have DOE-approved SARs that are currently undergoing review in an SAR Update Program to meet requirements of new DOE Orders (OR DOE 1994:E-3). The SAR Update Program would reflect the long-term storage upgrade at Y–12 in a Conceptual Design Report for these structural modifications as part of the Stockpile Management Restructuring Initiative that DOE is pursuing.

One of the natural phenomena initiators of accident scenarios analyzed (nuclear criticality, fire, and mechanical upset) in the Y–12 EA included a design basis accident earthquake. For the earthquake scenario, the present evaluation criterion for the design basis earthquake corresponds to a hazard exceedance frequency of 5×10^{-4} per year. The Y–12 long-term storage buildings would be upgraded to meet the performance goal for a moderate hazard facility of Performance Category 3 in DOE Order 5480.28, *Natural Phenomena Hazards Mitigation*. The structures, systems, and components in a Performance Category 3 facility pose a potential hazard to worker and public health and safety and to the environment because radioactive or toxic materials are present in significant quantities. Design considerations for this category are to limit facility damage so that hazardous materials can be controlled and confined, occupants are protected, and functioning of the facility is not interrupted. A performance goal for Performance Category 3 is a hazard exceedance frequency of 1×10^{-4} per year (DOE Order 5480.28). Meeting this performance goal would reduce the expected risk for the design basis accidents analyzed in the Y–12 EA for Building 9212 by approximately 80 percent, resulting in a latent cancer fatality risk of 5.1×10^{-7} to the MEI and 5.7×10^{-8} to a noninvolved worker, and potential latent cancer fatalities of 7.4×10^{-6} for the 80-km (50-mi) offsite population.

The HEU EIS describes the disposition of surplus HEU currently stored at ORR. As surplus HEU is removed for disposition, the quantity of material in storage would be reduced, and therefore fewer buildings would be needed for storage. As this is a reduction in the storage footprint, the risk would be reduced accordingly. The combination of upgrading the buildings with structural modifications (as discussed above) and reducing the storage footprint as surplus HEU disposition continues are expected to result in overall reduction in the risk to the public and workers from facility accidents.

M.5.2.7 Upgrade of Existing Storage Facilities at Savannah River Site

Under the Upgrade at SRS alternative, the Actinide Packaging and Storage Facility at SRS would be upgraded to accommodate additional RFETS and LANL material. For No Action conditions, potential accidents and their consequences have previously been addressed and documented according to requirements in DOE Orders. The estimated impacts of potential accidents are shown in Table M.5.2.7–1. The estimates are based on the impacts shown in Table M.5.2.1.2–5 for a new consolidated storage facility with adjustments for the reduced quantities of plutonium that would be stored for the upgraded case. The adjustments are based on data supplied by Westinghouse Savannah River Company (SRS 1996a:6).

Table M.5.2.7–1. Upgrade With Rocky Flats Environmental Technology Site Plutonium and Los Alamos National Laboratory Plutonium Subalternative Accident Impacts at Savannah River Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
PCV puncture by forklift	7.2×10^{-3}	2.9×10^{-6}	1.4×10^{-4}	7.1×10^{-8}	0.068	3.4×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	7.2×10^{-4}	2.9×10^{-7}	1.4×10^{-5}	7.1×10^{-9}	0.068	3.4×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.029	1.2×10^{-5}	5.8×10^{-4}	2.9×10^{-7}	2.8	1.4×10^{-3}	6.6×10^{-3}
Vault fire	1.6	7.1×10^{-4}	0.031	1.5×10^{-5}	148	0.072	1.0×10^{-7}
Truck bay fire	1.0	4.0×10^{-4}	0.020	9.9×10^{-6}	95.5	0.048	1.0×10^{-7}
Spontaneous combustion	1.4×10^{-3}	5.8×10^{-7}	2.8×10^{-5}	1.4×10^{-8}	0.14	6.9×10^{-5}	7.0×10^{-7}
Explosion in the vault	2.5	9.6×10^{-5}	4.7×10^{-3}	2.4×10^{-6}	23.0	0.011	1.0×10^{-7}
Explosion outside of vault	0.011	4.3×10^{-6}	2.1×10^{-4}	1.1×10^{-7}	1.0	5.1×10^{-4}	1.0×10^{-7}
Nuclear criticality	6.9×10^{-2}	2.8×10^{-6}	1.1×10^{-4}	5.7×10^{-8}	0.094	4.7×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	2.8	1.3×10^{-3}	0.054	2.8×10^{-5}	265	0.13	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred. Impacts not dependent on the quantity of Pu would be the same as those for the new consolidated storage facility.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the impacts in Table M.5.2.1.2–5 with adjustments to reflect smaller quantities of Pu for upgraded storage.

Under the Preferred Alternative, the Actinide Packaging and Storage Facility at SRS would be upgraded to accommodate additional RFETS non-pit Pu material for the Preferred Alternative. The estimated impacts of potential accidents are shown in Table M.5.2.7–2. The estimates are based on the impacts shown in Table M.5.2.1.2–5 for a new consolidated storage facility with adjustments for the reduced quantities of Pu that would be stored for the upgraded case. The adjustments are based on data supplied by Westinghouse Savannah River Company (SRS 1996a:6).

M.5.2.8 Nevada Test Site Storage Facility

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for the NTS Storage Facility in the *Beyond Design Basis Accident Analysis*. The studies postulated a set of accident scenarios

Table M.5.2.7–2. Upgrade With Rocky Flats Environmental Technology Site Non-Pit Plutonium Subalternative (Preferred Alternative) Accident Impacts at Savannah River Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
PCV puncture by forklift	7.2×10^{-3}	2.9×10^{-6}	1.4×10^{-4}	7.1×10^{-8}	0.068	3.4×10^{-4}	6.0×10^{-4}
PCV breach by firearms discharge	7.2×10^{-4}	2.9×10^{-7}	1.4×10^{-5}	7.1×10^{-9}	0.068	3.4×10^{-5}	3.5×10^{-4}
PCV penetration by corrosion	0.029	1.2×10^{-5}	5.8×10^{-4}	2.9×10^{-7}	2.8	1.4×10^{-3}	4.8×10^{-3}
Vault fire	1.1	5.2×10^{-4}	0.023	1.1×10^{-5}	108	0.054	1.0×10^{-7}
Truck bay fire	1.0	4.0×10^{-4}	0.020	9.9×10^{-6}	95.5	0.048	1.0×10^{-7}
Spontaneous combustion	1.4×10^{-3}	5.8×10^{-7}	2.8×10^{-5}	1.4×10^{-8}	0.14	6.9×10^{-5}	7.0×10^{-7}
Explosion in the vault	0.18	7.1×10^{-5}	3.5×10^{-3}	1.7×10^{-6}	16.8	8.3×10^{-3}	1.0×10^{-7}
Explosion outside of vault	0.011	4.3×10^{-6}	2.1×10^{-4}	1.1×10^{-7}	1.0	5.1×10^{-4}	1.0×10^{-7}
Nuclear criticality	6.9×10^{-2}	2.8×10^{-6}	1.1×10^{-4}	5.7×10^{-8}	0.094	4.7×10^{-5}	1.0×10^{-7}
Beyond evaluation basis earthquake	2.0	9.8×10^{-4}	0.040	2.0×10^{-5}	194	0.098	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred. Impacts not dependent on the quantity of Pu would be the same as those for the new consolidated storage facility.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the impacts in Table M.5.2.1.2–5 with adjustments to reflect smaller quantities of Pu for upgraded storage.

that were representative of the risks and consequences for workers and the public that can be expected if the facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risks that are expected to envelop the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7} per year are anticipated that will have consequences and risks larger than those described in this section.

M.5.2.8.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the facility. Through a screening process, three evaluation basis accidents and four beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.2.8.1–1 and M.5.2.8.1–2. As discussed in Section M.5.1.2, Pu-based accidents bound the consequences of similar uranium-based accidents when both functions are performed at a collocated facility. Therefore, only Pu-based accidents are provided in Tables M.5.2.8.1–1 and M.5.2.8.1–2. Accident source term information is provided in Tables M.5.2.8.1–3 and M.5.2.8.1–4. Descriptions of the accident scenarios provided in Table M.5.2.8.1–5.

Table M.5.2.8.1–1. Evaluation Basis Accident Scenarios at Nevada Test Site for Plutonium Storage Facility—P-Tunnel

Accident Scenario	Accident Frequency (per year)	Source Term at Risk (PCV)[a]	Source Term Released to Environment
PCV puncture by forklift	6.0×10^{-4}	2	0.038 g Pu
PCV breach by firearms discharge	3.5×10^{-4}	1	3.8×10^{-3} g Pu
PCV penetration by corrosion	6.4×10^{-2}	1	0.163 g Pu

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
Source: DOE 1995mm.

Table M.5.2.8.1–2. Beyond Evaluation Basis Accident Scenarios at Nevada Test Site for Plutonium Storage Facility—P-Tunnel

Accident Scenario	Accident Frequency (per year)	Source Term at Risk (PCV)[a]	Source Term Released to Environment
Truck bay fire	1.0×10^{-7}	12	5.42 g Pu
Spontaneous combustion	7.0×10^{-7}	2	7.6×10^{-3} g Pu
Explosion outside of vault	1.0×10^{-7}	1	0.054 g Pu
Nuclear criticality	1.0×10^{-7}	[b]	1.0×10^{19} fissions[b]

[a] Primary containment vessel (PCV) is assumed to contain up to 4,500 g of weapons-grade Pu as a bounding case.
[b] See Table M.5.2.1.1–5.
Source: DOE 1995mm.

Table M.5.2.8.1–3. Evaluation Basis Accident Source Terms at Nevada Test Site for Plutonium and Highly Enriched Uranium Storage Facility—P-Tunnel

	Accident Scenario		
Accident Parameter	PCV Puncture by Forklift	PCV Breach by Firearms Discharge	PCV Penetration by Corrosion
Frequency of occurrence (per year)	6.0×10^{-4}	3.5×10^{-4}	6.4×10^{-2}
Pu released to environment (g)	0.038	3.8×10^{-3}	0.163
Isotope Released to Environment (Ci)			
Pu-238	6.00×10^{-5}	6.00×10^{-6}	2.58×10^{-4}
Pu-239	2.17×10^{-3}	2.17×10^{-4}	9.32×10^{-3}
Pu-240	5.78×10^{-4}	5.78×10^{-5}	2.48×10^{-3}
Pu-241	2.05×10^{-3}	2.05×10^{-4}	8.79×10^{-3}
Pu-242	8.47×10^{-8}	8.47×10^{-9}	3.63×10^{-7}
Am-241	1.08×10^{-5}	1.08×10^{-6}	4.63×10^{-5}

Note: PCV=primary containment vessel.
Source: Derived from Table M.5.1.3.4–1 and M.5.2.8.1–1.

Table M.5.2.8.1–4. Beyond Evaluation Basis Accident Source Terms at Nevada Test Site for Plutonium and Highly Enriched Uranium Storage Facility—P-Tunnel

	Accident Scenario			
Accident Parameter	**Truck Bay Fire**	**Spontaneous Combustion**	**Explosion Outside of Vault**	**Nuclear Criticality**
Frequency of occurrence (per year)	1.0×10^{-7}	7.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}
Pu released to environment (g)	5.42	7.6×10^{-3}	0.054	NA
Fissions	NA	NA	NA	1.0×10^{19}
Isotope Released to Environment (Ci)				
Pu-238	8.56×10^{-3}	1.20×10^{-5}	1.53×10^{-5}	0
Pu-239	0.310	4.35×10^{-4}	3.09×10^{-3}	0
Pu-240	0.0824	1.16×10^{-4}	8.21×10^{-4}	0
Pu-241	0.292	4.10×10^{-4}	2.91×10^{-3}	0
Pu-242	1.21×10^{-5}	1.69×10^{-8}	1.20×10^{-7}	0
Am-241	1.54×10^{-3}	2.16×10^{-6}	1.53×10^{-5}	0
Kr-83m	0	0	0	55.0
Kr-85m	0	0	0	35.5
Kr-85	0	0	0	4.05×10^{-3}
Kr-87	0	0	0	215
Kr-88	0	0	0	115
Kr-89	0	0	0	6.5×10^{3}
Xe-131m	0	0	0	0.05
Xe-133m	0	0	0	1.10
Xe-133	0	0	0	13.5
Xe-135m	0	0	0	1.65×10^{3}
Xe-135	0	0	0	205
Xe-137	0	0	0	2.45×10^{4}
Xe-138	0	0	0	5.5×10^{3}
I-131	0	0	0	0.55
I-132	0	0	0	60.0
I-133	0	0	0	8.0
I-134	0	0	0	215
I-135	0	0	0	22.5

Note: NA=not applicable.
Source: Derived from Tables M.5.1.3.4–1 and M.5.2.8.1–2.

Table M.5.2.8.1–5. Accident Scenario Descriptions for Nevada Test Site Plutonium Storage Facility—P-Tunnel

Accident Scenario	Accident Description
Evaluation Basis Accident	
PCV puncture by forklift	A forklift driver attempting to pick up a pallet containing PCVs in the shipping/receiving area encounters a situation in which the fork is positioned such that it contacts the PCVs. Before the operator responds to the contact, the forward motion of the forklift punctures two PCVs with the tines of the fork. The operator backs the forklift away from the structure, and the PCVs fall off the fork, spilling some of the contents on the floor.
PCV breach by firearms discharge	Because of the armed security guard force at the storage facility, it is necessary to consider possible breach of a PCV caused by a bullet from accidental discharge of the guard's firearm. The PCV is not designed to withstand such an impact, and its effect would be to potentially penetrate the container and cause some dispersal of the contents. This can occur only where the PCVs are above the operating floor, and would be most likely in the shipping/receiving area and possibly some material handling areas.
PCV penetration by corrosion	The PCV is presumed to fail because of long-term corrosion, gradual buildup of internal pressure, or other causes generally internal to the PCV itself, and probably related to its contents. These events would generally be the result of errors in packaging the contents or in sealing the PCV. The failure would take place over an extended period, and the initial progress of the failure would be undetectable through casual external observation. Eventually, the PCV closure seal would be breached, and a small slit or crack would develop. The opening would be enlarged through continuation of the driving force and eventually some PCV contents would be expelled into the storage area or into the storage area or into one of the handling/inspection areas.
Beyond Evaluation Basis Accident	
Truck bay fire	A fire occurs following the rupture of a truck's fuel tank and ignition of the spilled fuel. A single trailer is engulfed by flames and is heated to at least the ignition point of Pu.
Spontaneous combustion	Due to improper packaging, the contents of two PCVs ignite spontaneously after being punctured by a forklift accident.
Explosion outside of vault	An explosion of undefined origin is assumed to occur in the repackaging area. The blast has sufficient force to breach the glovebox, exposing the contents to the room atmosphere and bypassing two levels of filtration provided by the glovebox.
Nuclear criticality	The only way a criticality event could occur would be in the case of multiple operational errors or an accident scenario that breaches PCVs and the fissile material somehow collects in a criticality favorable geometry.

Note: PCV=primary containment vessel.
Source: DOE 1995mm.

M.5.2.8.2 *Accident Impacts*

The estimated impacts of the postulated accidents at NTS are provided in Table M.5.2.8.2–1. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.2.8.1–3 and M.5.2.8.1–4 using the MACCS computer code. [Text deleted.]

Table M.5.2.8.2–1. Plutonium and Highly Enriched Uranium Storage Facility Accident Impacts at Nevada Test Site—P-Tunnel

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
PCV puncture by forklift	7.4×10^{-3}	2.9×10^{-6}	1.0×10^{-3}	5.0×10^{-7}	7.4×10^{-3}	3.7×10^{-6}	6.0×10^{-4}
PCV breach by firearms discharge	7.4×10^{-4}	2.9×10^{-7}	1.0×10^{-4}	5.0×10^{-8}	7.4×10^{-4}	3.7×10^{-7}	3.5×10^{-4}
PCV penetration by corrosion	0.032	1.3×10^{-5}	4.3×10^{-3}	2.2×10^{-6}	0.032	1.6×10^{-5}	0.064
Truck bay fire	1.1	4.2×10^{-4}	0.14	7.2×10^{-5}	1.1	5.3×10^{-4}	1.0×10^{-7}
Spontaneous combustion	1.5×10^{-3}	5.9×10^{-7}	2.0×10^{-4}	1.0×10^{-7}	1.5×10^{-3}	7.5×10^{-7}	1.0×10^{-7}
Explosion outside of vault	0.011	4.2×10^{-6}	1.4×10^{-3}	7.2×10^{-7}	0.011	5.3×10^{-6}	1.0×10^{-7}
Nuclear criticality	7.7×10^{-3}	3.1×10^{-6}	1.2×10^{-3}	6.1×10^{-7}	1.3×10^{-3}	6.4×10^{-7}	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the incident has occurred.

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

Note: All values are mean values; PCV=primary containment vessel.

Source: Calculated using the source terms in Tables M.5.2.8.1–3 and M.5.2.8.1–4 and the MACCS computer code.

M.5.3 DISPOSITION ALTERNATIVES

M.5.3.1 Pit Disassembly/Conversion Facility

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for a pit disassembly/conversion facility in the *Fissile Material Disposition Program PEIS Data Call Input Report: Pit Disassembly and Conversion Facility.* The studies postulated a set of accident scenarios that were representative of the risks and consequences for workers and the public that can be expected if the facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risks that are expected to envelop the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7}/yr are anticipated that will have consequences and risks larger than those described in this section. [Text deleted.]

M.5.3.1.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the facility. Through a screening process, four evaluation basis accidents and four beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.3.1.1–1 and M.5.3.1.1–2. Accident source term information is provided in Tables M.5.3.1.1–3 through M.5.3.1.1–5. Descriptions of the accident scenarios are provided in Table M.5.3.1.1–6.

Table M.5.3.1.1–1. Evaluation Basis Accident Scenarios for the Pit Disassembly/Conversion Facility

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Fire on the loading dock	1.0×10^{-4} to 1.0×10^{-3}	18 g Pu	0.8 g Pu
Fire in a process cell	1.0×10^{-5} to 1.0×10^{-3}	24 g Pu	4.8×10^{-6} g Pu
Deflagration inside a glovebox	1.0×10^{-5} to 1.0×10^{-3}	10 kg Pu	1.0×10^{-3} g Pu
Impact induced spill	4.5×10^{-5}	4 kg PuO_2	1.7×10^{-9} g Pu

Source: LANL 1996d.

Table M.5.3.1.1–2. Beyond Evaluation Basis Accident Scenarios for the Pit Disassembly/Conversion Facility

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Nuclear criticality	$<1.0 \times 10^{-7}$	5.0×10^{17} fissions; gaseous by-products released[a]	[a]
Beyond design basis fire in a process cell	$<1.0 \times 10^{-7}$	24 g Pu	0.034 g Pu
Oxyacetylene explosion in a process cell	$<1.0 \times 10^{-7}$	10 kg Pu	50 g Pu
Beyond evaluation basis earthquake	$<1.0 \times 10^{-7}$	10 kg Pu	25 g Pu

[a] See Table M.5.3.1.1–3.
Source: LANL 1996d.

Table M.5.3.1.1–3. Pit Disassembly/Conversion Facility Criticality Source Terms

Nuclide	Produced (Ci)	Released (Ci)
Kr-83m	5.5	5.5
Kr-85m	3.55	3.55
Kr-85	4.05×10^{-4}	4.05×10^{-4}
Kr-87	21.5	21.5
Kr-88	11.5	11.5
Kr-89	650	650
Xe-131m	5.0×10^{-3}	5.0×10^{-3}
Xe-133m	0.11	0.11
Xe-133	1.35	1.35
Xe-135m	165	165
Xe-135	20.5	20.5
Xe-137	2,450	2,450
Xe-138	550	550
I-131	0.55	0.138
I-132	60	15
I-133	8	2.0
I-134	215	53.8
I-135	22.5	5.36

Source: LANL 1996d.

Table M.5.3.1.1–4. Pit Disassembly/Conversion Facility Evaluation Basis Accident Source Terms

	Accident Scenario			
Accident Parameter	Fire on Loading Dock	Beyond Design Basis Fire in Process Cell	Deflagration Inside a Glovebox	Impact Induced Spill
Frequency of occurrence (per year)	5.0×10^{-4} [a]	1.0×10^{-4} [a]	1.0×10^{-4} [a]	4.5×10^{-5}
Pu released to environment (g)	0.8	4.8×10^{-6}	1.0×10^{-3}	1.7×10^{-9}
Isotope Released to Environment (Ci)				
Pu-238	4.50×10^{-4}	2.70×10^{-9}	5.63×10^{-7}	9.57×10^{-13}
Pu-239	0.0462	2.77×10^{-7}	5.78×10^{-5}	9.83×10^{-11}
Pu-240	0.0109	6.53×10^{-8}	1.36×10^{-5}	2.31×10^{-11}
Pu-241	0.0344	2.06×10^{-7}	4.30×10^{-5}	7.31×10^{-11}
Pu-242	3.14×10^{-7}	1.88×10^{-12}	3.92×10^{-10}	6.66×10^{-16}
Am-241	0.0182	1.09×10^{-7}	2.27×10^{-5}	3.86×10^{-11}

[a] Midpoint of the estimated frequency range.

Source: Derived from Tables M.5.1.3.4–2 and M.5.3.1.1–1.

Table M.5.3.1.1–5. Pit Disassembly/Conversion Facility Beyond Evaluation Basis Accident Source Terms

	Accident Scenario			
Accident Parameter	**Nuclear Criticality**	**Beyond Design Basis Fire in a Process Cell**	**Oxyacetylene Explosion in a Process Cell**	**Beyond Design Basis Earthquake**
Frequency of occurrence (per year)[a]	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}
Pu released to environment (g)	NA	0.034	50	25
Fissions	5.0×10^{17}	NA	NA	NA
Isotope Released to Environment (Ci)				
Pu-238	0	1.91×10^{-5}	0.0281	0.0141
Pu-239	0	1.97×10^{-3}	2.89	1.44
Pu-240	0	4.62×10^{-4}	0.680	0.340
Pu-241	0	1.46×10^{-3}	2.15	1.08
Pu-242	0	1.34×10^{-8}	1.96×10^{-5}	9.80×10^{-6}
Am-241	0	7.72×10^{-4}	1.13	0.567
Kr-83m	5.5	0	0	0
Kr-85m	3.55	0	0	0
Kr-85	4.05×10^{-4}	0	0	0
Kr-87	21.5	0	0	0
Kr-88	11.5	0	0	0
Kr-89	650	0	0	0
Xe-131m	5.0×10^{-3}	0	0	0
Xe-133m	0.11	0	0	0
Xe-133	1.35	0	0	0
Xe-135m	165	0	0	0
Xe-135	20.5	0	0	0
Xe-137	2.45×10^{3}	0	0	0
Xe-138	550	0	0	0
I-131	0.138	0	0	0
I-132	15	0	0	0
I-133	2.0	0	0	0
I-134	53.8	0	0	0
I-135	5.36	0	0	0

[a] Maximum value of the estimated frequency range.
Note: NA=not applicable.
Source: Derived from Tables M.5.1.3.4–2, M.5.3.1.1–2, and M.5.3.1.1–3.

Table M.5.3.1.1–6. Accident Scenario Descriptions for the Pit Disassembly/Conversion Facility

Accident Scenario	Accident Description
Evaluation Basis Accidents	
Fire on the loading dock	The fire is caused by welding, cleaning solvents, electrical shorts, or other miscellaneous causes. The scenario assumes an open garage door and that a single drum of combustible waste is involved in the fire.
Fire in a process cell	It is assumed that a process cell contains a glovebox used for final processing of plutonium oxide powder. The gloves, stowed outside the glovebox, are coated with a layer of Pu dust. A flammable cleaning liquid such as acetone or isopropyl alcohol is brought into the process cell in violation of operating procedures, spills, and ignites. The initial extent and intensity of the fire are sufficient to completely incinerate the gloves. The sprinkler system activates and protects the glovebox from further damage. The ventilation system with HEPA filters continues to function throughout the accident.
Deflagration inside a glovebox	The bounding evaluation basis explosion is a deflagration of a flammable gas mixture inside a glovebox. It is assumed that through some unforeseen set of failures, a combustible gas mixture accumulates inside a glovebox and is ignited, possibly by an electrical spark from an operating electrical device. The deflagration blows out the HEPA filter from the glovebox ventilation system exit. Gloves may also be blown out. The room volumes are sufficient to attenuate the pressure wave to levels below that needed to damage building ventilation system HEPA filters.
Impact induced spill	The most catastrophic case of leak or spill of nuclear material would result from a forklift or other large vehicle running over a package of nuclear material, breaching the containment, and causing airborne release to the room. Three stage HEPA filtration is available for the facility exhaust to limit the release to the environment.
Beyond Evaluation Basis Accidents	
Nuclear criticality	The postulated criticality accident was caused by improper stacking or handling of bulk nuclear material. Multiple operational errors in the material spacing, packing density, manner, and type of containment, and maximum quantities of fissile material permitted in the area would be required for postulated criticality accident to occur.
Beyond evaluation basis fire in a process cell	A typical fire with coincident failures of two or more major safety systems constitutes a beyond evaluation basis fire. The evaluation postulated the fire in a process cell, discussed above, with the sprinkler system and ventilation system with HEPA filtration inoperative during the accident.
Oxyacetylene explosion in a process cell	The evaluation postulated the explosion of a welding rig oxyacetylene bottle in a process cell. The explosion is sufficient to blow out the HEPA filters and cause significant damage to the ventilation system and nearby equipment.
Beyond evaluation basis earthquake	The following assumptions were used in the evaluation: (1) the earthquake disables the ventilation system; (2) there is sufficient structural damage to the building and it does not totally collapse; (3) a ceiling slab falls on the glovebox with the most material at risk and severely damages the glovebox; (4) the process cell with the most material at risk is located on an outside wall; (5) the outside wall cracks; and (6) the wind is blowing and the cracks are located on the lee side of the building.

Source: LANL 1996d.

M.5.3.1.2 *Accident Impacts*

The estimated impacts of the postulated accidents at each site are provided in Tables M.5.3.1.2–1 through M.5.3.1.2–6. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.3.1.1–4 and M.5.3.1.1–5 using the MACCS code. [Text deleted.]

Table M.5.3.1.2–1. Pit Disassembly/Conversion Facility Accident Impacts at Hanford Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Fire on the loading dock	0.32	1.3×10^{-4}	0.010	5.1×10^{-6}	18.5	9.3×10^{-3}	5.0×10^{-5}
Fire in a process cell	1.9×10^{-6}	7.6×10^{-10}	6.1×10^{-8}	3.0×10^{-11}	1.1×10^{-4}	5.5×10^{-8}	1.0×10^{-4}
Deflagration inside a glovebox	4.0×10^{-4}	1.6×10^{-7}	1.3×10^{-5}	6.4×10^{-9}	0.023	1.2×10^{-5}	1.0×10^{-4}
Impact induced spill	6.8×10^{-10}	2.7×10^{-13}	2.2×10^{-11}	1.1×10^{-14}	3.9×10^{-8}	2.0×10^{-11}	4.5×10^{-5}
Nuclear criticality	1.7×10^{-3}	6.9×10^{-7}	5.7×10^{-5}	2.9×10^{-8}	0.016	7.8×10^{-6}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	0.014	5.4×10^{-6}	4.3×10^{-4}	2.2×10^{-7}	0.79	3.9×10^{-4}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	19.9	9.4×10^{-3}	0.63	3.2×10^{-4}	1150	0.58	1.0×10^{-7}
Beyond evaluation basis earthquake	9.9	4.1×10^{-3}	0.32	1.6×10^{-4}	576	0.29	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.1.1–4 and M.5.3.1.1–5 and the MACCS computer code.

Table M.5.3.1.2–2. Pit Disassembly/Conversion Facility Accident Impacts at Nevada Test Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Fire on the loading dock	0.22	8.7×10^{-5}	4.0×10^{-3}	2.0×10^{-6}	0.42	2.1×10^{-4}	5.0×10^{-5}
Fire in a process cell	1.3×10^{-6}	5.2×10^{-10}	2.4×10^{-8}	1.2×10^{-11}	2.5×10^{-6}	1.3×10^{-9}	1.0×10^{-4}
Deflagration inside a glovebox	2.7×10^{-4}	1.1×10^{-7}	5.1×10^{-6}	2.5×10^{-9}	5.2×10^{-4}	2.6×10^{-7}	1.0×10^{-4}
Impact induced spill	4.6×10^{-10}	1.9×10^{-13}	8.6×10^{-12}	4.3×10^{-15}	8.9×10^{-10}	4.5×10^{-13}	4.5×10^{-5}
Nuclear criticality	1.3×10^{-3}	5.0×10^{-7}	2.2×10^{-5}	1.1×10^{-8}	3.2×10^{-4}	1.6×10^{-7}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	9.3×10^{-3}	3.7×10^{-6}	1.7×10^{-4}	8.6×10^{-8}	0.018	8.9×10^{-6}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	13.6	6.3×10^{-3}	0.25	1.3×10^{-4}	26.1	1.3×10^{-2}	1.0×10^{-7}
Beyond evaluation basis earthquake	6.8	2.7×10^{-3}	0.13	6.3×10^{-5}	13.0	6.5×10^{-3}	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.1.1–4 and M.5.3.1.1–5 and the MACCS computer code.

Table M.5.3.1.2–3. Pit Disassembly/Conversion Facility Accident Impacts at Idaho National Engineering Laboratory

Accident Scenario	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		Accident Frequency (per year)
	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	
Fire on the loading dock	0.30	1.2×10^{-4}	2.6×10^{-3}	1.3×10^{-6}	5.5	2.8×10^{-3}	5.0×10^{-5}
Fire in a process cell	1.8×10^{-6}	7.1×10^{-10}	1.5×10^{-8}	7.7×10^{-12}	3.3×10^{-5}	1.7×10^{-8}	1.0×10^{-4}
Deflagration inside a glovebox	3.7×10^{-4}	1.5×10^{-7}	3.2×10^{-6}	1.6×10^{-9}	6.9×10^{-3}	3.5×10^{-6}	1.0×10^{-4}
Impact induced spill	6.3×10^{-10}	2.5×10^{-13}	5.5×10^{-12}	2.7×10^{-15}	1.2×10^{-8}	5.9×10^{-12}	4.5×10^{-5}
Nuclear criticality	1.7×10^{-3}	6.7×10^{-7}	1.3×10^{-5}	6.7×10^{-9}	4.2×10^{-3}	2.1×10^{-6}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	0.013	5.1×10^{-6}	1.1×10^{-4}	5.5×10^{-8}	0.24	1.2×10^{-4}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	18.6	9.2×10^{-3}	0.16	8.0×10^{-5}	346	0.17	1.0×10^{-7}
Beyond evaluation basis earthquake	9.3	3.7×10^{-3}	0.080	4.0×10^{-5}	173	0.086	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.1.1–4 and M.5.3.1.1–5 and the MACCS computer code.

Table M.5.3.1.2–4. Pit Disassembly/Conversion Facility Accident Impacts at Pantex Plant

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Fire on the loading dock	0.13	5.1×10^{-5}	0.041	2.0×10^{-5}	6.3	3.2×10^{-3}	5.0×10^{-5}
Fire in a process cell	7.7×10^{-7}	3.1×10^{-10}	2.5×10^{-7}	1.2×10^{-10}	3.8×10^{-5}	1.9×10^{-8}	1.0×10^{-4}
Deflagration inside a glovebox	1.6×10^{-4}	6.4×10^{-8}	5.1×10^{-5}	2.6×10^{-8}	7.9×10^{-3}	3.9×10^{-6}	1.0×10^{-4}
Impact induced spill	2.7×10^{-10}	1.1×10^{-13}	8.7×10^{-11}	4.3×10^{-14}	1.3×10^{-8}	6.7×10^{-12}	4.5×10^{-5}
Nuclear criticality	7.7×10^{-4}	3.1×10^{-7}	2.9×10^{-4}	1.4×10^{-7}	9.5×10^{-3}	4.8×10^{-6}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	5.5×10^{-3}	2.2×10^{-6}	1.7×10^{-3}	8.7×10^{-7}	0.27	1.3×10^{-4}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	8.0	3.3×10^{-3}	2.6	1.3×10^{-3}	393	0.20	1.0×10^{-7}
Beyond evaluation basis earthquake	4.0	1.6×10^{-3}	1.3	6.4×10^{-4}	196	0.098	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.1.1–4 and M.5.3.1.1–5 and the MACCS computer code.

Table M.5.3.1.2–5. Pit Disassembly/Conversion Facility Accident Impacts at Oak Ridge Reservation

	Worker at 772 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Fire on the loading dock	0.37	1.5×10^{-4}	0.37	1.9×10^{-4}	69.3	0.035	5.0×10^{-5}
Fire in a process cell	2.2×10^{-6}	8.9×10^{-10}	2.2×10^{-6}	1.1×10^{-9}	4.2×10^{-4}	2.1×10^{-7}	1.0×10^{-4}
Deflagration inside a glovebox	4.7×10^{-4}	1.9×10^{-7}	4.7×10^{-4}	2.3×10^{-7}	0.087	4.3×10^{-5}	1.0×10^{-4}
Impact induced spill	7.9×10^{-10}	3.2×10^{-13}	7.9×10^{-10}	4.0×10^{-13}	1.5×10^{-7}	7.4×10^{-11}	4.5×10^{-5}
Nuclear criticality	2.0×10^{-3}	7.8×10^{-7}	2.0×10^{-3}	9.8×10^{-7}	0.13	6.6×10^{-5}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	0.016	6.3×10^{-6}	0.016	7.9×10^{-6}	3.0	1.5×10^{-3}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	23.3	1.1×10^{-2}	23.3	0.014	4,320	2.2	1.0×10^{-7}
Beyond evaluation basis earthquake	11.6	4.6×10^{-3}	11.6	5.8×10^{-3}	2,160	1.1	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m or the site boundary [772 m for this facility at ORR], 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.1.1–4 and M.5.3.1.1–5 and the MACCS computer code.

Table M.5.3.1.2–6. Pit Disassembly/Conversion Facility Accident Impacts at Savannah River Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Fire on the loading dock	0.21	8.4×10^{-5}	4.1×10^{-3}	2.0×10^{-6}	19.8	9.9×10^{-3}	5.0×10^{-5}
Fire in a process cell	1.3×10^{-6}	5.0×10^{-10}	2.5×10^{-8}	1.2×10^{-11}	1.2×10^{-4}	5.9×10^{-8}	1.0×10^{-4}
Deflagration inside a glovebox	2.6×10^{-4}	1.0×10^{-7}	5.1×10^{-6}	2.6×10^{-9}	0.025	1.2×10^{-5}	1.0×10^{-4}
Impact induced spill	4.4×10^{-10}	1.8×10^{-13}	8.7×10^{-12}	4.4×10^{-15}	4.2×10^{-8}	2.1×10^{-11}	4.5×10^{-5}
Nuclear criticality	1.1×10^{-3}	4.5×10^{-7}	2.0×10^{-5}	1.0×10^{-8}	0.020	1.0×10^{-5}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	8.9×10^{-3}	3.6×10^{-6}	1.7×10^{-4}	8.7×10^{-8}	0.84	4.2×10^{-4}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	13.0	5.8×10^{-3}	0.26	1.3×10^{-4}	1,240	0.62	1.0×10^{-7}
Beyond evaluation basis earthquake	6.5	2.8×10^{-3}	0.13	6.4×10^{-5}	618	0.31	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.1.1–4 and M.5.3.1.1–5 and the MACCS computer code.

M.5.3.2 Mixed Oxide Fuel Fabrication Facility

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for a MOX fuel fabrication at DOE sites and a generic site in the *Fissile Material Disposition Program PEIS Data Call Input Report: Mixed Oxide Fuel Fabrication.* The studies postulated a set of accident scenarios that were representative of the risks and consequences for workers and the public that can be expected if the facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risks that are expected to envelop the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7}/yr are anticipated that will have consequences and risks larger than those described in this section. [Text deleted.]

M.5.3.2.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the facility. Through a screening process, four evaluation basis accidents and four beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.3.2.1–1 and M.5.3.2.1–2. Accident source term information is provided in Tables M.5.3.2.1–3 through M.5.3.2.1–5. Description of accident scenarios are provided in Table M.5.3.2.1–6.

Table M.5.3.2.1–1. Evaluation Basis Accident Scenarios for the Mixed Oxide Fuel Fabrication Facility

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Fire on open loading dock	1.0×10^{-4} to 1.0×10^{-3}	18 g Pu	0.8 g Pu
Fire in process cell	1.0×10^{-5} to 1.0×10^{-3}	24 g Pu	4.8×10^{-6} g Pu
Leaks or spills from breach of containment	4.5×10^{-5}	4 kg Pu	1.7×10^{-9} g Pu
Explosion inside a glovebox	1.0×10^{-5} to 1.0×10^{-3}	10 kg Pu	1.0×10^{-3} g Pu

Source: LANL 1996b.

Table M.5.3.2.1–2. Beyond Evaluation Basis Accident Scenarios for the Mixed Oxide Fuel Fabrication Facility

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Nuclear criticality	1.0×10^{-7}	5.0×10^{17} fissions; gaseous by-products released[a]	[a]
Beyond evaluation basis fire	1.0×10^{-7}	24 g Pu	0.034 g Pu
Beyond evaluation basis explosion	1.0×10^{-7}	10 kg Pu	50 g Pu
Beyond evaluation basis earthquake	1.0×10^{-7}	10 kg Pu	25 g Pu

[a] See Table M.5.3.2.1–3.
Source: LANL 1996b.

Table M.5.3.2.1–3. Mixed Oxide Fuel Fabrication Facility Criticality Source Terms

Nuclide	Produced (Ci)	Released (Ci)
Kr-83m	5.5	2.75
Kr-85m	3.5	1.75
Kr-85	4.0×10^{-4}	2.0×10^{-4}
Kr-87	21.5	11
Kr-88	11.5	6
Kr-89	650	325
Xe-131m	5.0×10^{-3}	2.5×10^{-3}
Xe-133m	0.1	0.05
Xe-133	1.5	0.75
Xe-135m	165	85
Xe-135	20.5	10
Xe-137	2,450	1,225
Xe-138	550	275
I-131	0.5	0.025
I-132	60	3
I-133	8	0.4
I-134	215	11
I-135	22.5	1.0

Source: LANL 1996b.

Table M.5.3.2.1–4. Mixed Oxide Fuel Fabrication Facility Evaluation Basis Accident Source Terms

	Accident Scenario			
Accident Parameter	**Fire on Loading Dock**	**Fire in Process Cell**	**Leaks or Spills from Breach of Containment**	**Explosion Inside Glovebox**
Frequency of occurrence (per year)	5.0×10^{-4} [a]	1.0×10^{-4} [a]	4.5×10^{-5}	1.0×10^{-4} [a]
Pu released to environment (g)	0.8	4.8×10^{-6}	1.7×10^{-9}	1.0×10^{-3}
Isotope Released to Environment (curies)				
Pu-238	4.50×10^{-4}	2.70×10^{-9}	9.57×10^{-13}	5.63×10^{-7}
Pu-239	0.0462	2.77×10^{-7}	9.83×10^{-11}	5.78×10^{-5}
Pu-240	0.0109	6.53×10^{-8}	2.31×10^{-11}	1.36×10^{-5}
Pu-241	0.0344	2.06×10^{-7}	7.31×10^{-11}	4.30×10^{-5}
Pu-242	3.14×10^{-7}	1.88×10^{-12}	6.66×10^{-16}	3.92×10^{-10}
Am-241	0.0182	1.09×10^{-7}	3.86×10^{-11}	2.27×10^{-5}

[a] Midpoint of the estimated frequency range.
Note: Am=Americium.
Source: Derived from Tables M.5.1.3.4–2 and M.5.3.2.1–1.

Table M.5.3.2.1–5. Mixed Oxide Fuel Fabrication Facility Beyond Evaluation Basis Accident Source Terms

	Accident Scenario			
Accident Parameter	**Nuclear Criticality**	**Beyond Evaluation Basis Fire**	**Beyond Evaluation Basis Explosion**	**Beyond Evaluation Basis Earthquake**
Frequency of occurrence (per year)	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}
Pu released to environment (g)	NA	0.034	50	25
Fissions	5.0×10^{17}	NA	NA	NA
Isotope Released to Environment (Ci)				
Pu-238	0	1.91×10^{-5}	0.0281	0.0141
Pu-239	0	1.97×10^{-3}	2.89	1.44
Pu-240	0	4.62×10^{-4}	0.680	0.340
Pu-241	0	1.46×10^{-3}	2.15	1.08
Pu-242	0	1.33×10^{-8}	1.96×10^{-5}	9.80×10^{-6}
Am-241	0	7.72×10^{-4}	1.13	0.567
Kr-83m	2.75	0	0	0
Kr-85m	1.75	0	0	0
Kr-85	2.0×10^{-4}	0	0	0
Kr-87	11	0	0	0
Kr-88	6	0	0	0
Kr-89	325	0	0	0
Xe-131m	2.5×10^{-3}	0	0	0
Xe-133m	0.05	0	0	0
Xe-133	0.75	0	0	0
Xe-135m	85	0	0	0
Xe-135	10	0	0	0
Xe-137	1,225	0	0	0
Xe-138	275	0	0	0
I-131	0.025	0	0	0
I-132	3	0	0	0
I-133	0.4	0	0	0
I-134	11	0	0	0
I-135	1.0	0	0	0

Note: NA=not applicable.

Source: Derived from Tables M.5.1.3.4–2, M.5.3.2.1–2, and M.5.3.2.1–3.

Table M.5.3.2.1–6. Accident Scenario Descriptions for the Mixed Oxide Fuel Fabrication Facility

Accident Scenario	Accident Description
Evaluation Basis Accidents	
Fire on the loading dock	The fire is caused by welding, cleaning solvents, electrical shorts, or other miscellaneous causes. The scenario assumes an open garage door and that a single drum of combustible waste is involved in the fire.
Fire in a process cell	It is assumed that a process cell contains a glovebox used for final processing of plutonium oxide powder. The gloves, stowed outside the glovebox, are coated with a layer of Pu dust. A flammable cleaning liquid such as acetone or isopropyl alcohol is brought into the process cell in violation of operating procedures, spills, and ignites. The initial extent and intensity of the fire are sufficient to completely incinerate the gloves. The sprinkler system activates and protects the glovebox from further damage. The ventilation system with HEPA filters continues to function throughout the accident.
Deflagration inside a glovebox	The bounding evaluation basis explosion is a deflagration of a flammable gas mixture inside a glovebox. It is assumed that through some unforeseen set of failures, a combustible gas mixture accumulates inside a glovebox and is ignited, possibly by an electrical spark from an operating electrical device. The deflagration blows out the HEPA filter from the glovebox ventilation system exit. Gloves may also be blown out. The room volumes are sufficient to attenuate the pressure wave to levels below that needed to damage the building ventilation system HEPA filters.
Impact-induced spill	The most catastrophic case of leak or spill of nuclear material would result from a forklift or other large vehicle running over a package of nuclear material, breaching the containment, and causing an airborne release to the room. Three-stage HEPA filtration is available for the facility exhaust to limit the release to the environment.
Beyond Evaluation Basis Accidents	
Nuclear criticality	There will not be sufficient quantities of plutonium solutions at the facility to cause a criticality accident. The most likely cause of a criticality event involving Pu oxides would be improper stacking or handling of bulk nuclear material. Multiple operational errors in the material spacing, packing density, manner, and type of containment, and maximum quantities of fissile material permitted in the area would be required for the postulated criticality accident to occur.
Beyond evaluation basis fire in a process cell	A typical fire with coincident failures of two or more major safety systems constitutes a beyond evaluation basis fire. The evaluation postulated the fire in a process cell, discussed above, with the sprinkler system and ventilation system with HEPA filtration inoperative during the accident.
Oxyacetylene explosion in a process cell	The evaluation postulated the explosion of a welding rig oxyacetylene bottle in a process cell. The explosion is sufficient to blow out the HEPA filters and cause significant damage to the ventilation system and nearby equipment.
Beyond evaluation basis earthquake	The following assumptions were used in the evaluation: (1) the earthquake disables the ventilation system; (2) there is significant structural damage to the building but it does not totally collapse; (3) a ceiling slab fall on the glovebox with the most material at risk and severely damages the glovebox; (4) the process cell with the most material at risk is located on an outside wall; (5) the outside wall cracks and the cracks; and (6) the wind is blowing and the cracks are located on the lee side of the building.

Source: LANL 1996b.

M.5.3.2.2 *Accident Impacts*

The estimated impacts of the postulated accidents at each site are provided in Tables M.5.3.2.2–1 through M.5.3.2.2–6. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.3.2.1–4 and M.5.3.2.1–5 using the MACCS computer code. [Text deleted.]

Table M.5.3.2.2–1. Mixed Oxide Fuel Fabrication Facility Accident Impacts at Hanford Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Evaluation basis fire on open loading dock	0.32	1.3×10^{-4}	0.010	5.1×10^{-6}	18.5	9.3×10^{-3}	5.0×10^{-4}
Evaluation basis fire in process cell	1.9×10^{-6}	7.6×10^{-10}	6.1×10^{-8}	3.0×10^{-11}	1.1×10^{-4}	5.5×10^{-8}	1.0×10^{-4}
Leaks or spills from breach of containment	6.8×10^{-10}	2.7×10^{-13}	2.2×10^{-11}	1.1×10^{-14}	3.9×10^{-8}	2.0×10^{-11}	4.5×10^{-5}
Evaluation basis explosion inside a glovebox	4.0×10^{-4}	1.6×10^{-7}	1.3×10^{-5}	6.4×10^{-9}	0.023	1.2×10^{-5}	1.0×10^{-4}
Nuclear criticality	5.2×10^{-4}	2.1×10^{-7}	1.7×10^{-5}	8.4×10^{-9}	3.4×10^{-3}	1.7×10^{-6}	1.0×10^{-7}
Beyond evaluation basis fire	0.014	5.4×10^{-6}	4.3×10^{-4}	2.2×10^{-7}	0.79	3.9×10^{-4}	1.0×10^{-7}
Beyond evaluation basis explosion	19.9	9.4×10^{-3}	0.63	3.2×10^{-4}	1150	0.58	1.0×10^{-7}
Beyond evaluation basis earthquake	9.9	4.1×10^{-3}	0.32	1.6×10^{-4}	576	0.29	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source term in Tables M.5.3.2.1–4 and M.5.3.2.1–5 and the MACCS computer code.

Table M.5.3.2.2–2. Mixed Oxide Fuel Fabrication Facility Accident Impacts at Nevada Test Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Evaluation basis fire on open loading dock	0.22	8.7×10^{-5}	4.0×10^{-3}	2.0×10^{-6}	0.42	2.1×10^{-4}	5.0×10^{-4}
Evaluation basis fire in process cell	1.3×10^{-6}	5.2×10^{-10}	2.4×10^{-8}	1.2×10^{-11}	2.5×10^{-6}	1.3×10^{-9}	1.0×10^{-4}
Leaks or spills from breach of containment	4.6×10^{-10}	1.8×10^{-13}	8.6×10^{-12}	4.3×10^{-15}	8.9×10^{-10}	4.5×10^{-13}	4.5×10^{-5}
Evaluation basis explosion inside a glovebox	2.7×10^{-4}	1.1×10^{-7}	5.1×10^{-6}	2.5×10^{-9}	5.2×10^{-4}	2.6×10^{-7}	1.0×10^{-4}
Nuclear criticality	3.9×10^{-4}	1.5×10^{-7}	6.5×10^{-6}	3.3×10^{-9}	6.6×10^{-5}	3.3×10^{-8}	1.0×10^{-7}
Beyond evaluation basis fire	9.3×10^{-3}	3.7×10^{-6}	1.7×10^{-4}	8.6×10^{-8}	0.018	8.9×10^{-6}	1.0×10^{-7}
Beyond evaluation basis explosion	13.6	6.3×10^{-3}	0.25	1.3×10^{-4}	26.1	0.013	1.0×10^{-7}
Beyond evaluation basis earthquake	6.8	2.7×10^{-3}	0.13	6.3×10^{-5}	13.0	6.5×10^{-3}	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source term in Tables M.5.3.2.1–4 and M.5.3.2.1–5 and the MACCS computer code.

Table M.5.3.2.2–3. Mixed Oxide Fuel Fabrication Facility Accident Impacts at Idaho National Engineering Laboratory

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Evaluation basis fire on open loading dock	0.30	1.2×10^{-4}	2.6×10^{-3}	1.3×10^{-6}	5.5	2.8×10^{-3}	5.0×10^{-4}
Evaluation basis fire in process cell	1.8×10^{-6}	7.1×10^{-10}	1.5×10^{-8}	7.7×10^{-12}	3.3×10^{-5}	1.7×10^{-8}	1.0×10^{-4}
Leaks or spills from breach of containment	6.3×10^{-10}	2.5×10^{-13}	5.5×10^{-12}	2.7×10^{-15}	1.2×10^{-10}	5.9×10^{-12}	4.5×10^{-5}
Evaluation basis explosion inside a glovebox	3.7×10^{-4}	1.5×10^{-7}	3.2×10^{-6}	1.6×10^{-9}	6.9×10^{-3}	3.5×10^{-6}	1.0×10^{-4}
Nuclear criticality	5.0×10^{-4}	2.0×10^{-7}	3.9×10^{-6}	1.9×10^{-9}	8.5×10^{-4}	4.3×10^{-7}	1.0×10^{-7}
Beyond evaluation basis fire	0.013	5.1×10^{-6}	1.1×10^{-4}	5.5×10^{-8}	0.24	1.2×10^{-4}	1.0×10^{-7}
Beyond evaluation basis explosion	18.6	9.2×10^{-3}	0.16	8.0×10^{-5}	346	0.17	1.0×10^{-7}
Beyond evaluation basis earthquake	9.3	3.7×10^{-3}	0.080	4.0×10^{-5}	173	0.086	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source term in Tables M.5.3.2.1–4 and M.5.3.2.1–5 and the MACCS computer code.

Table M.5.3.2.2–4. Mixed Oxide Fuel Fabrication Facility Accident Impacts at Pantex Plant

Accident Scenario	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		Accident Frequency (per year)
	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	
Evaluation basis fire on open loading dock	0.13	5.1×10^{-5}	0.041	2.0×10^{-5}	6.3	3.2×10^{-3}	5.0×10^{-4}
Evaluation basis fire in process cell	7.7×10^{-7}	3.1×10^{-10}	2.4×10^{-7}	1.2×10^{-10}	3.8×10^{-5}	1.9×10^{-8}	1.0×10^{-4}
Leaks or spills from breach of containment	2.7×10^{-10}	1.1×10^{-13}	8.7×10^{-11}	4.3×10^{-14}	1.3×10^{-8}	6.7×10^{-12}	4.5×10^{-5}
Evaluation basis explosion inside a glovebox	1.6×10^{-4}	6.4×10^{-8}	5.1×10^{-5}	2.6×10^{-8}	7.9×10^{-3}	3.9×10^{-6}	1.0×10^{-4}
Nuclear criticality	2.4×10^{-4}	9.7×10^{-8}	9.3×10^{-5}	4.6×10^{-8}	2.3×10^{-3}	1.1×10^{-6}	1.0×10^{-7}
Beyond evaluation basis fire	5.5×10^{-3}	2.2×10^{-6}	1.7×10^{-3}	8.7×10^{-7}	0.27	1.3×10^{-4}	1.0×10^{-7}
Beyond evaluation basis explosion	8.0	3.3×10^{-3}	2.6	1.3×10^{-3}	393	0.20	1.0×10^{-7}
Beyond evaluation basis earthquake	4.0	1.6×10^{-3}	1.3	6.4×10^{-4}	196	9.8×10^{-2}	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source term in Tables M.5.3.2.1–4 and M.5.3.2.1–5 and the MACCS computer code.

Table M.5.3.2.2–5. Mixed Oxide Fuel Fabrication Facility Accident Impacts at Oak Ridge Reservation

	Worker at 801[a]m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Evaluation basis fire on open loading dock	0.36	1.4×10^{-4}	0.36	1.8×10^{-4}	74	0.037	5.0×10^{-4}
Evaluation basis fire in process cell	2.2×10^{-6}	8.7×10^{-10}	2.2×10^{-6}	1.1×10^{-9}	4.4×10^{-4}	2.2×10^{-7}	1.0×10^{-4}
Leaks or spills from breach of containment	7.7×10^{-10}	3.1×10^{-13}	7.7×10^{-10}	3.8×10^{-13}	1.6×10^{-7}	7.9×10^{-1}	4.5×10^{-5}
Evaluation basis explosion inside a glovebox	4.5×10^{-4}	1.8×10^{-7}	4.5×10^{-4}	2.3×10^{-7}	0.092	4.6×10^{-5}	1.0×10^{-4}
Nuclear criticality	5.8×10^{-4}	2.3×10^{-7}	5.8×10^{-4}	2.9×10^{-7}	0.040	2.0×10^{-5}	1.0×10^{-7}
Beyond evaluation basis fire	0.015	6.2×10^{-6}	0.015	7.7×10^{-6}	3.2	1.6×10^{-3}	1.0×10^{-7}
Beyond evaluation basis explosion	22.6	0.011	22.6	0.013	4,620	2.3	1.0×10^{-7}
Beyond evaluation basis earthquake	11.3	4.5×10^{-3}	11.3	5.6×10^{-3}	2,310	1.2	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m or the site boundary [801 m for this facility at ORR], 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source term in Tables M.5.3.2.1–4 and M.5.3.2.1–5 and the MACCS computer code.

Table M.5.3.2.2–6. Mixed Oxide Fuel Fabrication Facility Accident Impacts at Savannah River Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Evaluation basis fire on open loading dock	0.21	8.4×10^{-5}	5.9×10^{-3}	2.9×10^{-6}	19.8	9.9×10^{-3}	5.0×10^{-4}
Evaluation basis fire in process cell	1.3×10^{-6}	5.0×10^{-10}	3.5×10^{-8}	1.8×10^{-11}	1.2×10^{-4}	5.9×10^{-8}	1.0×10^{-4}
Leaks or spills from breach of containment	4.4×10^{-10}	1.8×10^{-13}	1.2×10^{-11}	6.2×10^{-15}	4.2×10^{-8}	2.1×10^{-11}	4.5×10^{-5}
Evaluation basis explosion inside a glovebox	2.6×10^{-4}	1.0×10^{-7}	7.3×10^{-6}	3.7×10^{-9}	0.025	1.2×10^{-5}	1.0×10^{-4}
Nuclear criticality	3.5×10^{-4}	1.4×10^{-7}	9.0×10^{-6}	4.5×10^{-9}	4.6×10^{-3}	2.3×10^{-6}	1.0×10^{-7}
Beyond evaluation basis fire	8.9×10^{-3}	3.6×10^{-6}	2.5×10^{-4}	1.2×10^{-7}	0.84	4.2×10^{-4}	1.0×10^{-7}
Beyond evaluation basis explosion	13.0	5.8×10^{-3}	0.36	1.8×10^{-4}	1,240	0.62	1.0×10^{-7}
Beyond evaluation basis earthquake	6.5	2.8×10^{-3}	0.18	9.1×10^{-5}	618	0.31	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source term in Tables M.5.3.2.1–4 and M.5.3.2.1–5 and the MACCS computer code.

M.5.3.3 Direct Disposition Alternative for a Deep Borehole Complex

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for a deep borehole disposal facility and the direct emplacement of Pu and Plutonium Dioxide in the *Fissile Material Disposition Program Deep Borehole Disposal Facility PEIS Data Input Report for Direct Disposal–Direct Disposal of Plutonium/Plutonium Dioxide in Compound Canisters*. The studies postulated a set of accident scenarios that were representative of the risks and consequences for workers and the public that can be expected if the facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risks that are expected to envelop the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7}/yr are anticipated that will have consequences and risks larger than those described in this section.

M.5.3.3.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the facility. Through a screening process, seven evaluation basis accidents with release to the environment and three beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.3.3.1–1 and M.5.3.3.1–2. Accident source term information is provided in Tables M.5.3.3.1–3 and M.5.3.3.1–4. Descriptions of accident scenarios are provided in Table M.5.3.3.1–5.

Table M.5.3.3.1–1. Evaluation Basis Accident Scenarios for Direct Disposition at the Deep Borehole Complex

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Evaluation basis earthquake	1.0×10^{-6} to 1.0×10^{-4}	NA	No release
Evaluation basis tornado	1.0×10^{-6} to 1.0×10^{-4}	NA	No release
Evaluation basis flood	1.0×10^{-6} to 1.0×10^{-4}	NA	No release
Pu storage container breakage during storage	1.0×10^{-4} to 0.01	4.5 kg Pu	4.5×10^{-10} g Pu
Pu storage container breakage during handling	1.0×10^{-4} to 0.01	4.5 kg Pu	4.5×10^{-8} g Pu
Emplacement canister dropped during handling	1.0×10^{-4} to 0.01	40.5 kg Pu	No release
Onsite emplacement canister transportation accident	1.0×10^{-4} to 0.01	40.5 kg Pu	No release
Nuclear Criticality during emplacement canister filling	1.0×10^{-6} to 1.0×10^{-4}	1.0×10^{19} prompt fissions in 8 hrs; noble gas and halogen fission products release. Release factors: 1.0 noble gas, 0.25 halogen.	[a]
Nuclear Criticality during Pu storage container spill	1.0×10^{-6} to 1.0×10^{-4}	1.0×10^{19} prompt fissions in 8 hrs; noble gas and halogen fission products release. Release factors: 1.0 noble gas, 0.25 halogen.	[a]
Fire in facility process areas	1.0×10^{-6} to 1.0×10^{-4}	40.5 kg Pu	4.05×10^{-5} g Pu
Failure of ventilation filter	0.01 to 0.1	NA	No release
Failure of ventilation blower	0.5	NA	No release
Loss of electrical power	1.0	NA	No release
Canister string dropped during emplacement—ruptured in emplacement zone	1.0×10^{-6} to 1.0×10^{-4}	1,012 kg Pu	4.05×10^{-4} g Pu
Canister string dropped during emplacement—ruptured and stuck in isolation zone	1.0×10^{-6} to 1.0×10^{-4}	1,012 kg Pu	2.43×10^{-7} g Pu
Canister string struck in emplacement zone	1.0×10^{-6} to 1.0×10^{-4}	1,012 kg Pu	No release
Canister string struck in isolation zone	1.0×10^{-6} to 1.0×10^{-4}	1,012 kg Pu	No release
Emplacement facility fire - electrical	1.0×10^{-6} to 1.0×10^{-4}	1,012 kg Pu	No release

[a] See Table M.5.3.3.1–3.

Note: NA=not applicable.

Source: LLNL 1996a.

Table M.5.3.3.1–2. Beyond Evaluation Basis Accident Scenarios for Direct Disposition at the Deep Borehole Complex

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Uncontrolled chemical reaction	$<1.0 \times 10^{-6}$	NA	No release
Pu container nuclear criticality in storage	$<1.0 \times 10^{-6}$	1.0×10^{19} prompt fissions in 8 hr; noble gas and halogen fission products release. Release factors: 1.0 noble gas, 0.25 halogen.	[a]
Emplacement canister nuclear criticality in storage	$<1.0 \times 10^{-6}$	1.0×10^{19} prompt fissions in 8 hr; noble gas and halogen fission products release. Release factors: 1.0 noble gas, 0.25 halogen.	[a]
Nuclear criticality of canister contents at bottom of emplacement zone upon rupture of dropped canister string	$<1.0 \times 10^{-6}$	1.0×10^{19} prompt fissions in 8 hr; noble gas and halogen fission products release. Release factors: 1.0 noble gas, 0.25 halogen.	[a]

[a] See Table M.5.3.3.1–4.
Note: NA=not applicable.
Source: LLNL 1996a.

Table M.5.3.3.1–3. Direct Disposition at the Deep Borehole Complex Evaluation Basis Accident Source Terms

	Accident Scenario						
Accident Parameter	**Pu Storage Container Breakage During Storage**	**Pu Storage Container Breakage During Handling**	**Nuclear Criticality During Emplacement Canister Filling[a]**	**Nuclear Criticality During Pu Storage Container Spill[b]**	**Fire in Facility Process Area**	**Canister String Dropped During Emplacement—Ruptured in Emplacement Zone**	**Canister String Dropped During Emplacement—Ruptured and Stuck in Isolation Zone**
Frequency of occurrence[b] (per year)	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}
Pu released to environment (g)	4.5×10^{-10}	4.5×10^{-8}	–	–	4.05×10^{-5}	4.05×10^{-4}	2.43×10^{-7}
Fissions	–	–	1.0×10^{19}	1.0×10^{19}	–	–	–
Isotope Released to Environment (Ci)							
Pu-238	7.11×10^{-13}	7.11×10^{-11}	0	0	6.40×10^{-8}	6.40×10^{-7}	3.84×10^{-10}
Pu-239	2.57×10^{-11}	2.57×10^{-9}	0	0	2.32×10^{-6}	2.32×10^{-5}	1.39×10^{-8}
Pu-240	6.84×10^{-12}	6.84×10^{-10}	0	0	6.16×10^{-7}	6.16×10^{-6}	3.69×10^{-9}
Pu-241	2.43×10^{-11}	2.43×10^{-9}	0	0	2.18×10^{-6}	2.18×10^{-5}	1.31×10^{-8}
Pu-242	1.00×10^{-15}	1.00×10^{-13}	0	0	9.03×10^{-11}	9.03×10^{-10}	5.42×10^{-13}
Am-241	1.28×10^{-13}	1.28×10^{-11}	0	0	1.15×10^{-8}	1.15×10^{-7}	6.90×10^{-11}
Kr-83m	0	0	110	110	0	0	0
Kr-85m	0	0	71	71	0	0	0
Kr-85	0	0	8.1×10^{-4}	8.1×10^{-4}	0	0	0
Kr-87	0	0	430	430	0	0	0
Kr-88	0	0	230	230	0	0	0
Kr-89	0	0	1.3×10^{-4}	1.3×10^{-4}	0	0	0
Xe-131m	0	0	0.1	0.1	0	0	0
Xe-133m	0	0	2.2	2.2	0	0	0
Xe-133	0	0	27	27	0	0	0
Xe-135m	0	0	3.3×10^{3}	3.3×10^{3}	0	0	0
Xe-135	0	0	410	410	0	0	0
Xe-137	0	0	4.9×10^{4}	4.9×10^{4}	0	0	0
Xe-138	0	0	1.1×10^{4}	1.1×10^{4}	0	0	0
I-131	0	0	2.75	2.75	0	0	0
I-132	0	0	300	300	0	0	0
I-133	0	0	40	40	0	0	0
I-134	0	0	1.08×10^{3}	1.08×10^{3}	0	0	0
I-135	0	0	113	113	0	0	0

[a] Curies produced (by isotope) for the 1.0×10^{19} fission criticality were scaled from Table M.5.3.1.1–3.

[b] Midpoint of the estimated frequency range.

Source: Derived from Tables M.5.1.3.4–1, M.5.3.1.1–3, and M.5.3.3.1–1.

Table M.5.3.3.1–4. Direct Disposition at the Deep Borehole Complex Beyond Evaluation Basis Accident Source Terms

	Accident Scenario		
Accident Parameter	**Pu Container Nuclear Criticality in Storage[a]**	**Emplacement Canister Nuclear Criticality in Storage[a]**	**Nuclear Criticality of Canister Contents at Bottom of Emplacement Zone Upon Rupture of Dropped Canister String[a]**
Frequency of occurrence (per year)	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}
Pu released to environment	NA	NA	NA
Fissions	1.0×10^{19}	1.0×10^{19}	1.0×10^{19}
Isotope Released to Environment (Ci)			
[Text deleted.]			
Kr-83m	110	110	110
Kr-85m	71	71	71
Kr-85	8.1×10^{-4}	8.1×10^{-4}	8.1×10^{-4}
Kr-87	430	430	430
Kr-88	230	230	230
Kr-89	1.3×10^{-4}	1.3×10^{-4}	1.3×10^{-4}
Xe-131m	0.1	0.1	0.1
Xe-133m	2.2	2.2	2.2
Xe-133	27	27	27
Xe-135m	3.3×10^{3}	3.3×10^{3}	3.3×10^{3}
Xe-135	410	410	410
Xe-137	4.9×10^{4}	4.9×10^{4}	4.9×10^{4}
Xe-138	1.1×10^{4}	1.1×10^{4}	1.1×10^{4}
I-131	2.75	2.75	2.75
I-132	300	300	300
I-133	40	40	40
I-134	1.08×10^{3}	1.08×10^{3}	1.08×10^{3}
I-135	113	113	113

[a] Curies produced (by isotope) for the 1.0×10^{19} fission criticality were scaled from Table M.5.3.1.1–3.
Note: NA=not applicable.
Source: Derived from Tables M.5.1.3.4–1, M.5.3.1.1–3, and M.5.3.3.1–2.

Table M.5.3.3.1–5. Accident Scenario Descriptions for Direct Disposition at the Deep Borehole Complex

Accident Scenario	Accident Description
Evaluation Basis Accidents	
Pu storage container breakage during storage	It is postulated that a Pu storage container is ruptured due to overpressurization of the container. The overpressurization could occur as a result of volume expansion caused by either complete oxidation of Pu metal buttons stored in cans or pressure buildup due to radiolysis of residual moisture in Pu oxide and helium gas from the alpha decay of Pu and daughter products. Respirable Pu fines are released to the storage area.
Pu storage container breakage during handling	It is postulated that a 2R Pu container is dropped and breaches in container handling operations. The force of the drop ruptures the container and respirable oxide fines are released to the process area.
Nuclear criticality during emplacement canister filling	Mishandling of the Pu containers during handling operations could lead to a criticality accident. At least three independent and concurrent equipment failures or operation errors must occur before a criticality accident could occur. It is postulated that additional Pu containers are introduced into the emplacement canister filling process area in violation of procedural controls and a criticality occurs as a result of the containers being spaced too closely.
Nuclear criticality during Pu storage canister spill	A nuclear criticality could occur if Pu containers were damaged in handling and the mass of the spilled Pu oxide containers exceeds the critical mass. Because each 2R primary container contains a limited amount of Pu, a criticality accident would require successively damaging several containers.
Fire in process area	The combustible loading in the process area is very low because the process does not involve any combustible materials. However, it is postulated that a large fire occurs in the process area for emplacement canister filling, the containers are breached by the fire, and the contents are exposed to the fire. The ventilation system two-stage HEPA filters are operational during the fire.
Canister string dropped during emplacement, ruptured in emplacement zone	A canister string could be dropped into the borehole as a result of either a structural failure in the crane and associated hoisting and securing equipment or as a result of operator error. A free-falling canister string could rupture upon impact at the bottom of the borehole.
Canister string dropped during emplacement, ruptured and stuck in isolation zone	A canister string could be dropped into the borehole as a result of either a structural failure in the crane and associated hoisting and securing equipment or as a result of operator error. The canister string impacts a projecting ledge at a change in the diameter of the well casings, ruptures, and remains stuck in the isolation zone instead of falling to the bottom of the borehole.
Beyond Evaluation Basis Accidents	
Pu container nuclear criticality in storage	The Pu storage facility is designed to ensure that an accidental criticality during dry or flood conditions is not credible. The assumed criticality accident severity is based on guidance provided in NRC Regulatory Guide 3.35.
Emplacement canister nuclear criticality in storage	The storage racks are designed to maintain the geometry of the array under all postulated accident and natural conditions. The assumed criticality accident severity is based on guidance provided in NRC Regulatory Guide 3.35.
Nuclear criticality of canister contents at bottom of emplacement zone upon rupture of dropped canister string	A canister string could be dropped into the borehole as a result of either a structural failure in the crane and associated hoisting and securing equipment or as a result of operator error. A free-falling canister string could rupture upon impact at the bottom of the borehole. The evaluation assumed that Pu released from the ruptured string would collect in a critical mass at the bottom of the borehole. The assumed criticality accident severity is based on guidance provided in NRC Regulatory Guide 3.35.

Source: LLNL 1996a.

M.5.3.3.2 *Accident Impacts*

The estimated range of impacts of the postulated accidents at reference sites is provided in Table M.5.3.3.2–1. The estimated range of environmental data (wet to dry site) and the general public population density data (low to high density) for the reference sites envelope the site characteristics expected for the direct disposition site. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.3.3.1–3 and M.5.3.3.1–4 using the MACCS computer code. [Text deleted.]

[Text deleted.]

Table M.5.3.3.2–1. Direct Disposition at the Deep Borehole Complex Accident Impacts Ranges at Generic Sites

Accident Scenario	Worker at 1,000 m				Maximum Offsite Individual				Population to 80 km				Accident Frequency (per year)
	Dose (rem)		Probability of Cancer Fatality[a]		Dose (rem)		Probability of Cancer Fatality[a]		Dose (person-rem)		Number of Cancer Fatalities[b]		
	High	Low	High	Low	High	Low	High	Low	High	High	Low	Low	
Pu storage container breakage during storage	1.3×10^{-10}	5.3×10^{-11}	5.1×10^{-14}	2.1×10^{-14}	2.1×10^{-11}	9.4×10^{-13}	1.0×10^{-14}	4.7×10^{-16}	1.8×10^{-8}	1.7×10^{-10}	9×10^{-12}	8.4×10^{-14}	1.0×10^{-3}
Pu storage container breakage during handling	1.3×10^{-8}	5.3×10^{-9}	5.1×10^{-12}	2.1×10^{-12}	2.1×10^{-9}	9.4×10^{-11}	1.0×10^{-12}	4.7×10^{-14}	1.8×10^{-6}	1.7×10^{-8}	9.0×10^{-10}	8.4×10^{-12}	1.0×10^{-3}
Nuclear criticality during emplacement canister filling	3.5×10^{-2}	1.6×10^{-2}	1.4×10^{-5}	6.2×10^{-6}	5.8×10^{-3}	2.0×10^{-4}	2.9×10^{-6}	1.0×10^{-7}	1.3	4.6×10^{-2}	6.3×10^{-4}	2.3×10^{-5}	1.0×10^{-5}
Nuclear criticality during Pu storage canister spill	3.5×10^{-2}	1.6×10^{-2}	1.4×10^{-5}	6.2×10^{-6}	5.8×10^{-3}	2.0×10^{-4}	2.9×10^{-6}	1.0×10^{-7}	1.2	6.6×10^{-3}	6.0×10^{-4}	3.3×10^{-6}	1.0×10^{-5}
Fire in process area	1.2×10^{-5}	4.7×10^{-6}	4.6×10^{-9}	1.9×10^{-9}	1.9×10^{-6}	8.4×10^{-8}	9.3×10^{-10}	4.2×10^{-11}	1.6×10^{-3}	6.6×10^{-3}	8.1×10^{-7}	3.3×10^{-6}	1.0×10^{-5}
Canister string dropped during emplacement, ruptured in emplacement zone	1.2×10^{-4}	4.7×10^{-5}	4.6×10^{-8}	1.9×10^{-8}	1.9×10^{-5}	8.5×10^{-7}	9.3×10^{-9}	4.2×10^{-12}	1.6×10^{-2}	1.5×10^{-4}	8.1×10^{-6}	7.6×10^{-8}	1.0×10^{-5}
Canister string dropped during emplacement, ruptured in isolation zone	6.9×10^{-8}	2.8×10^{-8}	2.8×10^{-11}	1.1×10^{-11}	1.1×10^{-8}	5.0×10^{-10}	5.6×10^{-12}	2.5×10^{-13}	9.7×10^{-6}	9.0×10^{-8}	4.9×10^{-9}	4.5×10^{-11}	1.0×10^{-5}
Pu container nuclear criticality in storage	3.5×10^{-2}	1.6×10^{-2}	1.4×10^{-5}	6.2×10^{-6}	5.8×10^{-3}	2.0×10^{-4}	2.9×10^{-6}	1.0×10^{-7}	1.3	6.6×10^{-3}	6.3×10^{-4}	3.3×10^{-6}	1.0×10^{-6}
Emplacement canister nuclear criticality in storage	3.5×10^{-2}	1.6×10^{-2}	1.4×10^{-5}	6.2×10^{-6}	5.8×10^{-3}	2.0×10^{-4}	2.9×10^{-6}	1.0×10^{-7}	1.3	6.6×10^{-3}	6.3×10^{-4}	3.3×10^{-6}	1.0×10^{-6}
Nuclear criticality of canister contents at bottom of emplacement zone upon rupture of dropped canister string	3.5×10^{-2}	1.6×10^{-2}	1.4×10^{-5}	6.2×10^{-6}	5.8×10^{-3}	2.0×10^{-4}	2.9×10^{-6}	1.0×10^{-7}	1.3	6.6×10^{-3}	6.3×10^{-4}	3.3×10^{-6}	1.0×10^{-6}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Source: Calculated using the source terms in Tables M.5.3.3.1–3 and M.5.3.3.1–4 and the MACCS computer code.

M.5.3.4 Immobilized Disposition Alternative for a Deep Borehole Complex

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for a deep borehole immobilized disposal facility in the *Fissile Material Disposition Program: Deep Borehole Disposal Facility PEIS Data Input Report for Immobilized Disposal—Immobilized Disposal of Plutonium in Coated Ceramic Pellets in Grout Without Canisters*. The studies postulated a set of accidents scenarios that were representative of the risks and consequences for workers and the public that can be expected if the facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risk that are expected to envelop the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7}/yr are anticipated that will have consequences and risks larger than those described in this section.

M.5.3.4.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the facility. Through a screening process, 14 evaluation basis accidents and 4 beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.3.4.1–1 and M.5.3.4.1–2. Accident source term information is provided in Tables M.5.3.4.1–3 and M.5.3.4.1–4. Descriptions of accident scenarios are provided in Table M.5.3.4.1–5.

Table M.5.3.4.1–1. Evaluation Basis Accident Scenarios for Immobilized Disposition at the Deep Borehole Complex

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Earthquake	$1.0x10^{-6}$ to $1.0x10^{-4}$	5 kg Pu	$5.0x10^{-10}$ g Pu
Tornado	$1.0x10^{-6}$ to $1.0x10^{-4}$	NA	No release
Flood	$1.0x10^{-6}$ to $1.0x10^{-4}$	NA	No release
Pu storage container breakage	$1.0x10^{-4}$ to 0.01	5 kg Pu	$5.0x10^{-12}$ g Pu
Pu storage container breach	$1.0x10^{-4}$ to 0.01	5 kg Pu	$5.0x10^{-12}$ g Pu
Onsite pellet transporter accident	$1.0x10^{-4}$ to 0.01	5 kg Pu	No release
Pellet-grout mixing process facility fire	$1.0x10^{-6}$ to $1.0x10^{-4}$	5 kg Pu	$5.0x10^{-10}$ g Pu
Ceramic pellet spill	$1.0x10^{-4}$ to 0.01	0.5 kg Pu	$5.0x10^{-13}$ g Pu
Pellet-grout mix spill	0.01 to 0.1	0.5 kg Pu	$3.0x10^{-11}$ g Pu
Failure of ventilation blower	0.01 to 0.1	NA	No release
Loss of electrical power	0.01 to 0.1	NA	No release
Bucket Emplacement			
Bucket dropped during emplacement	$1.0x10^{-6}$ to $1.0x10^{-4}$	834 kg Pu	$5.0x10^{-7}$ g Pu
Bucket stuck in the isolation zone	$1.0x10^{-6}$ to $1.0x10^{-4}$	834 kg Pu	No release
Bucket stuck in emplacement zone	$1.0x10^{-6}$ to $1.0x10^{-4}$	834 kg Pu	No release
Failure of release—fails to open	$1.0x10^{-6}$ to $1.0x10^{-4}$	834 kg Pu	No release
Failure of release—opens early during bucket emplacement	$1.0x10^{-6}$ to $1.0x10^{-4}$	834 kg Pu	$2.5x10^{-6}$ g Pu
Pellet-grout sets in bucket	$1.0x10^{-6}$ to $1.0x10^{-4}$	834 kg Pu	No release
Mixing system breaks pellets during bucket emplacement	$1.0x10^{-4}$ to 0.01	834 kg Pu	$5.0x10^{-8}$ g Pu
Pellets break during bucket emplacement release	$1.0x10^{-4}$ to 0.01	834 kg Pu	$5.0x10^{-8}$ g Pu
Emplacement facility fire - combustibles	$1.0x10^{-6}$ to $1.0x10^{-4}$	834 kg Pu	No release
Emplacement facility fire - electrical	$1.0x10^{-6}$ to $1.0x10^{-4}$	834 kg Pu	No release
Loss of electrical power	0.01 to 0.1	NA	No release
Pumped Emplacement			
Rupture of delivery pipe during pumped emplacement	$1.0x10^{-6}$ to $1.0x10^{-4}$	100 kg Pu	$3.0x10^{-7}$ g Pu
Pellet-grout solidifies in delivery pipe	$1.0x10^{-4}$ to 0.01	100 kg Pu	No release
Delivery pipe dropped during pumped emplacement	$1.0x10^{-6}$ to $1.0x10^{-4}$	100 kg Pu	$6.0x10^{-8}$ g Pu
Delivery pipe stuck in the borehole	$<1.0x10^{-6}$	100 kg Pu	No release
Mixing system breaks pellets during pumped emplacement	$1.0x10^{-4}$ to 0.01	100 kg Pu	$6.0x10^{-9}$ g Pu
Pellets break during release during pumped emplacement	$1.0x10^{-4}$ to 0.01	100 kg Pu	$6.0x10^{-9}$ g Pu

Table M.5.3.4.1–1. Evaluation Basis Accident Scenarios for Immobilized Disposition at the Deep Borehole Complex—Continued

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Emplacement facility fire—combustibles	1.0×10^{-6} to 1.0×10^{-4}	100 kg Pu	No release
Emplacement facility fire—electrical	1.0×10^{-6} to 1.0×10^{-4}	100 kg Pu	No release
Loss of electrical power	0.01 to 0.1	NA	No release

Note: NA=not applicable.

Source: LLNL 1996h.

Table M.5.3.4.1–2. Beyond Evaluation Basis Accident Scenarios for Immobilized Disposition at the Deep Borehole Complex

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Failure of ventilation filter	$<1.0 \times 10^{-6}$	0.5 kg Pu	3.0×10^{-9} g Pu
Uncontrolled chemical reaction	$<1.0 \times 10^{-6}$	5 kg Pu	5.0×10^{-9} g Pu
Pellet storage nuclear criticality	$<1.0 \times 10^{-6}$	1.0×10^{19} prompt fissions in 8 hrs; noble gas and halogen fission products release. Release factors: 1.0 noble gas, 0.25 halogen	[a]
Pellet-grout mixing nuclear criticality	$<1.0 \times 10^{-6}$	1.0×10^{19} prompt fissions in 8 hrs; noble gas and halogen fission products release. Release factors: 1.0 noble gas, 0.25 halogen.	[a]

[a] See Table M.5.3.4.1–4.

Source: LLNL 1996h.

Table M.5.3.4.1–3. Immobilized Disposition at the Deep Borehole Complex Evaluation Basis Accident Source Terms

Accident Parameter	Accident Scenario						
	Earthquake	Pu Storage Container Breakage	Pu Storage Container Breach	Pellet-Grout Mixing Process Facility Fire	Ceramic Pellet Spill	Pellet-Grout Mix Spill	Dropped Bucket During Emplacement
Frequency of occurrence (per year)	1.0×10^{-5}	1.0×10^{-3}	1.0×10^{-3}	1.0×10^{-5}	1.0×10^{-3}	0.05	1.0×10^{-5}
Pu released to environment (g)	5.0×10^{-10}	5.0×10^{-12}	5.0×10^{-12}	5.0×10^{-10}	5.0×10^{-13}	3.0×10^{-11}	5.0×10^{-7}
Isotope Released to Environment (Ci)							
Pu-238	7.90×10^{-13}	7.90×10^{-15}	7.90×10^{-15}	7.90×10^{-13}	7.90×10^{-16}	4.74×10^{-14}	7.90×10^{-10}
Pu-239	2.86×10^{-11}	2.86×10^{-13}	2.86×10^{-13}	2.86×10^{-11}	2.86×10^{-14}	1.72×10^{-12}	2.86×10^{-8}
Pu-240	7.60×10^{-12}	7.60×10^{-14}	7.60×10^{-14}	7.60×10^{-12}	7.60×10^{-15}	4.56×10^{-13}	7.60×10^{-9}
Pu-241	2.69×10^{-11}	2.69×10^{-13}	2.69×10^{-13}	2.69×10^{-11}	2.69×10^{-14}	1.62×10^{-12}	2.69×10^{-8}
Pu-242	1.11×10^{-15}	1.12×10^{-17}	1.12×10^{-17}	1.11×10^{-15}	1.12×10^{-18}	6.69×10^{-17}	1.12×10^{-12}
Am-241	1.42×10^{-13}	1.42×10^{-15}	1.42×10^{-15}	1.42×10^{-13}	1.42×10^{-16}	8.52×10^{-15}	1.42×10^{-10}

Table M.5.3.4.1–3. Immobilized Disposition at the Deep Borehole Complex Evaluation Basis Accident Source Terms—Continued

Accident Parameter	Accident Scenario						
	Failure of Release - Opens Early During Bucket Emplacement	Mixing System Breaks Pellets During Bucket Emplacement	Pellets Break During Bucket Emplacement Release	Rupture of Delivery Pipe During Pumped Emplacement	Delivery Pipe Dropped During Pumped Emplacement	Mixing System Breaks Pellets During Pumped Emplacement	Pellets Break During Pumped Emplacement Release
Frequency of occurrence[a](per year)	$1.0x10^{-5}$	$1.0x10^{-3}$	$1.0x10^{-3}$	$1.0x10^{-5}$	$1.0x10^{-5}$	$1.0x10^{-3}$	$1.0x10^{-3}$
Pu released to environment (g)	$2.5x10^{-6}$	$5.0x10^{-8}$	$5.0x10^{-8}$	$3.0x10^{-7}$	$6.0x10^{-8}$	$6.0x10^{-9}$	$6.0x10^{-9}$
Isotope Released to Environment (Ci)							
Pu-238	$3.95x10^{-9}$	$7.90x10^{-11}$	$7.90x10^{-11}$	$4.74x10^{-10}$	$9.48x10^{-11}$	$9.48x10^{-12}$	$9.48x10^{-12}$
Pu-239	$1.43x10^{-7}$	$2.86x10^{-9}$	$2.86x10^{-9}$	$1.72x10^{-8}$	$3.43x10^{-9}$	$3.43x10^{-10}$	$3.43x10^{-10}$
Pu-240	$3.80x10^{-8}$	$7.60x10^{-10}$	$7.60x10^{-10}$	$4.56x10^{-9}$	$9.12x10^{-10}$	$9.12x10^{-11}$	$9.12x10^{-11}$
Pu-241	$1.35x10^{-7}$	$2.70x10^{-9}$	$2.70x10^{-9}$	$1.62x10^{-8}$	$3.23x10^{-9}$	$3.23x10^{-10}$	$3.23x10^{-10}$
Pu-242	$5.58x10^{-12}$	$1.12x10^{-13}$	$1.12x10^{-13}$	$6.69x10^{-13}$	$1.34x10^{-13}$	$1.34x10^{-14}$	$1.34x10^{-15}$
Am-241	$7.10x10^{-10}$	$1.42x10^{-11}$	$1.42x10^{-11}$	$8.52x10^{-11}$	$1.70x10^{-11}$	$1.70x10^{-12}$	$1.70x10^{-12}$

[a]Midpoint of the estimated frequency range.

Note: Am=Americium.

Source: Derived from Tables M.5.1.3.4–1 and M.5.3.4.1–1.

Table M.5.3.4.1–4. Immobilized Disposition at the Deep Borehole Complex Beyond Evaluation Basis Accident Source Terms

	Accident Scenario			
Accident Parameter	**Failure of Ventilation Filter**	**Uncontrolled Chemical Reaction**	**Pellet Storage Nuclear Criticality[a]**	**Pellet-Grout Mixing Process Nuclear Criticality**
Frequency of occurrence (per year)	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}
Pu released to environment (g)	3.0×10^{-9}	5.0×10^{-9}	NA	NA
Fissions	NA	NA	1.0×10^{19}	1.0×10^{19}
Isotope Released to Environment (Ci)				
Pu-238	4.74×10^{-12}	7.90×10^{-12}	0	0
Pu-239	1.72×10^{-10}	2.86×10^{-10}	0	0
Pu-240	4.56×10^{-11}	7.60×10^{-11}	0	0
Pu-241	1.62×10^{-10}	2.70×10^{-10}	0	0
Pu-242	6.69×10^{-15}	1.12×10^{-14}	0	0
Am-241	8.52×10^{-13}	1.42×10^{-12}	0	0
Kr-83m	0	0	110	110
Kr-85m	0	0	71	71
Kr-85	0	0	8.1×10^{-4}	8.1×10^{-4}
Kr-87	0	0	430	430
Kr-88	0	0	230	230
Kr-89	0	0	1.3×10^{-4}	1.3×10^{-4}
Xe-131m	0	0	0.1	0.1
Xe-133m	0	0	2.2	2.2
Xe-133	0	0	27	27
Xe-135m	0	0	3.3×10^{3}	3.3×10^{3}
Xe-135	0	0	410	410
Xe-137	0	0	4.9×10^{4}	4.9×10^{4}
Xe-138	0	0	1.1×10^{4}	1.1×10^{4}
I-131	0	0	2.75	2.75
I-132	0	0	300	300
I-133	0	0	40	40
I-134	0	0	1.08×10^{3}	1.08×10^{3}
I-135	0	0	113	113

[a] Curies produced (by isotope) for the 1.0×10^{19} fission criticality were scaled from Table M.5.3.1.1–3.

Note: NA=not applicable.

Source: Derived from Tables M.5.1.3.4–1, M.5.3.1.1–3, and M.5.3.4.1–2.

Table M.5.3.4.1–5. Accident Scenario Descriptions for Immobilized Disposition at the Deep Borehole Complex

Accident Scenario	Accident Description
Evaluation Basis Accidents	
Earthquake	It was postulated that the evaluation basis earthquake would rupture the ceramic pellet grouting vessel and lines. The Pu-containing particulate would be removed from the grouting area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Pu storage container breakage	It is postulated that a container breakage could occur in ceramic pellet storage. Respirable fines of ceramic are released to the storage area and collected by the ventilation system. The airborne fines pass through the ventilation HEPA filters and are released to the environment.
Pu storage container breach	It is postulated that a container breach could occur in ceramic pellet container handling operations. A container is punctured during handling and ceramic pellets spill from the punctured container. Respirable fines of ceramic are released to the process area and collected by the ventilation system. The airborne fines pass through the ventilation HEPA filters and are released to the environment.
Pellet-grout mixing process facility fire	It is postulated that an unimpeded fire begins in the process area which houses the grouting vessel. The fire breaches the vessel enclosure that contains the Pu-loaded ceramic pellets. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Ceramic pellet spill	It is postulated that the ceramic pellets overflow the grouting feed bin and spill onto the floor. The spill spreads out in a safe geometry. The spill is cleaned up in two hours but some of the spill material converts to an aerosol and becomes airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Pellet-grout mix spill	It is postulated that the grouting vessel or the bucket overflows and spills onto the floor. The spill spreads out in a safe geometry. The spill is cleaned up in 2 hours but some of the spill material converts to an aerosol and becomes airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Dropped bucket during emplacement	A bucket could be dropped into the borehole as a result of either a structural failure in the crane, the associated hoisting and securing equipment, or as a result of operator error. A free-falling bucket could rupture upon impact at the bottom of the borehole.
Failure of release—opens early during bucket emplacement	The valve at the bottom of the bucket opens prematurely and the pellets and the cement free fall to the bottom of the borehole. This would probably result in some broken or fractured pellets.
Mixing system breaks pellets during emplacement	The pellets are mixed with cement and pushed with water, air pressure, or gravity into the bucket. It is postulated that some of the pellets may break or crack due to unforeseen events in the emplacement process.
Pellets break during bucket emplacement release	Upon release, the pellets and cement will flow out into the borehole. The weight of the column in the bucket and the pressure that will likely be needed to push out the mix could cause some of the pellets to break due to some unforeseen events in the emplacement process.
Rupture of delivery pipe during pumped emplacement	If the delivery pipe were to rupture, the pellets and cement would free fall to the bottom of the borehole. This would probably result in some broken or fractured pellets.

Table M.5.3.4.1–5. Accident Scenario Descriptions for Immobilized Disposition at the Deep Borehole Complex—Continued

Accident Scenario	Accident Description
Delivery pipe dropped during pumped emplacement	A delivery pipe could be dropped into the borehole as a result of either a structural failure in the crane or drill rig, or as a result to operator error. Substantial quantities of ceramic pellets could be broken or cracked upon impact at the bottom of the borehole.
Mixing system breaks pellets during pumped emplacement	The pellets are mixed with cement and pushed with water, air pressure, or gravity into the delivery pipe. It is postulated that some of the pellets may break or crack due to unforeseen events in the process.
Pellets break during pumped emplacement release	Upon release from the end of the delivery pipe, the pellets and cement will flow out into the borehole. The weight of the column in the pipe and the pressure that will likely be needed to push out the mix could cause some of the pellets to break due to some unforeseen events in the emplacement process.
Beyond Evaluation Basis Accidents	
Failure of ventilation filter	A HEPA filter could fail due to moisture collection on the filter, excessive pressure loading from exhaust blower, excessive heat from a fire, or mechanical shock. It is postulated that the HEPA filter servicing the grout mixing process fails concurrently with a grouting process spill accident. Some of the spill material converts to an aerosol and becomes airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system, pass through the failed HEPA filters and be released to the environment.
Uncontrolled chemical reaction	It is postulated that hydrogen produced in the battery of the uninterruptible power system detonates in the grout mix vessel area, fractures pellets in the process, and some of the fractured pellets becomes airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Pellet storage nuclear criticality	The designed Pu concentration in the ceramic pellet is sufficiently low to maintain criticality safe under all postulated accidents and natural conditions. The facility is designed to preclude flooding in the storage area. A nuclear criticality accident in the pellet storage vault area is not credible. However, a criticality accident was postulated, and the assumed criticality accident severity is based on guidance provided in NRC Regulatory Guide 3.35.
Pellet-grout mixing process nuclear criticality	The designed Pu concentration in the ceramic pellet is sufficiently low to maintain criticality safe under all postulated accidents during grout mixing process conditions. A nuclear criticality accident in the pellet storage vault area is not credible. However, a criticality accident was postulated, and the assumed criticality accident severity is based on guidance provided in NRC Regulatory Guide 3.35.

Source: LLNL 1996a.

M.5.3.4.2 *Accident Impacts*

The estimated range of impacts of the postulated accidents at reference sites are provided in Table M.5.3.4.2–1. The estimated range of environmental data (wet to dry site) and the general public population density data (low to high density) for the reference sites envelop the site characteristics expected for the emplacement site. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.3.4.1–3 and M.5.3.4.1–4 using the MACCS computer code.

[Text deleted.]

Table M.5.3.4.2–1. Immobilized Disposition at the Deep Borehole Complex Accident Impacts Ranges at Generic Site

	Worker at 1,000 m				Maximum Offsite Individual				Population to 80 km				
	Dose (rem)		Probability of Cancer Fatality[a]		Dose (rem)		Probability of Cancer Fatality[a]		Dose (person-rem)		Number of Cancer Fatalities[b]		Accident Frequency (per year)
Accident Scenario	**High**	**Low**	**High**	**Low**	**High**	**Low**	**High**	**Low**	**High**	**High**	**Low**	**Low**	
Earthquake	1.4×10^{-10}	5.8×10^{-11}	5.7×10^{-14}	2.3×10^{-14}	2.3×10^{-11}	1.0×10^{-12}	1.2×10^{-14}	$5./2\times10^{-16}$	2.0×10^{-8}	1.9×10^{-10}	1.0×10^{-11}	9.3×10^{-14}	1.0×10^{-5}
Pu storage container breakage	1.4×10^{-12}	5.8×10^{-13}	5.7×10^{-16}	2.3×10^{-16}	2.3×10^{-13}	1.0×10^{-14}	1.2×10^{-16}	5.2×10^{-18}	2.0×10^{-10}	1.9×10^{-12}	1.0×10^{-13}	9.3×10^{-16}	1.0×10^{-3}
Pu storage container breach	1.4×10^{-12}	5.8×10^{-13}	5.7×10^{-16}	2.3×10^{-16}	2.3×10^{-13}	1.0×10^{-14}	1.2×10^{-16}	5.2×10^{-18}	2.0×10^{-10}	1.9×10^{-12}	1.0×10^{-13}	9.3×10^{-16}	1.0×10^{-3}
Pellet-grout mixing process facility fire	1.4×10^{-10}	5.8×10^{-11}	5.7×10^{-14}	2.3×10^{-14}	2.3×10^{-11}	1.0×10^{-12}	1.2×10^{-14}	5.2×10^{-16}	2.0×10^{-8}	1.9×10^{-10}	1.0×10^{-11}	9.3×10^{-14}	1.0×10^{-5}
Ceramic pellet spill	1.4×10^{-13}	5.8×10^{-14}	5.7×10^{-17}	2.3×10^{-17}	2.3×10^{-14}	1.0×10^{-15}	1.2×10^{-17}	5.2×10^{-19}	2.0×10^{-11}	1.9×10^{-13}	1.0×10^{-14}	9.3×10^{-17}	1.0×10^{-3}
Pellet grout mix spill	8.5×10^{-12}	3.4×10^{-12}	3.4×10^{-15}	1.4×10^{-15}	1.4×10^{-12}	6.2×10^{-14}	6.9×10^{-16}	3.1×10^{-17}	1.2×10^{-9}	1.1×10^{-11}	6.0×10^{-13}	5.5×10^{-15}	5.0×10^{-2}
Bucket dropped during emplacement	1.4×10^{-7}	5.8×10^{-8}	5.7×10^{-11}	2.3×10^{-11}	2.3×10^{-8}	1.0×10^{-9}	1.2×10^{-11}	5.2×10^{-13}	2.0×10^{-5}	1.9×10^{-7}	1.0×10^{-8}	9.3×10^{-11}	1.0×10^{-5}
Failure of release - opens early during bucket emplacement	7.1×10^{-7}	2.8×10^{-7}	2.8×10^{-10}	1.1×10^{-10}	1.2×10^{-7}	5.1×10^{-9}	5.8×10^{-11}	2.6×10^{-12}	1.0×10^{-4}	$9.1\times10x^{-7}$	5.0×10^{-8}	4.6×10^{-10}	1.0×10^{-5}
Mixing system breaks pellets during emplacement	1.4×10^{-8}	5.8×10^{-9}	5.7×10^{-12}	2.3×10^{-12}	2.3×10^{-9}	1.0×10^{-10}	1.2×10^{-12}	5.2×10^{-14}	2.0×10^{-6}	1.9×10^{-8}	1.0×10^{-9}	9.3×10^{-12}	1.0×10^{-3}
Pellets break during bucket emplacement release	1.4×10^{-8}	5.8×10^{-9}	5.7×10^{-12}	2.3×10^{-12}	2.3×10^{-9}	1.0×10^{-10}	1.2×10^{-12}	5.2×10^{-14}	2.0×10^{-6}	1.9×10^{-8}	1.0×10^{-9}	9.3×10^{-12}	1.0×10^{-3}

Table M.5.3.4.2–1. Immobilized Disposition at the Deep Borehole Complex Accident Impacts Ranges at Generic Site—Continued

	Worker at 1,000 m				Maximum Offsite Individual				Population to 80 km				
	Dose (rem)		Probability of Cancer Fatality[a]		Dose (rem)		Probability of Cancer Fatality[a]		Dose (person-rem)		Number of Cancer Fatalities[b]		Accident Frequency (per year)
Accident Scenario	High	Low	High	Low	High	Low	High	Low	High	High	Low	Low	
Rupture of delivering pipe during pumped emplacement	8.5×10^{-8}	3.4×10^{-8}	3.4×10^{-11}	1.4×10^{-11}	1.4×10^{-8}	6.2×10^{-10}	7.2×10^{-12}	3.1×10^{-13}	1.2×10^{-5}	1.1×10^{-7}	6.0×10^{-9}	5.5×10^{-11}	1.0×10^{-5}
Delivering pipe dropped during pumped emplacement	1.7×10^{-8}	6.9×10^{-9}	6.8×10^{-12}	2.7×10^{-12}	2.8×10^{-9}	1.2×10^{-10}	1.4×10^{-12}	6.2×10^{-14}	2.4×10^{-6}	2.2×10^{-8}	1.2×10^{-9}	1.1×10^{-11}	1.0×10^{-5}
Mixing system breaks pellets during pumped emplacement	1.7×10^{-9}	6.9×10^{-10}	6.8×10^{-13}	2.7×10^{-13}	2.8×10^{-10}	1.2×10^{-11}	1.4×10^{-13}	6.2×10^{-15}	2.4×10^{-7}	2.2×10^{-9}	1.2×10^{-10}	1.1×10^{-12}	1.0×10^{-3}
Pellets break during pumped emplacement release	1.7×10^{-9}	6.9×10^{-10}	6.8×10^{-13}	2.7×10^{-13}	2.8×10^{-10}	1.2×10^{-11}	1.4×10^{-13}	6.2×10^{-15}	2.4×10^{-7}	2.2×10^{-9}	1.2×10^{-10}	1.1×10^{-12}	1.0×10^{-3}
Failure of ventilation filter	8.5×10^{-10}	3.4×10^{-10}	3.4×10^{-13}	1.4×10^{-13}	1.4×10^{-10}	6.2×10^{-12}	6.9×10^{-14}	3.1×10^{-15}	1.2×10^{-7}	1.1×10^{-9}	6.0×10^{-11}	5.5×10^{-13}	1.0×10^{-6}
Uncontrolled chemical reaction	1.4×10^{-9}	5.8×10^{-10}	5.7×10^{-13}	2.3×10^{-13}	2.3×10^{-10}	1.0×10^{-11}	1.2×10^{-13}	5.2×10^{-15}	2.0×10^{-7}	1.9×10^{-9}	1.0×10^{-10}	9.3×10^{-13}	1.0×10^{-6}
Pellet storage nuclear criticality	3.5×10^{-2}	1.6×10^{-2}	1.4×10^{-5}	6.2×10^{-6}	5.8×10^{-3}	2.0×10^{-4}	2.9×10^{-6}	1.0×10^{-7}	1.3	6.6×10^{-3}	6.3×10^{-4}	3.3×10^{-6}	1.0×10^{-6}
Pellet-grout mixing nuclear criticality	3.5×10^{-2}	1.6×10^{-2}	1.4×10^{-5}	6.2×10^{-6}	5.8×10^{-3}	2.0×10^{-4}	2.9×10^{-6}	1.0×10^{-7}	1.3	6.6×10^{-3}	6.3×10^{-4}	3.3×10^{-6}	1.0×10^{-6}

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Source: Calculated using the source terms in Tables M.5.3.3.1–3 and M.5.3.3.1–4 and the MACCS computer code.

M.5.3.5 Vitrification Alternative

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for a vitrification facility in the *Fissile Material Disposition Program PEIS Data Call Input Report: New Glass Vitrification Facility.* The studies postulated a set of accident scenarios that were representative of the risks and consequences for workers and the public that can be expected if the facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risk that are expected to envelope the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7}/yr are anticipated that will have consequences and risks larger than those described in this section. The potential for an aircraft crash has been considered and dismissed because the probability of a crash into a facility and causing sufficient damage to release Pu is much less than 10^{-7}/yr.

M.5.3.5.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the facility. Through a screening process, seven evaluation basis accidents and three beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.3.5.1–1 and M.5.3.5.1–2. Accident source term information is provided in Tables M.5.3.5.1–3 and M.5.3.5.1–4. Descriptions of accident scenarios are provided in Table M.5.3.5.1–5.

Table M.5.3.5.1–1. Evaluation Basis Accident Scenarios for the Vitrification Alternative

		Source Term	
Accident Scenario	**Accident Frequency (per year)**	**At Risk**	**Released to Environment**
Blender spill	0.01 to 0.1	82.5 kg Pu 9.23×10^{4} Ci Cs	1.24×10^{-5} g Pu 1.38×10^{-5} Ci Cs
Loss of offsite power	0.01 to 0.1	No release	No release
Melter spill	1.0×10^{-4} to 0.01	82.5 kg Pu 9.23×10^{4} Ci Cs	6.2×10^{-7} g Pu 6.9×10^{-7} Ci Cs
Cs capsule drop	1.0×10^{-4} to 0.01	70 kCi Cs	1.75×10^{-5} Ci Cs
Canister drop	1.0×10^{-4} to 0.01	82.5 kg Pu 9.23×10^{4} Ci Cs	6.1×10^{-6} g Pu 6.83×10^{-6} Ci Cs
CPC ion column fire	1.0×10^{-4} to 0.01	9.23×10^{4} Ci Cs	0.23 Ci Cs
Pu oxide solids fire oven	1.0×10^{-4} to 0.01	5 kg Pu	1.5×10^{-8} g Pu
Earthquake	1.0×10^{-6} to 1.0×10^{-4}	170 kg Pu 347 Ci Cs	3.86×10^{-4} g Pu 6.59×10^{-7} Ci Cs
Tornado	1.0×10^{-6} to 1.0×10^{-4}	No release	No release
Flood	1.0×10^{-6} to 1.0×10^{-4}	No release	No release

Source: LLNL 1996c.

Table M.5.3.5.1–2. Beyond Evaluation Basis Accident Scenarios for the Vitrification Alternative

Accident Scenario	Accident Frequency (per year)	Source Term: At Risk	Source Term: Released to Environment
Cs fire	$<1.0 \times 10^{-6}$	1.3×10^{6} Ci Cs	1.3 Ci Cs
Blender fire	$<1.0 \times 10^{-6}$	82.5 kg Pu 9.23×10^{4} Ci Cs	5.16×10^{-5} g Pu 0.231 Ci Cs
Nuclear criticality in Pu oxide oven	$<1.0 \times 10^{-6}$	1.0×10^{18} fissions. Release fractions: 0.5 noble gases, 0.05 iodine.	[a]

[a] See Table M.5.3.5.1–4.

Source: LLNL 1996c.

Table M.5.3.5.1–3. Vitrification Alternative Evaluation Basis Accident Source Terms

	Accident Scenario						
Accident Parameter	**Blender Spill**	**Melter Spill**	**Cs Capsule Drop**	**Canister Drop**	**CPC Ion Column Fire**	**Pu Oxide Oven Solids Fire**	**Earthquake**
Frequency of occurrence[a](per year)	0.05	$1.0x10^{-3}$/yr	$1.0x10^{-3}$/yr	$1.0x10^{-3}$/yr	$1.0x10^{-3}$/yr	$1.0x10^{-3}$/yr	$1.0x10^{-5}$/yr
Pu released to environment (g)	$1.24x10^{-5}$	$6.2x10^{-7}$	NA	$6.1x10^{-6}$	NA	$1.5x10^{-8}$	$3.86x10^{-4}$
Cs released to environment (Ci)	$1.38x10^{-5}$	$6.9x10^{-7}$	$1.75x10^{-5}$	$6.83x10^{-6}$	0.23	NA	$6.59x10^{-7}$
Isotope Released to Environment (Ci)							
Pu-238	$1.96x10^{-8}$	$9.80x10^{-10}$	0	$9.64x10^{-9}$	0	$2.37x10^{-11}$	$6.10x10^{-7}$
Pu-239	$7.09x10^{-7}$	$3.55x10^{-8}$	0	$3.49x10^{-7}$	0	$8.58x10^{-10}$	$2.21x10^{-5}$
Pu-240	$1.88x10^{-7}$	$9.42x10^{-9}$	0	$9.27x10^{-8}$	0	$2.28x10^{-10}$	$5.87x10^{-6}$
Pu-241	$6.68x10^{-7}$	$3.34x10^{-8}$	0	$3.29x10^{-7}$	0	$8.08x10^{-10}$	$2.08x10^{-5}$
Pu-242	$2.77x10^{-11}$	$1.38x10^{-12}$	0	$1.36x10^{-11}$	0	$3.34x10^{-14}$	$8.61x10^{-10}$
Am-241	$3.52x10^{-9}$	$1.76x10^{-10}$	0	$1.73x10^{-9}$	0	$4.26x10^{-12}$	$1.10x10^{-7}$
Cs-137	$1.38x10^{-5}$	$6.9x10^{-7}$	$1.75x10^{-5}$	$6.83x10^{-6}$	0.23	0	$6.59x10^{-7}$

[a] Midpoint of the estimated frequency range.

Note: Am=Americium; NA=not applicable.

Source: Derived from Tables M.5.1.3.4–1 and M.5.3.5.1–1.

Table M.5.3.5.1–4. Vitrification Alternative Beyond Evaluation Basis Accident Source Terms

	Accident Scenario		
Accident Parameter	**Cs Fire**	**Blender Fire**	**Nuclear Criticality in Pu Oxide Furnace[a]**
Frequency of occurrence (per year)	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}
Pu released to environment (g)	NA	5.16×10^{-5}	NA
Cs released to environment (Ci)	1.3	0.231	NA
Fissions	NA	NA	1.0×10^{18}
Isotope Released to Environment (Ci)			
Pu-238	0	8.15×10^{-8}	0
Pu-239	0	2.95×10^{-6}	0
Pu-240	0	7.84×10^{-7}	0
Pu-241	0	2.78×10^{-6}	0
Pu-242	0	1.15×10^{-10}	0
Am-241	0	1.47×10^{-8}	0
Cs-137	1.3	0.231	0
Kr-83m	0	0	5.5
Kr-85m	0	0	3.55
Kr-85	0	0	4.05×10^{-5}
Kr-87	0	0	21.5
Kr-88	0	0	11.5
Kr-89	0	0	650
Xe-131m	0	0	5.0×10^{-3}
Xe-133m	0	0	0.11
Xe-133	0	0	1.35
Xe-135m	0	0	165
Xe-135	0	0	20.5
Xe-137	0	0	2.45×10^{3}
Xe-138	0	0	550
I-131	0	0	0.055
I-132	0	0	6
I-133	0	0	0.8
I-134	0	0	21.5
I-135	0	0	2.25

[a] Curies produced (by isotope) or the 1.0×10^{18} fission criticality were scaled from Table M.5.3.1.1–3.

Note: NA=not applicable.

Source: Derived from Tables M.5.1.3.4–1, M.5.3.1.1–3, and M.5.3.5.1–2.

Table M.5.3.5.1–5. Accident Scenario Descriptions for the Vitrification Alternative

Accident Scenario	Accident Description
Evaluation Basis Accidents	
Blender spill	The spill occurs as material is transferred from the blender to the melter. Fine particulate materials that became airborne during the spill would be removed from the area by the ventilation system and passed through HEPA filters before release to the environment.
Melter spill	The source document provided summary data in a tabular format for this accident scenario. This accident scenario was not described in the source document.
Cs capsule drop	The source document provided summary data in a tabular format for this accident scenario. This accident scenario was not described in the source document.
Canister drop	It was postulated that the impact shatters the glass and disperses the fragments into the cell atmosphere. Fine particulate materials that became airborne would be removed from the area by the ventilation system and passed through HEPA filters before release to the environment.
CPC ion column fire	The source document provided summary data in a tabular format for this accident scenario. This accident scenario was not described in the source document.
Pu oxide oven solids fire	The source document provided summary data in a tabular format for this accident scenario. This accident scenario was not described in the source document.
Earthquake	Contents of the blender, melter, Pu oxide oven the Cs preparation cell would be spilled. Fine particulate materials that became airborne during the spills would be removed from the area by the ventilation system and passed through HEPA filters before release to the environment.
Beyond Evaluation Basis Accidents	
Cs fire	The combustible load for the processes involving Cs is very low. The Cs is in the form of CsCl which is not flammable. A large fire was postulated in the process area and all Cs effected by the fire was released to the area ventilation system and passed through HEPA filters before release to the environment.
Blender fire	The combustible load in the process is very low. The process involves no flammable material. A large fire was postulated in the process cell. It is assumed that the fire ruptures the blender and the blender contents are exposed to the fire. The resultant airborne material is removed by the area ventilation system and passed through HEPA filters before release to the environment.
Nuclear criticality in Pu oxide furnace	A criticality event was assumed to occur in the Pu oxide oven process area and the assumed criticality accident severity is based on guidance provided in NRC Regulatory Guide 3.35.

Source: LLNL 1996c.

M.5.3.5.2 *Accident Impacts*

The estimated impacts of the postulated accidents at each site are provided in Tables M.5.3.5.2–1 through M.5.3.5.2–6. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.3.5.1–3 and M.5.3.5.1–4 using the MACCS computer code. [Text deleted.]

Table M.5.3.5.2–1. Vitrification Alternative Accident Impacts at Hanford Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Blender spill	4.6×10^{-6}	1.8×10^{-9}	3.4×10^{-8}	1.7×10^{-11}	3.1×10^{-4}	1.5×10^{-7}	0.05
Melter spill	2.3×10^{-7}	9.1×10^{-11}	1.7×10^{-9}	8.6×10^{-13}	1.5×10^{-5}	7.6×10^{-9}	1.0×10^{-3}
Cs capsule drop	1.3×10^{-6}	5.3×10^{-10}	8.0×10^{-9}	4.0×10^{-12}	1.3×10^{-4}	6.4×10^{-8}	1.0×10^{-3}
Canister drop	2.2×10^{-6}	9.0×10^{-10}	1.7×10^{-8}	8.5×10^{-12}	1.5×10^{-4}	7.5×10^{-8}	1.0×10^{-3}
Cs ion processing fire	0.017	6.9×10^{-6}	1.1×10^{-4}	5.3×10^{-8}	1.7	8.4×10^{-4}	1.0×10^{-3}
Pu oxide oven solids fire	4.3×10^{-9}	1.7×10^{-12}	3.4×10^{-11}	1.7×10^{-14}	2.5×10^{-7}	1.2×10^{-10}	1.0×10^{-3}
Earthquake	1.1×10^{-4}	4.4×10^{-8}	8.8×10^{-7}	4.4×10^{-10}	6.4×10^{-3}	3.2×10^{-6}	1.0×10^{-5}
Cs fire	0.098	3.9×10^{-5}	6.0×10^{-4}	3.0×10^{-7}	9.5	4.7×10^{-3}	1.0×10^{-6}
Blender fire	0.017	7.0×10^{-6}	1.1×10^{-4}	5.3×10^{-8}	1.7	8.4×10^{-4}	1.0×10^{-6}
Nuclear criticality in Pu oxide furnace	1.0×10^{-3}	4.2×10^{-7}	6.5×10^{-6}	3.3×10^{-9}	7.0×10^{-3}	3.5×10^{-6}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.5.1–3 and M.5.3.5.1–4 and the MACCS computer code.

Table M.5.3.5.2–2. Vitrification Alternative Accident Impacts at Nevada Test Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Blender spill	3.1×10^{-6}	1.2×10^{-9}	5.5×10^{-8}	2.8×10^{-11}	7.0×10^{-6}	3.5×10^{-9}	0.05
Melter spill	1.6×10^{-7}	6.2×10^{-11}	2.7×10^{-9}	1.4×10^{-12}	3.5×10^{-7}	1.7×10^{-10}	1.0×10^{-3}
Cs capsule drop	8.9×10^{-7}	3.6×10^{-10}	1.3×10^{-8}	6.6×10^{-12}	3.0×10^{-6}	1.5×10^{-9}	1.0×10^{-3}
Canister drop	1.5×10^{-6}	6.1×10^{-10}	2.7×10^{-8}	1.4×10^{-11}	3.4×10^{-6}	1.7×10^{-9}	1.0×10^{-3}
Cs ion processing fire	0.012	4.7×10^{-6}	1.7×10^{-4}	8.7×10^{-8}	0.039	1.9×10^{-5}	1.0×10^{-3}
Pu oxide oven solids fire	2.9×10^{-9}	1.2×10^{-12}	5.4×10^{-11}	2.7×10^{-14}	5.6×10^{-9}	2.8×10^{-12}	1.0×10^{-3}
Earthquake	7.5×10^{-5}	3.0×10^{-8}	1.4×10^{-6}	9.0×10^{-10}	1.4×10^{-4}	7.2×10^{-8}	1.0×10^{-5}
Cs fire	0.066	2.6×10^{-5}	9.8×10^{-4}	4.9×10^{-7}	0.22	1.1×10^{-4}	1.0×10^{-6}
Blender fire	0.012	4.7×10^{-6}	1.8×10^{-4}	8.7×10^{-8}	0.039	2.0×10^{-5}	1.0×10^{-6}
Nuclear criticality in Pu oxide furnace	7.7×10^{-4}	3.1×10^{-7}	1.3×10^{-5}	6.5×10^{-9}	1.4×10^{-4}	6.9×10^{-8}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.5.1–3 and M.5.3.5.1–4 and the MACCS computer code.

Table M.5.3.5.2–3. Vitrification Alternative Accident Impacts at Idaho National Engineering Laboratory

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Blender spill	4.2×10^{-6}	1.7×10^{-9}	3.5×10^{-8}	1.7×10^{-11}	9.3×10^{-5}	4.7×10^{-8}	0.05
Melter spill	2.1×10^{-7}	8.4×10^{-11}	1.7×10^{-9}	8.7×10^{-13}	4.7×10^{-6}	2.3×10^{-9}	1.0×10^{-3}
Cs capsule drop	1.2×10^{-6}	4.6×10^{-10}	7.7×10^{-9}	3.8×10^{-12}	4.0×10^{-5}	2.0×10^{-8}	1.0×10^{-3}
Canister drop	2.1×10^{-6}	8.3×10^{-10}	1.7×10^{-8}	8.5×10^{-12}	4.6×10^{-5}	2.3×10^{-8}	1.0×10^{-3}
Cs ion processing fire	0.015	6.1×10^{-6}	1.0×10^{-4}	5.0×10^{-8}	0.53	2.6×10^{-4}	1.0×10^{-3}
Pu oxide oven solids fire	4.0×10^{-9}	1.6×10^{-12}	3.4×10^{-11}	1.7×10^{-14}	7.4×10^{-8}	3.7×10^{-11}	1.0×10^{-3}
Earthquake	1.0×10^{-4}	4.1×10^{-8}	8.8×10^{-7}	4.4×10^{-10}	1.9×10^{-3}	9.6×10^{-7}	1.0×10^{-5}
Cs fire	0.086	3.4×10^{-5}	5.7×10^{-4}	2.9×10^{-7}	3.0	1.5×10^{-3}	1.0×10^{-6}
Blender fire	0.015	6.1×10^{-6}	1.0×10^{-4}	5.1×10^{-8}	0.53	2.7×10^{-4}	1.0×10^{-6}
Nuclear criticality in Pu oxide furnace	1.0×10^{-3}	4.0×10^{-7}	7.7×10^{-6}	3.9×10^{-9}	1.8×10^{-3}	9.0×10^{-7}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.5.1–3 and M.5.3.5.1–4 and the MACCS computer code.

Table M.5.3.5.2–4. Vitrification Alternative Accident Impacts at Pantex Plant

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Blender spill	1.9×10^{-6}	7.5×10^{-10}	4.2×10^{-7}	2.1×10^{-10}	1.0×10^{-4}	5.0×10^{-8}	0.05
Melter spill	9.3×10^{-8}	3.7×10^{-11}	2.1×10^{-8}	1.0×10^{-11}	5.0×10^{-6}	2.5×10^{-9}	1.0×10^{-3}
Cs capsule drop	5.6×10^{-7}	2.2×10^{-10}	1.2×10^{-7}	5.9×10^{-11}	3.8×10^{-5}	1.9×10^{-8}	1.0×10^{-3}
Canister drop	9.2×10^{-7}	3.7×10^{-10}	2.1×10^{-7}	1.0×10^{-10}	4.9×10^{-5}	2.5×10^{-8}	1.0×10^{-3}
Cs ion processing fire	7.4×10^{-3}	3.0×10^{-6}	1.5×10^{-3}	7.7×10^{-7}	0.51	2.5×10^{-4}	1.0×10^{-3}
Pu oxide oven solids fire	1.7×10^{-9}	6.9×10^{-13}	4.0×10^{-10}	2.0×10^{-13}	8.4×10^{-8}	4.2×10^{-11}	1.0×10^{-3}
Earthquake	4.4×10^{-5}	1.8×10^{-8}	1.0×10^{-5}	5.1×10^{-9}	2.2×10^{-3}	1.1×10^{-6}	1.0×10^{-5}
Cs fire	0.042	1.7×10^{-5}	8.7×10^{-3}	4.4×10^{-6}	2.9	1.4×10^{-3}	1.0×10^{-6}
Blender fire	7.4×10^{-3}	3.0×10^{-6}	1.6×10^{-3}	7.8×10^{-7}	0.51	2.5×10^{-4}	1.0×10^{-6}
Nuclear criticality in Pu oxide furnace	4.8×10^{-4}	1.9×10^{-7}	1.4×10^{-4}	7.0×10^{-8}	4.5×10^{-3}	2.2×10^{-6}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.5.1–3 and M.5.3.5.1–4 and the MACCS computer code.

Table M.5.3.5.2–5. Vitrification Alternative Accident Impacts at Oak Ridge Reservation

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Blender spill	4.3×10^{-6}	1.7×10^{-9}	7.3×10^{-7}	3.7×10^{-10}	7.1×10^{-4}	3.5×10^{-7}	0.05
Melter spill	2.1×10^{-7}	8.6×10^{-11}	3.7×10^{-8}	1.8×10^{-11}	3.5×10^{-5}	1.8×10^{-8}	1.0×10^{-3}
Cs capsule drop	1.3×10^{-6}	5.2×10^{-10}	2.1×10^{-7}	1.0×10^{-10}	2.7×10^{-4}	1.3×10^{-7}	1.0×10^{-3}
Canister drop	2.1×10^{-6}	8.4×10^{-10}	3.6×10^{-7}	1.8×10^{-10}	3.5×10^{-4}	1.7×10^{-7}	1.0×10^{-3}
Cs ion processing fire	0.017	6.8×10^{-6}	2.7×10^{-3}	1.4×10^{-6}	3.5	1.8×10^{-3}	1.0×10^{-3}
Pu oxide oven solids fire	3.9×10^{-9}	1.6×10^{-12}	6.9×10^{-10}	3.5×10^{-13}	6.0×10^{-7}	3.0×10^{-10}	1.0×10^{-3}
Earthquake	1.0×10^{-4}	4.1×10^{-8}	1.8×10^{-5}	8.9×10^{-9}	0.015	7.7×10^{-6}	1.0×10^{-5}
Cs fire	0.095	3.8×10^{-5}	0.015	7.7×10^{-6}	19.8	9.9×10^{-3}	1.0×10^{-6}
Blender fire	0.017	6.8×10^{-6}	2.7×10^{-3}	1.4×10^{-6}	3.5	1.8×10^{-3}	1.0×10^{-6}
Nuclear criticality in Pu oxide furnace	9.5×10^{-4}	3.8×10^{-7}	1.7×10^{-4}	8.5×10^{-8}	0.031	1.6×10^{-5}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m or the site boundary [1,000 m for this facility at ORR], 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.5.1–3 and M.5.3.5.1–4 and the MACCS computer code.

Table M.5.3.5.2–6. Vitrification Alternative Accident Impacts at Savannah River Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Blender spill	3.0×10^{-6}	1.2×10^{-9}	5.8×10^{-8}	2.9×10^{-11}	3.2×10^{-4}	1.6×10^{-7}	0.05
Melter spill	1.5×10^{-7}	6.1×10^{-11}	2.9×10^{-9}	1.5×10^{-12}	1.6×10^{-5}	8.0×10^{-9}	1.0×10^{-3}
Cs capsule drop	9.2×10^{-7}	3.7×10^{-10}	1.6×10^{-8}	8.2×10^{-12}	1.3×10^{-4}	6.4×10^{-8}	1.0×10^{-3}
Canister drop	1.5×10^{-6}	6.0×10^{-10}	2.9×10^{-8}	1.4×10^{-11}	1.6×10^{-4}	7.9×10^{-8}	1.0×10^{-3}
Cs ion processing fire	0.012	4.8×10^{-6}	2.2×10^{-4}	1.1×10^{-7}	1.7	8.4×10^{-4}	1.0×10^{-3}
Pu oxide oven solids fire	2.8×10^{-9}	1.1×10^{-12}	5.5×10^{-11}	2.7×10^{-12}	2.7×10^{-7}	1.3×10^{-10}	1.0×10^{-3}
Earthquake	7.2×10^{-5}	2.9×10^{-8}	1.4×10^{-6}	7.1×10^{-10}	6.8×10^{-3}	3.4×10^{-8}	1.0×10^{-5}
Cs fire	0.068	2.7×10^{-5}	1.2×10^{-3}	6.1×10^{-7}	9.5	4.7×10^{-3}	1.0×10^{-6}
Blender fire	0.012	4.8×10^{-6}	2.2×10^{-4}	1.1×10^{-7}	1.7	8.4×10^{-4}	1.0×10^{-6}
Nuclear criticality in Pu oxide furnace	6.9×10^{-4}	2.8×10^{-7}	1.1×10^{-5}	5.7×10^{-9}	9.4×10^{-3}	4.7×10^{-6}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.5.1–3 and M.5.3.5.1–4 and the MACCS computer code.

M.5.3.6 Ceramic Immobilization Alternative

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for a ceramic immobilized facility in the *Fissile Material Disposition Program PEIS Data Call Input Report: Ceramic Immobilization Facility with Radionuclides*. The studies postulated a set of accidents scenarios that were representative of the risks and consequences for workers and the public that can be expected if the facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risk that are expected to envelop the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7} per year are anticipated that will have consequences and risks larger than those described in this section. The potential for an aircraft crash has been considered and dismissed because the probability of a crash into a facility and causing sufficient damage to release Pu is much less than 10^{-7}/yr.

M.5.3.6.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the facility. Through a screening process, nine evaluation basis accidents and four beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.3.6.1–1 and M.5.3.6.1–2. Accident source term information is provided in Tables M.5.3.6.1–3 and M.5.3.6.1–4. Descriptions of accident scenarios are provided in Table M.5.3.6.1–5.

Table M.5.3.6.1–1. Evaluation Basis Accident Scenarios for the Ceramic Immobilization Alternative

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Earthquake	1.0×10^{-6} to 1.0×10^{-4}	7 kg Pu	7.0×10^{-6} g Pu
Tornado	1.0×10^{-6} to 1.0×10^{-4}	No Release	No Release
Flood	1.0×10^{-6} to 1.0×10^{-4}	No Release	No Release
Glovebox fire	1.0×10^{-6} to 1.0×10^{-4}	7 kg Pu	7.0×10^{-6} g Pu
Glovebox nuclear criticality	1.0×10^{-6} to 1.0×10^{-4}	10^{18} fissions. Release fractions: 1.0 noble gases, 0.25 halogens.	[a]
Mixing tank nuclear criticality	1.0×10^{-6} to 1.0×10^{-4}	1.0×10^{19} fissions total. 1.0×10^{18} fissions initial, 47 pulses of 1.0×10^{17} fissions at 10 minute intervals. Release fractions: 1.0 noble gases, 0.25 halogens.	[a]
Bellows drop	1.0×10^{-4} to 0.01	4 kg Pu 4,330 Ci Cs	4.0×10^{-9}g Pu 4.3×10^{-9} Ci Cs
Canister drop	1.0×10^{-4} to 0.01	No Release	No Release
Cs capsule drop	1.0×10^{-4} to 0.01	4.0×10^{7} Ci Cs	4.0×10^{-5} Ci Cs
Plutonyl nitrate dissolver spill	0.01 to 0.1	0.4 kg Pu	2.4×10^{-11} g Pu
Calciner feed spill	0.01 to 0.1	2.5 kg Pu 2,740 Ci Cs	1.25×10^{-10} g Pu 1.37×10^{-10} Ci Cs
Calciner product spill	0.01 to 0.1	5 kg Pu 5,480 Ci Cs	3.5×10^{-8} g Pu 3.8×10^{-8} Ci Cs
Loss of off-site power	0.01 to 0.1	No Release	No Release

[a] See Table M.5.3.6.1–3.
Source: LLNL 1996d; NRC 1979a.

Table M.5.3.6.1–2. Beyond Evaluation Basis Accident Scenarios for the Ceramic Immobilization Alternative

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Cs fire	$<1.0 \times 10^{-6}$	1.3×10^{6} Ci Cs	1.3×10^{-5} Ci Cs
Process cell fire	$<1.0 \times 10^{-6}$	50 kg Pu	5.0×10^{-7} g Pu
Nuclear criticality	$<1.0 \times 10^{-6}$	3.0×10^{20} fissions total. 5.0×10^{19} fissions initial, 47 pulses of 5.0×10^{18} fissions at 10 minute intervals. Release fractions: 1.0 noble gases, 0.25 halogens.	[a]
Uncontrolled chemical reaction	$<1.0 \times 10^{-6}$	25 kg Pu 27,400 Ci Cs	2.5×10^{-7} g Pu 2.74×10^{-7} Ci Cs

[a] See Table M.5.3.6.1–4.
Source: LLNL 1996d; NRC 1979a.

Table M.5.3.6.1–3. Ceramic Immobilization Alternative Evaluation Basis Accident Source Terms

	Accident Scenario								
Accident Parameter	**Earthquake**	**Glovebox Fire**	**Glovebox Nuclear Criticality[a]**	**Mixing Tank Nuclear Criticality[a]**	**Bellows Drop**	**Cs Capsule Drop**	**Plutonyl Nitrate Dissolver Spill**	**Calciner Feed Spill**	**Calciner Product Spill**
Frequency of occurrence[b] (per year)	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-3}	1.0×10^{-3}	0.05	0.05	0.05
Pu released to environment (g)	7.0×10^{-6}	7.0×10^{-6}	NA	NA	4.0×10^{-9}	NA	2.4×10^{-11}	1.25×10^{-10}	3.5×10^{-8}
Cs released to environment (Ci)	NA	NA	NA	NA	4.3×10^{-9}	4.0×10^{-5}	NA	1.37×10^{-10}	3.8×10^{-8}
Fissions	NA	NA	1.0×10^{18}	1.0×10^{19}	NA	NA	NA	NA	NA
Isotope Released to Environment (Ci)									
Pu-238	1.11×10^{-8}	1.11×10^{-8}	0	0	6.32×10^{-12}	0	3.79×10^{-14}	1.98×10^{-13}	5.53×10^{-11}
Pu-239	4.00×10^{-7}	4.06×10^{-7}	0	0	2.79×10^{-10}	0	1.37×10^{-12}	7.15×10^{-12}	2.00×10^{-9}
Pu-240	1.06×10^{-7}	1.06×10^{-7}	0	0	6.08×10^{-11}	0	3.65×10^{-13}	1.90×10^{-12}	5.32×10^{-10}
Pu-241	3.77×10^{-7}	3.77×10^{-7}	0	0	2.16×10^{-10}	0	1.29×10^{-12}	6.74×10^{-12}	1.89×10^{-9}
Pu-242	1.56×10^{-11}	1.56×10^{-11}	0	0	8.92×10^{-15}	0	5.35×10^{-17}	2.79×10^{-16}	7.81×10^{-14}
Am-241	1.99×10^{-9}	1.99×10^{-9}	0	0	1.14×10^{-12}	0	6.82×10^{-15}	3.55×10^{-14}	9.94×10^{-12}
Cs-137	0	0	0	0	4.3×10^{-9}	4.0×10^{-5}	0	1.37×10^{-10}	3.8×10^{-8}
Kr-83m	0	0	11	110	0	0	0	0	0
Kr-85m	0	0	7.1	71	0	0	0	0	0
Kr-85	0	0	8.1×10^{-5}	8.1×10^{-4}	0	0	0	0	0
Kr-87	0	0	43	430	0	0	0	0	0
Kr-88	0	0	23	230	0	0	0	0	0
Kr-89	0	0	1.3×10^{3}	1.3×10^{4}	0	0	0	0	0

Table M.5.3.6.1–3. Ceramic Immobilization Alternative Evaluation Basis Accident Source Terms—Continued

	Accident Scenario								
Accident Parameter	**Earthquake**	**Glovebox Fire**	**Glovebox Nuclear Criticality[a]**	**Mixing Tank Nuclear Criticality[a]**	**Bellows Drop**	**Cs Capsule Drop**	**Plutonyl Nitrate Dissolver Spill**	**Calciner Feed Spill**	**Calciner Product Spill**
Xe-131m	0	0	0.01	0.1	0	0	0	0	0
Xe-133m	0	0	0.22	2.2	0	0	0	0	0
Xe-133	0	0	2.7	27	0	0	0	0	0
Xe-135m	0	0	330	3.3×10^3	0	0	0	0	0
Xe-135	0	0	41	410	0	0	0	0	0
Xe-137	0	0	4.9×10^3	4.9×10^4	0	0	0	0	0
Xe-138	0	0	1.1×10^3	1.1×10^4	0	0	0	0	0
I-131	0	0	0.28	2.75	0	0	0	0	0
I-132	0	0	30	300	0	0	0	0	0
I-133	0	0	4	40	0	0	0	0	0
I-134	0	0	108	1.08×10^3	0	0	0	0	0
I-135	0	0	11.3	113	0	0	0	0	0

[a] Curies produced (by isotope) for the 1.0×10^{18} and 1.0×10^{19} fission criticalities were scaled from Table M.5.3.1.1–3.

[b] Midpoint of estimated frequency range.

Note: NA=not applicable.

Source: Derived from Tables M.5.1.3.4–1, M.5.3.1.1–3, and M.5.3.6.1–1.

Table M.5.3.6.1–4. Ceramic Immobilization Alternative Beyond Evaluation Basis Accident Source Terms

	Accident Scenario			
Accident Parameter	**Cs Fire**	**Process Cell Fire**	**Nuclear Criticality[a]**	**Uncontrolled Chemical Reaction**
Frequency of occurrence (per year)[b]	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}
Pu released to environment (g)	NA	5.0×10^{-7}	NA	2.5×10^{-7}g
Cs released to environment (Ci)	1.3×10^{-5}	NA	NA	2.74×10^{-7}
Fissions	NA	NA	3.0×10^{20}	NA
Isotope Released to Environment (Ci)				
Pu-238	0	7.9×10^{-10}	0	3.95×10^{-10}
Pu-239	0	2.86×10^{-8}	0	1.43×10^{-8}
Pu-240	0	7.60×10^{-9}	0	3.80×10^{-9}
Pu-241	0	2.69×10^{-8}	0	1.35×10^{-8}
Pu-242	0	1.12×10^{-12}	0	5.58×10^{-13}
Am-241	0	1.42×10^{-10}	0	7.10×10^{-11}
Cs-137	1.3×10^{-5}	0	0	2.74×10^{-7}
Kr-83m	0	0	3.3×10^{3}	0
Kr-85m	0	0	2.13×10^{3}	0
Kr-85	0	0	0.0243	0
Kr-87	0	0	1.29×10^{4}	0
Kr-88	0	0	6.90×10^{3}	0
Kr-89	0	0	3.90×10^{5}	0
Xe-131m	0	0	3.0	0
Xe-133m	0	0	66	0
Xe-133	0	0	810	0
Xe-135m	0	0	9.9×10^{4}	0
Xe-135	0	0	1.23×10^{4}	0
Xe-137	0	0	1.47×10^{6}	0
Xe-138	0	0	3.3×10^{5}	0
I-131	0	0	82.5	0
I-132	0	0	9.0×10^{3}	0
I-133	0	0	1.2×10^{3}	0
I-134	0	0	3.23×10^{4}	0
I-135	0	0	3.38×10^{3}	0

[a] Curies produced (by isotope) for the 3.0×10^{20} fission criticality was scaled from Table M.5.3.1.1–3.
[b] Midpoint of estimated frequency range.
Note: NA=not applicable.
Source: Derived from Tables M.5.1.3.4–1, M.5.3.1.1–3, and M.5.3.6.1–2.

Table M.5.3.6.1–5. Accident Scenario Descriptions for Ceramic Immobilization Alternative

Accident Scenario	Accident Description
Evaluation Basis Accidents	
Earthquake	It is assumed that the earthquake starts a fire in the room housing the Pu metal glovebox line. The fire is unimpeded and breaches a glovebox containing Pu. The glovebox inert atmosphere is lost and the Pu ignites. The ventilation system removes the Pu-containing gases from the area. The gasses pass through HEPA filters and are then released to the environment.
Glovebox fire	It is assumed that an unimpeded fire begins in the room housing the Pu metal glovebox line and breaches a glovebox containing Pu. The glovebox inert atmosphere is lost and the Pu ignites. The ventilation system removes the Pu-containing gases from the area. The gases pass through HEPA filters and are then released to the environment.
Glovebox nuclear criticality	It is assumed that controls are violated so that additional fissile material is introduced into a double batched glovebox. This results in a criticality.
Mixing tank nuclear criticality	It is assumed that controls are violated so that limits on fissile materials and poison controls are violated. A pulsed criticality event results.
Bellows drop	A bellows is dropped 6 m during handling. The force of the drop factures the ceramic material and ruptures the bellows. Respirable fines of ceramic are released to the cell and collected by the ventilation system. The airborne fines pass through HEPA filters and are released to the environment.
Cs capsule drop	A capsule is dropped 6 m during handling. The force of the drop fractures the CsCl material and ruptures the capsule. Respirable fines of CsCl are released to the cell and collected by the ventilation system. The airborne fines pass through HEPA filters and are released to the environment.
Plutonyl nitrate dissolver spill	It is postulated that the dissolver overflows the spills onto the floor. The spill spreads out in a safe geometry. The spill is cleaned up in two hours but some of the spill material is aerosolized and becomes airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Calciner feed spill	It is postulated that the calciner feed make-up tank overflows and spills onto the floor. The spill spreads out in a safe geometry. The spill is cleaned up in two hours but some of the spill material is aerosolized and becomes airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Calciner product spill	It is postulated that the calciner product bin overflow and spills powder onto the floor. The spill spreads out in a safe geometry. The spill is cleaned up in two hours but some of the spill becomes airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Beyond Evaluation Basis Accidents	
Cs fire	The combustible load for the processes involving Cs is very low. The Cs is in the form of CsCl which is not flammable. A large fire was postulated in the process area and all Cs effected by the fire was released to the area ventilation system and passes through HEPA filters before release to the environment.
Process cell fire	The combustible load in the remote process cells is very low. The process involves no flammable material. A large fire was postulated in the process cell. It is assumed that the fire ruptures the calciner product bins and the contents are exposed to the fire. The resultant airborne material is removed by the area ventilation system and passed through HEPA filters before release to the environment.

Table M.5.3.6.1–5. Accident Scenario Descriptions for Ceramic Immobilization Alternative—Continued

Accident Scenario	Accident Description
Nuclear criticality	A criticality event was assumed to occur in the facility and the assumed criticality accident severity is based on guidance provided in NRC Regulatory Guide 3.35.
Uncontrolled chemical reaction	Radiolytic hydrogen will be produced in the solutions in the facility. It was assumed that hydrogen accumulated within tanks because the tanks were isolated from the gas treatment system from a considerable period of time. It was postulated that hydrogen detonated in the calciner feed tank and some of the tank contents became airborne. The resultant airborne material is removed by the area ventilation system and passed through HEPA filters before release to the environment.

Source: LLNL 1996d.

M.5.3.6.2 *Accident Impacts*

The estimated impacts of the postulated accidents at each site are provided in Tables M.5.3.6.2–1 through M.5.3.6.2–6. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.3.6.1–3 and M.5.3.6.1–4 using the MACCS computer code. [Text deleted.]

Table M.5.3.6.2–1. Ceramic Immobilization Alternative Accident Impacts at Hanford Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Earthquake	2.0×10^{-6}	7.9×10^{-10}	1.6×10^{-8}	7.9×10^{-12}	1.2×10^{-4}	5.8×10^{-8}	1.0×10^{-5}
Glovebox fire	2.0×10^{-6}	7.9×10^{-10}	1.6×10^{-8}	7.9×10^{-12}	1.2×10^{-4}	5.8×10^{-8}	1.0×10^{-5}
Glovebox criticality	3.5×10^{-3}	1.4×10^{-6}	2.3×10^{-5}	1.2×10^{-8}	0.032	1.6×10^{-5}	1.0×10^{-5}
Mixing tank criticality	0.035	1.4×10^{-5}	2.3×10^{-4}	1.2×10^{-7}	0.32	1.6×10^{-4}	1.0×10^{-5}
Bellows drop	1.5×10^{-9}	5.8×10^{-13}	1.1×10^{-11}	5.5×10^{-15}	9.7×10^{-8}	4.9×10^{-11}	1.0×10^{-3}
Cesium capsule drop	3.0×10^{-6}	1.2×10^{-9}	1.8×10^{-8}	9.2×10^{-12}	2.9×10^{-4}	1.5×10^{-7}	1.0×10^{-3}
Plutonyl nitrate dissolver spill	6.8×10^{-12}	2.7×10^{-15}	5.4×10^{-14}	2.7×10^{-17}	4.0×10^{-10}	2.0×10^{-13}	0.05
Calciner feed spill	4.6×10^{-11}	1.8×10^{-14}	3.5×10^{-13}	1.7×10^{-16}	3.1×10^{-9}	1.5×10^{-12}	0.05
Calciner product spill	1.3×10^{-8}	5.1×10^{-12}	9.7×10^{-11}	4.8×10^{-14}	8.5×10^{-7}	4.3×10^{-10}	0.05
Cesium fire	9.8×10^{-7}	3.9×10^{-10}	6.0×10^{-9}	3.0×10^{-12}	9.5×10^{-5}	4.7×10^{-8}	1.0×10^{-6}
Process cell fire	1.4×10^{-7}	5.7×10^{-11}	1.1×10^{-9}	5.7×10^{-13}	8.2×10^{-6}	4.1×10^{-9}	1.0×10^{-6}
Criticality	1.0	4.2×10^{-4}	6.9×10^{-3}	3.5×10^{-6}	9.5	4.8×10^{-3}	1.0×10^{-6}
Uncontrolled chemical reaction	9.1×10^{-8}	3.7×10^{-11}	7.0×10^{-10}	3.5×10^{-13}	6.1×10^{-6}	3.1×10^{-9}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.6.1–3 and M.5.3.6.1–4 and the MACCS computer code.

Table M.5.3.6.2–2. Ceramic Immobilization Alternative Accident Impacts at Nevada Test Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Earthquake	1.4×10^{-6}	5.4×10^{-10}	2.5×10^{-8}	1.3×10^{-11}	2.6×10^{-6}	1.3×10^{-9}	1.0×10^{-5}
Glovebox fire	1.4×10^{-6}	5.4×10^{-10}	2.5×10^{-8}	1.3×10^{-11}	2.6×10^{-6}	1.3×10^{-9}	1.0×10^{-5}
Glovebox criticality	2.5×10^{-3}	1.0×10^{-6}	4.5×10^{-5}	2.3×10^{-8}	6.5×10^{-4}	3.3×10^{-7}	1.0×10^{-5}
Mixing tank criticality	0.025	1.0×10^{-5}	4.5×10^{-4}	2.3×10^{-7}	6.5×10^{-3}	3.3×10^{-6}	1.0×10^{-5}
Bellows drop	9.9×10^{-10}	4.0×10^{-13}	1.8×10^{-11}	8.8×10^{-15}	2.2×10^{-9}	1.1×10^{-12}	1.0×10^{-3}
Cesium capsule drop	2.0×10^{-6}	8.1×10^{-10}	3.0×10^{-8}	1.5×10^{-11}	6.7×10^{-6}	3.4×10^{-9}	1.0×10^{-3}
Plutonyl nitrate dissolver spill	4.7×10^{-12}	1.9×10^{-15}	8.6×10^{-14}	4.3×10^{-17}	8.9×10^{-12}	4.5×10^{-15}	0.05
Calciner feed spill	3.1×10^{-11}	1.2×10^{-14}	5.5×10^{-13}	2.8×10^{-16}	7.0×10^{-11}	3.5×10^{-14}	0.05
Calciner product spill	8.7×10^{-9}	3.5×10^{-12}	1.5×10^{-10}	7.7×10^{-14}	2.0×10^{-8}	9.7×10^{-12}	0.05
Cesium fire	6.6×10^{-7}	2.6×10^{-10}	9.8×10^{-9}	4.9×10^{-12}	2.2×10^{-6}	1.1×10^{-9}	1.0×10^{-6}
Process cell fire	9.7×10^{-8}	3.9×10^{-11}	1.8×10^{-9}	9.0×10^{-13}	1.9×10^{-7}	9.3×10^{-11}	1.0×10^{-6}
Criticality	0.76	3.0×10^{-4}	0.014	6.8×10^{-6}	0.20	9.7×10^{-5}	1.0×10^{-6}
Uncontrolled chemical reaction	6.2×10^{-8}	2.5×10^{-11}	1.1×10^{-9}	5.5×10^{-13}	1.4×10^{-7}	7.0×10^{-11}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.6.1–3 and M.5.3.6.1–4 and the MACCS computer code.

Table M.5.3.6.2–3. Ceramic Immobilization Alternative Accident Impacts at Idaho National Engineering Laboratory

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Earthquake	1.9×10^{-6}	7.4×10^{-10}	1.6×10^{-8}	8.0×10^{-12}	3.5×10^{-5}	1.7×10^{-8}	1.0×10^{-5}
Glovebox fire	1.9×10^{-6}	7.4×10^{-10}	1.6×10^{-8}	8.0×10^{-12}	3.5×10^{-5}	1.7×10^{-8}	1.0×10^{-5}
Glovebox nuclear criticality	3.4×10^{-3}	1.4×10^{-6}	2.7×10^{-5}	1.3×10^{-8}	8.7×10^{-3}	4.3×10^{-6}	1.0×10^{-5}
Mixing tank nuclear criticality	0.034	1.4×10^{-5}	2.7×10^{-4}	1.3×10^{-7}	0.086	4.3×10^{-5}	1.0×10^{-5}
Bellows drop	1.3×10^{-9}	5.4×10^{-13}	1.1×10^{-11}	5.5×10^{-15}	3.0×10^{-8}	1.5×10^{-11}	1.0×10^{-3}
Cesium capsule drop	2.6×10^{-6}	1.1×10^{-9}	1.8×10^{-8}	8.8×10^{-12}	9.2×10^{-5}	4.6×10^{-8}	1.0×10^{-3}
Plutonyl nitrate dissolver spill	6.4×10^{-12}	2.5×10^{-15}	5.5×10^{-14}	2.8×10^{-17}	1.2×10^{-10}	5.9×10^{-14}	0.05
Calciner feed spill	4.2×10^{-11}	1.7×10^{-14}	3.5×10^{-13}	1.7×10^{-16}	9.3×10^{-10}	4.7×10^{-13}	0.05
Calciner product spill	1.2×10^{-8}	4.7×10^{-12}	9.7×10^{-11}	4.8×10^{-14}	2.6×10^{-7}	1.3×10^{-10}	0.05
Cesium fire	8.6×10^{-7}	3.4×10^{-10}	5.7×10^{-9}	2.9×10^{-12}	3.0×10^{-5}	1.5×10^{-8}	1.0×10^{-6}
Process cell fire	1.3×10^{-7}	5.3×10^{-11}	1.1×10^{-9}	5.7×10^{-13}	2.5×10^{-6}	1.2×10^{-9}	1.0×10^{-6}
Nuclear criticality	1.0	4.0×10^{-4}	8.1×10^{-3}	4.0×10^{-6}	2.6	1.3×10^{-3}	1.0×10^{-6}
Uncontrolled chemical reaction	8.4×10^{-8}	3.4×10^{-11}	6.9×10^{-10}	3.5×10^{-13}	1.9×10^{-6}	9.3×10^{-10}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

All values are mean values.

Calculated using the source terms in Tables M.5.3.6.1–3 and M.5.3.6.1–4 and the MACCS computer code.

Table M.5.3.6.2–4. Ceramic Immobilization Alternative Accident Impacts at Pantex Plant

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Earthquake	8.0×10^{-7}	3.2×10^{-10}	1.8×10^{-7}	9.2×10^{-11}	3.9×10^{-5}	2.0×10^{-8}	1.0×10^{-5}
Glovebox fire	8.0×10^{-7}	3.2×10^{-10}	1.8×10^{-7}	9.2×10^{-11}	3.9×10^{-5}	2.0×10^{-8}	1.0×10^{-5}
Glovebox nuclear criticality	1.5×10^{-3}	6.2×10^{-7}	4.4×10^{-4}	2.2×10^{-7}	0.019	9.5×10^{-6}	1.0×10^{-5}
Mixing tank nuclear criticality	0.015	6.2×10^{-6}	4.4×10^{-3}	2.2×10^{-6}	0.19	9.5×10^{-5}	1.0×10^{-5}
Bellows drop	6.0×10^{-10}	2.4×10^{-13}	1.3×10^{-10}	6.7×10^{-14}	3.2×10^{-8}	1.6×10^{-11}	1.0×10^{-3}
Cesium capsule drop	1.3×10^{-6}	5.1×10^{-10}	2.7×10^{-7}	1.3×10^{-10}	8.8×10^{-5}	4.4×10^{-8}	1.0×10^{-3}
Plutonyl nitrate dissolver spill	2.7×10^{-12}	1.1×10^{-15}	6.3×10^{-13}	3.2×10^{-16}	1.4×10^{-10}	6.7×10^{-14}	0.05
Calciner feed spill	1.9×10^{-11}	7.5×10^{-15}	4.2×10^{-12}	2.1×10^{-15}	1.0×10^{-9}	5.0×10^{-13}	0.05
Calciner product spill	5.2×10^{-9}	2.1×10^{-12}	1.2×10^{-9}	5.9×10^{-13}	2.8×10^{-7}	1.4×10^{-10}	0.05
Cesium fire	4.2×10^{-7}	1.7×10^{-10}	8.7×10^{-8}	4.4×10^{-11}	2.9×10^{-5}	1.4×10^{-8}	1.0×10^{-6}
Process cell fire	5.7×10^{-8}	2.3×10^{-11}	1.3×10^{-8}	6.6×10^{-12}	2.8×10^{-6}	1.4×10^{-9}	1.0×10^{-6}
Nuclear criticality	0.46	1.9×10^{-4}	0.13	6.5×10^{-5}	5.7	2.8×10^{-3}	1.0×10^{-6}
Uncontrolled chemical reaction	3.7×10^{-8}	1.5×10^{-11}	8.5×10^{-9}	4.2×10^{-12}	2.0×10^{-6}	1.0×10^{-9}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

All values are mean values.

Calculated using the source terms in Tables M.5.3.6.1–3 and M.5.3.6.1–4 and the MACCS computer code.

Table M.5.3.6.2–5. Ceramic Immobilization Alternative Accident Impacts at Oak Ridge Reservation

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Earthquake	1.8×10^{-6}	7.3×10^{-10}	3.2×10^{-7}	1.6×10^{-10}	2.8×10^{-4}	1.4×10^{-7}	1.0×10^{-5}
Glovebox fire	1.8×10^{-6}	7.3×10^{-10}	3.2×10^{-7}	1.6×10^{-10}	2.8×10^{-4}	1.4×10^{-7}	1.0×10^{-5}
Glovebox nuclear criticality	3.2×10^{-3}	1.3×10^{-6}	5.8×10^{-4}	2.9×10^{-7}	0.13	6.3×10^{-5}	1.0×10^{-5}
Mixing tank nuclear criticality	0.032	1.3×10^{-5}	5.8×10^{-3}	2.9×10^{-6}	1.3	6.3×10^{-4}	1.0×10^{-5}
Bellows drop	1.4×10^{-9}	5.5×10^{-13}	2.4×10^{-10}	1.2×10^{-13}	2.3×10^{-7}	1.1×10^{-10}	1.0×10^{-3}
Cesium capsule drop	2.9×10^{-6}	1.2×10^{-9}	4.7×10^{-7}	2.4×10^{-10}	6.1×10^{-4}	3.1×10^{7}	1.0×10^{-3}
Plutonyl nitrate dissolver spill	6.3×10^{-12}	2.5×10^{-15}	1.1×10^{-12}	5.5×10^{-16}	9.6×10^{-10}	4.8×10^{-13}	0.05
Calciner feed spill	4.3×10^{-11}	1.7×10^{-14}	7.4×10^{-12}	3.7×10^{-15}	7.1×10^{-9}	3.5×10^{-12}	0.05
Calciner product spill	1.2×10^{-8}	4.8×10^{-12}	2.1×10^{-9}	1.0×10^{-12}	2.0×10^{-6}	9.9×10^{-10}	0.05
Cesium fire	9.5×10^{-7}	3.8×10^{-10}	1.5×10^{-7}	7.7×10^{-11}	2.0×10^{-4}	9.9×10^{-8}	1.0×10^{-6}
Process cell fire	1.3×10^{-7}	5.3×10^{-11}	2.3×10^{-8}	1.2×10^{-11}	2.0×10^{-5}	1.0×10^{-8}	1.0×10^{-6}
Nuclear criticality	0.94	3.8×10^{-4}	0.17	8.6×10^{-5}	37.4	0.019	1.0×10^{-6}
Uncontrolled chemical reaction	8.6×10^{-8}	3.4×10^{-11}	1.5×10^{-8}	7.3×10^{-12}	0.14×10^{-5}	7.1×10^{-9}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m or the site boundary [1,000 m for this facility at ORR], 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.6.1–3 and M.5.3.6.1–4 and the MACCS computer code.

Table M.5.3.6.2–6. Ceramic Immobilization Alternative Accident Impacts at Savannah River Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Earthquake	1.3×10^{-6}	5.2×10^{-10}	2.6×10^{-8}	1.3×10^{-11}	1.2×10^{-4}	6.2×10^{-8}	1.0×10^{-5}
Glovebox fire	1.3×10^{-6}	5.2×10^{-10}	2.6×10^{-8}	1.3×10^{-11}	1.2×10^{-4}	6.2×10^{-8}	1.0×10^{-5}
Glovebox nuclear criticality	2.3×10^{-3}	9.1×10^{-7}	4.0×10^{-5}	2.0×10^{-8}	0.041	2.0×10^{-5}	1.0×10^{-5}
Mixing tank nuclear criticality	0.023	9.1×10^{-6}	4.0×10^{-4}	2.0×10^{-7}	0.41	2.0×10^{-4}	1.0×10^{-5}
Bellows drop	9.7×10^{-10}	3.9×10^{-13}	1.9×10^{-11}	9.3×10^{-15}	1.0×10^{-7}	5.1×10^{-11}	1.0×10^{-3}
Cesium capsule drop	2.1×10^{-6}	8.4×10^{-10}	3.8×10^{-8}	1.9×10^{-11}	2.9×10^{-4}	1.5×10^{-7}	1.0×10^{-3}
Plutonyl nitrate dissolver spill	4.5×10^{-12}	1.8×10^{-15}	8.8×10^{-14}	4.4×10^{-17}	4.2×10^{-10}	2.1×10^{-13}	0.05
Calciner feed spill	3.0×10^{-11}	1.2×10^{-14}	5.9×10^{-13}	2.9×10^{-16}	3.2×10^{-9}	1.6×10^{-12}	0.05
Calciner product spill	8.5×10^{-9}	3.4×10^{-12}	1.6×10^{-10}	8.2×10^{-12}	9.0×10^{-7}	4.5×10^{-10}	0.05
Cesium fire	6.8×10^{-7}	2.7×10^{-10}	1.2×10^{-8}	6.1×10^{-12}	9.5×10^{-5}	4.7×10^{-8}	1.0×10^{-6}
Process cell fire	9.3×10^{-8}	3.7×10^{-11}	1.8×10^{-9}	9.1×10^{-13}	8.8×10^{-6}	4.4×10^{-9}	1.0×10^{-6}
Nuclear criticality	0.68	2.7×10^{-4}	0.012	6.0×10^{-6}	12.2	6.1×10^{-3}	1.0×10^{-6}
Uncontrolled chemical reaction	6.1×10^{-8}	2.4×10^{-11}	1.2×10^{-9}	5.9×10^{-13}	6.4×10^{-6}	3.2×10^{-9}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.6.1–3 and M.5.3.6.1–4 and the MACCS computer code.

M.5.3.7 Ceramic Immobilization Facility for Immobilized Disposition Alternative

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for a ceramic immobilization facility in the *Fissile Material Disposition Program PEIS Data Call Input Report: Ceramic Immobilization Facility Using Coated Pellets Without Radionuclides*. The studies postulated a set of accident scenarios that represented the risks and consequences for workers and the public that can be expected if the facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risks that are expected to envelop the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7} per year are anticipated that will have consequences and risks larger than those described in this section. The potential for an aircraft crash has been considered and dismissed because the probability of a crash into a facility and causing sufficient damage to release Pu is much less than 10^{-7}/yr.

M.5.3.7.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public operating the facility. Through a screening process, nine evaluation basis accidents and three beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.3.7.1–1 and M.5.3.7.1–2. Accident source term information is provided in Tables M.5.3.7.1–3 and M.5.3.7.1–4. Descriptions of accident scenarios are provided in Table M.5.3.7.1–5.

Table M.5.3.7.1–1. Evaluation Basis Accident Scenarios for Immobilized Disposition at the Ceramic Immobilization Facility

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Earthquake	1.0×10^{-6} to 1.0×10^{-4}	20 kg Pu	2.0×10^{-5} g Pu
Tornado	1.0×10^{-6} to 1.0×10^{-4}	No Release	No Release
Flood	1.0×10^{-6} to 1.0×10^{-4}	No Release	No Release
Glovebox fire	1.0×10^{-6} to 1.0×10^{-4}	20 kg Pu	2.0×10^{-5} g Pu
Glovebox nuclear criticality	1.0×10^{-6} to 1.0×10^{-4}	1.0×10^{18} fissions. Release fractions: 1.0 noble gases, 0.25 halogens.	[a]
Calciner feed tank nuclear criticality	1.0×10^{-6} to 1.0×10^{-4}	1.0×10^{19} fissions total, 1.0×10^{18} fissions initial, 47 pulses of 1.0×10^{17} fissions at 10 minute intervals. Release fractions: 1.0 noble gases, 0.25 halogens.	[a]
Ceramic can drop	1.0×10^{-4} to 0.01	0.5 kg Pu	5.0×10^{-10} g Pu
Pellet container breakage	1.0×10^{-4} to 0.01	5 kg Pu	5.0×10^{-12} g Pu
Dissolver spill	0.01 to 0.1	0.4 kg Pu	2.4×10^{-11} g Pu
Calciner feed spill	0.01 to 0.1	1.4 kg Pu	7.0×10^{-11} g Pu
Calciner product spill	0.01 to 0.1	2.5 kg Pu	1.75×10^{-8} g Pu
Loss of offsite power	0.01 to 0.1	No Release	No Release

[a] See Table M.5.3.7.1–3.
Source: LLNL 1996e.

Table M.5.3.7.1–2. Beyond Evaluation Basis Accident Scenarios for Immobilized Disposition at the Ceramic Immobilization Facility

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Sintering furnace explosion	$<1.0 \times 10^{-6}$	3 kg Pu	3.0×10^{-4} g Pu
Uncontrolled chemical reaction	$<1.0 \times 10^{-6}$	14 kg Pu	1.4×10^{-5} g Pu
Pu storage nuclear criticality	$<1.0 \times 10^{-6}$	10^{18} fissions. Release fractions: 1.0 noble gases, 0.25 halogens.	[a]

[a] See Table M.5.3.7.1–4.
Source: LLNL 1996e.

Table M.5.3.7.1–3. Immobilized Disposition at the Ceramic Immobilization Facility Evaluation Basis Accident Source Terms

	Accident Scenario								
Accident Parameter	**Earthquake**	**Glovebox Fire**	**Glovebox Nuclear Criticality[a]**	**Calciner Feed Nuclear Criticality[a]**	**Ceramic Can Drop**	**Pellet Container Breakage**	**Plutonyl Nitrate Dissolver Spill**	**Calciner Feed Spill**	**Calciner Product Spill**
Frequency of occurrence (per year)[b]	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-3}	1.0×10^{-3}	0.05	0.05	0.05
Pu released to environment (g)	2.0×10^{-5}	2.0×10^{-5}	NA	NA	5.0×10^{-10}	5.0×10^{-12}	2.4×10^{-11}	7.0×10^{-11}	1.75×10^{-8}
Fissions	NA	NA	1.0×10^{18}	1.0×10^{19}	NA	NA	NA	NA	NA
Isotope released to environment (Ci)									
Pu-238	3.16×10^{-8}	3.16×10^{-8}	0	0	7.90×10^{-13}	7.90×10^{-15}	3.79×10^{-14}	1.11×10^{-13}	2.77×10^{-11}
Pu-239	1.14×10^{-6}	1.14×10^{-6}	0	0	2.86×10^{-11}	2.86×10^{-13}	1.37×10^{-12}	4.00×10^{-12}	1.00×10^{-9}
Pu-240	3.04×10^{-7}	3.04×10^{-7}	0	0	7.60×10^{-12}	7.60×10^{-14}	3.65×10^{-13}	1.06×10^{-12}	2.66×10^{-10}
Pu-241	1.08×10^{-6}	1.08×10^{-6}	0	0	2.69×10^{-11}	2.69×10^{-13}	1.20×10^{-12}	3.77×10^{-12}	9.43×10^{-10}
Pu-242	4.46×10^{-11}	4.46×10^{-11}	0	0	1.11×10^{-15}	1.16×10^{-17}	5.35×10^{-17}	1.56×10^{-16}	3.90×10^{-14}
Am-241	5.68×10^{-9}	5.68×10^{-9}	0	0	1.42×10^{-13}	1.42×10^{-15}	6.82×10^{-15}	1.99×10^{-14}	4.97×10^{-12}
Cs-137	0	0	0	0	0	0	0	0	0
Kr-83m	0	0	11	110	0	0	0	0	0
Kr-85m	0	0	7.1	71	0	0	0	0	0
Kr-85	0	0	8.1×10^{-5}	8.1×10^{-4}	0	0	0	0	0
Kr-87	0	0	43	430	0	0	0	0	0
Kr-88	0	0	23	230	0	0	0	0	0
Kr-89	0	0	1.3×10^{3}	1.3×10^{4}	0	0	0	0	0
Xe-131m	0	0	0.01	0.1	0	0	0	0	0
Xe-133m	0	0	0.22	2.2	0	0	0	0	0
Xe-133	0	0	2.7	27	0	0	0	0	0
Xe-135m	0	0	330	3.3×10^{3}	0	0	0	0	0
Xe-135	0	0	41	410	0	0	0	0	0

Table M.5.3.7.1–3. Immobilized Disposition at the Ceramic Immobilization Facility Evaluation Basis Accident Source Terms—Continued

	Accident Scenario								
Accident Parameter	Earthquake	Glovebox Fire	Glovebox Nuclear Criticality[a]	Calciner Feed Nuclear Criticality[a]	Ceramic Can Drop	Pellet Container Breakage	Plutonyl Nitrate Dissolver Spill	Calciner Feed Spill	Calciner Product Spill
Xe-137	0	0	4.9×10^3	4.9×10^4	0	0	0	0	0
Xe-138	0	0	1.1×10^3	1.1×10^4	0	0	0	0	0
I-131	0	0	0.28	2.75	0	0	0	0	0
I-132	0	0	30	300	0	0	0	0	0
I-133	0	0	4	40	0	0	0	0	0
I-134	0	0	108	1.08×10^3	0	0	0	0	0
I-135	0	0	11.3	113	0	0	0	0	0

[a] Curies produced (by isotope) for the 1.0×10^{18} and 1.0×10^{19} fission criticalities were scaled from Table M.5.3.1.1–3.

[b] Midpoint of the estimated frequency range.

Note: All values are mean values.

Source: Derived from Tables M.5.1.3.4–1, M.5.3.1.1–3, and M.5.3.7.1–1.

Table M.5.3.7.1–4. Immobilized Disposition at the Ceramic Immobilization Facility Beyond Evaluation Basis Accident Source Terms

Accident Parameter	Accident Scenario		
	Sintering Furnace Explosion	**Uncontrolled Chemical Reaction**	**Nuclear Criticality[a]**
Frequency of occurrence (per year)[b]	1.0×10^{-6}	1.0×10^{-6}	1.0×10^{-6}
Pu released to environment (g)	3.0×10^{-4}	1.4×10^{-5}	NA
Fissions	NA	NA	1.0×10^{18}
Isotope Released to Environment (Ci)			
Pu-238	4.74×10^{-7}	2.21×10^{-8}	0
Pu-239	1.72×10^{-5}	8.01×10^{-7}	0
Pu-240	4.56×10^{-6}	2.13×10^{-7}	0
Pu-241	1.62×10^{-5}	7.55×10^{-7}	0
Pu-242	6.69×10^{-10}	3.12×10^{-11}	0
Am-241	8.52×10^{-8}	3.98×10^{-9}	0
Kr-83m	0	0	11
Kr-85m	0	0	7.1
Kr-85	0	0	8.1×10^{-5}
Kr-87	0	0	43
Kr-88	0	0	23
Kr-89	0	0	1.3×10^{3}
Xe-131m	0	0	0.01
Xe-133m	0	0	0.22
Xe-133	0	0	2.7
Xe-135m	0	0	330
Xe-135	0	0	41
Xe-137	0	0	4.9×10^{3}
Xe-138	0	0	1.1×10^{3}
I-131	0	0	0.28
I-132	0	0	30
I-133	0	0	4
I-134	0	0	108
I-135	0	0	11.3

[a] Curies produced (by isotope) for the 1.0×10^{18} fission criticality were scaled from Table M.5.3.1.1–3.
[b] Maximum of the estimated frequency range.
Note: All values are mean values.
Source: Derived from Tables M.5.1.3.4–1, M.5.3.1.1–3, and M.5.3.7.1–1.

Table M.5.3.7.1–5. Accident Scenario Descriptions for Immobilized Disposition at the Ceramic Immobilization Facility

Accident Scenario	Accident Description
Evaluation Basis Accidents	
Earthquake	It is assumed that the earthquake starts a fire in the room housing the plutonium metal glovebox line. The fire is unimpeded and breaches a glovebox containing plutonium. The glovebox inert atmosphere is lost and the Pu ignites. The ventilation system removes the Pu-containing gasses from the area. The gasses pass through HEPA filters and are then released to the environment.
Glovebox fire	It is assumed that an unimpeded fire begins in the room housing the Pu metal glovebox line and breaches a glovebox containing plutonium. The glovebox inert atmosphere is lost and the Pu ignites. The ventilation system removes the Pu-containing gases from the area. The gasses pass through HEPA filters and are then released to the environment.
Glovebox nuclear criticality	It is assumed that controls are violated so that additional fissile material is introduced into a double batched glovebox. This results in a criticality.
Calciner feed tank nuclear criticality	Criticality safety of this tank depends on controlling the concentrations of the gadolinium and plutonyl nitrate solutions in the tank. It is assumed that controls are violated so that limits on fissile materials and poison controls are violated. A pulsed criticality event results.
Ceramic can drop	A can is dropped during handling. The ceramic powder spills from the overturned can. Respirable fines of ceramic are released to the process area and collected by the ventilation system. The airborne fines pass through HEPA filters and are released to the environment.
Pellet container breakage	Finished ceramic pellets are loaded in drum containers and stored in the product storage vault. It is postulated that a container breakage could occur in ceramic pellet storage. A ceramic pellet container develops leakage during storage. Respirable fines of ceramic are released to the process area and collected by the ventilation system. The airborne fines pass through HEPA filters and are released to the environment.
Dissolver spill	It is postulated that the dissolver overflows and spills onto the floor. The spill spreads out in a safe geometry. The spill is cleaned up in 2 hours, but some of the spill material converts to an aerosol and becomes airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Calciner feed spill	It is postulated that the calciner feed make-up tank overflows and spills onto the floor. The spill spreads out in a safe geometry. The spill is cleaned up in 2 hours, but some of the spill material converts to an aerosol and becomes airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Calciner product spill	It is postulated that the calciner product bin overflows and spills powder onto the floor. The spill spreads out in a safe geometry. The spill is cleaned up in 2 hours, but some of the spill becomes airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.

Table M.5.3.7.1–5. Accident Scenario Descriptions for Immobilized Disposition at the Ceramic Immobilization Facility—Continued

Accident Scenario	Accident Description
Beyond Evaluation Basis Accidents	
Sintering furnace explosion	A pressure excursion in the sintering furnace of undefined origin could rupture the furnace vessel. It was postulated that the force from the explosion would blow ceramic pellets out of the ends and become airborne as respirable particles. The Pu-containing particulate would be removed from the process area by the ventilation system. The particulate then passes through a HEPA filtration system before it is released to the environment.
Uncontrolled chemical reaction	Radiolytic hydrogen will be produced in the solutions in the facility. It was assumed that hydrogen accumulated within tanks because the tanks were isolated from the gas treatment system for a considerable period of time. It was postulated that hydrogen detonated in the calciner feed tank and some of it became airborne. The resultant airborne material is removed by the area ventilation system and passed through HEPA filters before release to the environment.
Nuclear criticality	A criticality event was assumed to occur in the facility and the assumed criticality accident severity is based on guidance provided in NRC Regulatory Guide 3.35.

Source: LLNL 1996e.

M.5.3.7.2 *Accident Impacts*

The estimated impacts of the postulated accidents at each site are provided in Tables M.5.3.7.2–1 through M.5.3.7.2–6. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.3.7.1–3 and M.5.3.7.1–4, using the MACCS computer code. [Text deleted.]

Table M.5.3.7.2–1. Immobilized Disposition at the Ceramic Immobilization Facility Accident Impacts at Hanford Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Earthquake	5.7×10^{-6}	2.3×10^{-9}	4.5×10^{-8}	2.3×10^{-11}	3.3×10^{-4}	1.6×10^{-7}	1.0×10^{-5}
Glovebox fire	5.7×10^{-6}	2.3×10^{-9}	4.5×10^{-8}	2.3×10^{-11}	3.3×10^{-4}	1.6×10^{-7}	1.0×10^{-5}
Glovebox nuclear criticality	3.5×10^{-3}	1.4×10^{-6}	2.3×10^{-5}	1.2×10^{-8}	0.032	1.6×10^{-5}	1.0×10^{-5}
Calciner feed nuclear criticality	0.035	1.4×10^{-5}	2.3×10^{-4}	1.2×10^{-7}	0.32	1.6×10^{-4}	1.0×10^{-5}
Ceramic can drop	1.4×10^{-10}	5.7×10^{-14}	1.1×10^{-12}	5.7×10^{-16}	8.2×10^{-9}	4.1×10^{-12}	1.0×10^{-3}
Pellet container drop	1.4×10^{-12}	5.7×10^{-16}	1.1×10^{-14}	5.7×10^{-18}	8.2×10^{-11}	4.1×10^{-14}	1.0×10^{-3}
Dissolver spill	6.8×10^{-12}	2.7×10^{-15}	5.4×10^{-14}	2.7×10^{-17}	4.0×10^{-10}	2.0×10^{-13}	0.05
Calciner feed spill	2.0×10^{-11}	7.9×10^{-15}	1.6×10^{-13}	7.9×10^{-17}	1.2×10^{-9}	5.8×10^{-13}	0.05
Calciner product spill	5.0×10^{-9}	2.0×10^{-12}	4.0×10^{-11}	2.0×10^{-14}	2.9×10^{-7}	1.4×10^{-10}	0.05
Sintering furnace explosion	8.5×10^{-5}	3.4×10^{-8}	6.8×10^{-7}	3.4×10^{-10}	5.0×10^{-3}	2.5×10^{-6}	1.0×10^{-6}
Uncontrolled chemical reaction	4.0×10^{-6}	1.6×10^{-9}	3.2×10^{-8}	1.6×10^{-11}	2.3×10^{-4}	1.2×10^{-7}	1.0×10^{-6}
Nuclear criticality	3.5×10^{-3}	1.4×10^{-6}	2.3×10^{-5}	1.2×10^{-8}	0.032	1.6×10^{-5}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.7.1–3 and M.5.3.7.1–4 and the MACCS computer code.

Table M.5.3.7.2–2. Immobilized Disposition at the Ceramic Immobilization Facility Accident Impacts at Nevada Test Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Earthquake	3.9×10^{-6}	1.6×10^{-9}	7.2×10^{-8}	3.6×10^{-11}	7.4×10^{-6}	3.7×10^{-9}	1.0×10^{-5}
Glovebox fire	3.9×10^{-6}	1.6×10^{-9}	7.2×10^{-8}	3.6×10^{-11}	7.4×10^{-6}	3.7×10^{-9}	1.0×10^{-5}
Glovebox nuclear criticality	2.5×10^{-3}	1.0×10^{-6}	4.5×10^{-5}	2.3×10^{-8}	6.5×10^{-4}	3.3×10^{-7}	1.0×10^{-5}
Calciner feed nuclear criticality	0.025	1.0×10^{-5}	4.5×10^{-4}	2.3×10^{-7}	6.5×10^{-3}	3.3×10^{-6}	1.0×10^{-5}
Ceramic can drop	9.7×10^{-11}	3.9×10^{-14}	1.8×10^{-12}	9.0×10^{-16}	1.9×10^{-10}	9.3×10^{-14}	1.0×10^{-3}
Pellet container drop	9.7×10^{-13}	3.9×10^{-16}	1.8×10^{-14}	9.0×10^{-18}	1.9×10^{-12}	9.3×10^{-16}	1.0×10^{-3}
Dissolver spill	4.7×10^{-12}	1.9×10^{-15}	8.6×10^{-14}	4.3×10^{-17}	8.9×10^{-12}	4.5×10^{-15}	0.05
Calciner feed spill	1.4×10^{-11}	5.4×10^{-15}	2.5×10^{-13}	1.3×10^{-16}	2.6×10^{-11}	1.3×10^{-14}	0.05
Calciner product spill	3.4×10^{-9}	1.4×10^{-12}	6.3×10^{-11}	3.1×10^{-14}	6.5×10^{-9}	3.3×10^{-12}	0.05
Sintering furnace explosion	5.8×10^{-5}	2.3×10^{-8}	1.1×10^{-6}	5.4×10^{-10}	1.1×10^{-4}	5.6×10^{-8}	1.0×10^{-6}
Uncontrolled chemical reaction	2.7×10^{-6}	1.1×10^{-9}	5.0×10^{-8}	2.5×10^{-11}	5.2×10^{-6}	2.6×10^{-9}	1.0×10^{-6}
Nuclear criticality	2.5×10^{-3}	1.0×10^{-6}	4.5×10^{-5}	2.3×10^{-8}	6.5×10^{-4}	3.3×10^{-7}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.7.1–3 and M.5.3.7.1–4 and the MACCS computer code.

Table M.5.3.7.2–3. Immobilized Disposition at the Ceramic Immobilization Facility Accident Impacts at Idaho National Engineering Laboratory

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Earthquake	5.3×10^{-6}	2.1×10^{-9}	4.6×10^{-8}	2.3×10^{-5}	9.9×10^{-5}	4.9×10^{-8}	1.0×10^{-5}
Glovebox fire	5.3×10^{-6}	2.1×10^{-9}	4.6×10^{-8}	2.3×10^{-5}	9.9×10^{-5}	4.9×10^{-8}	1.0×10^{-5}
Glovebox nuclear criticality	3.4×10^{-3}	1.4×10^{-6}	2.7×10^{-5}	2.4×10^{-8}	8.7×10^{-3}	4.3×10^{-6}	1.0×10^{-5}
Calciner feed nuclear criticality	0.034	1.4×10^{-5}	2.7×10^{-4}	1.3×10^{-7}	0.086	4.3×10^{-5}	1.0×10^{-5}
Ceramic can drop	1.3×10^{-10}	5.3×10^{-14}	1.1×10^{-12}	5.7×10^{-16}	2.5×10^{-9}	1.2×10^{-12}	1.0×10^{-3}
Pellet container drop	1.3×10^{-12}	5.3×10^{-16}	1.1×10^{-14}	5.7×10^{-18}	2.5×10^{-11}	1.2×10^{-14}	1.0×10^{-3}
Dissolver spill	6.4×10^{-12}	2.5×10^{-15}	5.5×10^{-14}	2.8×10^{-17}	1.2×10^{-10}	5.9×10^{-14}	0.05
Calciner feed spill	1.9×10^{-11}	7.4×10^{-15}	1.6×10^{-13}	8.0×10^{-17}	3.5×10^{-10}	1.7×10^{-13}	0.05
Calciner product spill	4.6×10^{-9}	1.9×10^{-12}	4.0×10^{-11}	2.0×10^{-14}	8.6×10^{-8}	4.3×10^{-11}	0.05
Sintering furnace explosion	8.0×10^{-5}	3.2×10^{-8}	6.9×10^{-7}	3.4×10^{-10}	1.5×10^{-3}	7.4×10^{-7}	1.0×10^{-6}
Uncontrolled chemical reaction	3.7×10^{-6}	1.5×10^{-9}	3.2×10^{-8}	1.6×10^{-11}	6.9×10^{-5}	3.5×10^{-8}	1.0×10^{-6}
Nuclear criticality	3.4×10^{-3}	1.4×10^{-6}	2.7×10^{-5}	1.3×10^{-8}	8.7×10^{-3}	4.3×10^{-6}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.7.1–3 and M.5.3.7.1–4 and the MACCS computer code.

Table M.5.3.7.2–4. Immobilized Disposition at the Ceramic Immobilization Facility Accident Impacts at Pantex Plant

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Earthquake	2.3×10^{-6}	9.1×10^{-10}	5.3×10^{-7}	2.6×10^{-10}	1.1×10^{-4}	5.6×10^{-8}	1.0×10^{-5}
Glovebox fire	2.3×10^{-6}	9.1×10^{-10}	5.3×10^{-7}	2.6×10^{-10}	1.1×10^{-4}	5.6×10^{-8}	1.0×10^{-5}
Glovebox nuclear criticality	1.5×10^{-3}	6.2×10^{-7}	4.4×10^{-4}	2.2×10^{-7}	0.019	9.5×10^{-6}	1.0×10^{-5}
Calciner feed nuclear criticality	0.015	6.2×10^{-6}	4.4×10^{-3}	2.2×10^{-6}	0.19	9.5×10^{-5}	1.0×10^{-5}
Ceramic can drop	5.7×10^{-11}	2.3×10^{-14}	1.3×10^{-11}	6.6×10^{-15}	2.8×10^{-9}	1.4×10^{-12}	1.0×10^{-3}
Pellet container drop	5.7×10^{-13}	2.3×10^{-16}	1.3×10^{-13}	6.6×10^{-17}	2.8×10^{-11}	1.4×10^{-14}	1.0×10^{-3}
Dissolver spill	2.7×10^{-12}	1.1×10^{-15}	6.3×10^{-13}	3.2×10^{-16}	1.4×10^{-10}	6.7×10^{-14}	0.05
Calciner feed spill	8.0×10^{-12}	3.2×10^{-15}	1.8×10^{-12}	9.2×10^{-16}	3.9×10^{-10}	2.0×10^{-13}	0.05
Calciner product spill	2.0×10^{-9}	8.0×10^{-13}	4.6×10^{-10}	2.3×10^{-13}	9.8×10^{-8}	4.9×10^{-11}	0.05
Sintering furnace explosion	3.4×10^{-5}	1.4×10^{-8}	8.0×10^{-6}	4.0×10^{-9}	1.7×10^{-3}	8.5×10^{-7}	1.0×10^{-6}
Uncontrolled chemical reaction	1.6×10^{-6}	6.4×10^{-10}	3.7×10^{-7}	1.9×10^{-10}	7.9×10^{-5}	3.9×10^{-8}	1.0×10^{-6}
Nuclear criticality	1.5×10^{-3}	6.2×10^{-7}	4.4×10^{-4}	2.2×10^{-7}	0.019	9.5×10^{-6}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.7.1–3 and M.5.3.7.1–4 and the MACCS computer code.

Table M.5.3.7.2–5. Immobilized Disposition at the Ceramic Immobilization Facility Accident Impacts at Oak Ridge Reservation

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Earthquake	5.2×10^{-6}	2.1×10^{-9}	9.2×10^{-7}	4.6×10^{-10}	8.0×10^{-4}	4.0×10^{-7}	1.0×10^{-5}
Glovebox fire	5.2×10^{-6}	2.1×10^{-9}	9.2×10^{-7}	4.6×10^{-10}	8.0×10^{-4}	4.0×10^{-7}	1.0×10^{-5}
Glovebox nuclear criticality	3.2×10^{-3}	1.3×10^{-6}	5.8×10^{-4}	2.9×10^{-7}	0.13	6.3×10^{-5}	1.0×10^{-5}
Calciner feed nuclear criticality	0.032	1.3×10^{-5}	5.8×10^{-3}	2.9×10^{-6}	1.3	6.3×10^{-4}	1.0×10^{-5}
Ceramic can drop	1.3×10^{-10}	5.3×10^{-14}	2.3×10^{-11}	1.2×10^{-14}	2.0×10^{-8}	1.0×10^{-11}	1.0×10^{-3}
Pellet container drop	1.3×10^{-12}	5.3×10^{-16}	2.3×10^{-13}	1.2×10^{-16}	2.0×10^{-10}	1.0×10^{-13}	1.0×10^{-3}
Dissolver spill	6.3×10^{-12}	2.5×10^{-15}	1.1×10^{-12}	5.5×10^{-16}	9.6×10^{-10}	4.8×10^{-13}	0.05
Calciner feed spill	1.8×10^{-11}	7.3×10^{-15}	3.2×10^{-12}	1.6×10^{-15}	2.8×10^{-9}	1.4×10^{-12}	0.05
Calciner product spill	4.6×10^{-9}	1.8×10^{-12}	8.1×10^{-10}	4.0×10^{-13}	7.0×10^{-7}	3.5×10^{-10}	0.05
Sintering furnace explosion	7.9×10^{-5}	3.2×10^{-8}	1.4×10^{-5}	6.9×10^{-9}	0.012	6.0×10^{-6}	1.0×10^{-6}
Uncontrolled chemical reaction	3.7×10^{-6}	1.5×10^{-9}	6.5×10^{-7}	3.2×10^{-10}	5.6×10^{-4}	2.8×10^{-7}	1.0×10^{-6}
Nuclear criticality	3.2×10^{-3}	1.3×10^{-6}	5.8×10^{-4}	2.9×10^{-7}	0.13	6.3×10^{-5}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m or the site boundary (1,000 m for this facility at ORR), 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.7.1–3 and M.5.3.7.1–4 and the MACCS computer code.

Table M.5.3.7.2–6. Immobilized Disposition at the Ceramic Immobilization Facility Accident Impacts at Savannah River Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Earthquake	3.7×10^{-6}	1.5×10^{-9}	7.3×10^{-8}	3.6×10^{-11}	3.5×10^{-4}	1.8×10^{-7}	1.0×10^{-5}
Glovebox fire	3.7×10^{-6}	1.5×10^{-9}	7.3×10^{-8}	3.6×10^{-11}	3.5×10^{-4}	1.8×10^{-7}	1.0×10^{-5}
Glovebox nuclear criticality	2.3×10^{-3}	9.1×10^{-7}	4.0×10^{-5}	2.0×10^{-8}	0.041	2.0×10^{-5}	1.0×10^{-5}
Calciner feed nuclear criticality	0.023	9.1×10^{-6}	4.0×10^{-4}	2.0×10^{-7}	0.41	2.0×10^{-4}	1.0×10^{-5}
Ceramic can drop	9.3×10^{-11}	3.7×10^{-14}	1.8×10^{-12}	9.1×10^{-16}	8.8×10^{-9}	4.4×10^{-12}	1.0×10^{-3}
Pellet container drop	9.3×10^{-13}	3.7×10^{-16}	1.8×10^{-14}	9.1×10^{-18}	8.8×10^{-11}	4.4×10^{-14}	1.0×10^{-3}
Dissolver spill	4.5×10^{-12}	1.8×10^{-15}	8.8×10^{-14}	4.4×10^{-17}	4.2×10^{-10}	2.1×10^{-13}	0.05
Calciner feed spill	1.3×10^{-11}	5.2×10^{-15}	2.6×10^{-13}	1.3×10^{-16}	1.2×10^{-9}	6.2×10^{-13}	0.05
Calciner product spill	3.3×10^{-9}	1.3×10^{-12}	6.4×10^{-11}	3.2×10^{-14}	3.1×10^{-7}	1.6×10^{-10}	0.05
Sintering furnace explosion	5.6×10^{-5}	2.2×10^{-8}	1.1×10^{-6}	5.5×10^{-10}	5.3×10^{-3}	2.7×10^{-6}	1.0×10^{-6}
Uncontrolled chemical reaction	2.6×10^{-6}	1.0×10^{-9}	5.1×10^{-8}	2.6×10^{-11}	2.5×10^{-4}	1.2×10^{-7}	1.0×10^{-6}
Nuclear criticality	2.3×10^{-3}	9.1×10^{-7}	4.0×10^{-5}	2.0×10^{-8}	0.041	2.0×10^{-5}	1.0×10^{-6}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.7.1–3 and M.5.3.7.1–4 and the MACCS computer code.

M.5.3.8 Evolutionary Light Water Reactor

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for an evolutionary LWR in the *Evolutionary/Advanced Light Water Reactor Data Report*. The studies postulated a set of accident scenarios that were representative of the risks and consequences for workers and the public that can be expected if the facility were constructed and operated. The advanced boiling reactor studies were selected as representative studies for the evolutionary LWRs.

M.5.3.8.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents from operating the facility were identified as candidates to represent the risks to workers and the public. Through a screening process, four evaluation basis accidents and two beyond evaluation basis accidents were selected for further definition and analysis. Supporting information for these accidents has been documented in *Assessment of Radioactive Releases to the Environment Due to the Incorporation of Tritium Targets into an Advanced Light Water Reactor to Produce Tritium,* October 1995.

Evaluation Basis Accidents

Failure of Small Primary Coolant Line Outside of Containment

This accident postulated the rupture of an instrument line outside the drywell but inside the reactor building. It is not possible to isolate the leak. The flow from the instrument line is limited by a 0.64-centimeter (0.25-inch) diameter flow-restricting orifice inside the drywell. The total integrated mass of fluid released into the reactor building is 5,442 kg (12,000 pounds [lb]), with approximately 2,270 kg (5,000 lb) flashed into steam. The accident sequence is terminated by the orderly shutdown and depressurization of the reactor. Table M.5.3.8.1–1 presents the source term released to the environment. The analysis did not estimate the accident annual frequency of occurrence. It is expected that the postulated annual frequency of occurrence would range from 0.01/yr to 1.0×10^{-4}/yr. For the purpose of calculating the point estimate of risk for the postulated accident, the accident annual frequency of occurrence is assumed to be 1.0×10^{-3}/yr.

Steam System Piping Break Outside Containment

This accident postulated a large steam line break outside of containment downstream of the outermost isolation valve. The plant is designed to immediately detect the break and initiate isolation of the broken line. Table M.5.3.8.1–1 presents the source term released to the environment. The analysis did not estimate the accident annual frequency of occurrence. It is expected that the postulated annual frequency of occurrence would range from 1.0×10^{-4}/yr to 1.0×10^{-6}/yr. For the purpose of calculating the point estimate of risk for the postulated accident, the accident annual frequency of occurrence is assumed to be 1.0×10^{-5}/yr.

Cleanup Water Line Break Outside Containment

This accident postulated a large cleanup water line break outside of containment. The analysis assumed that the non-filtered inventory in both the regenerative and non-regenerative heat exchangers is released through the break. The leak is automatically isolated approximately 75 seconds after the break. Table M.5.3.8.1–1 presents the source term released to the environment. The analysis did not estimate the accident annual frequency of occurrence. It is expected that the postulated annual frequency of occurrence would range from 1.0×10^{-4}/yr to 1.0×10^{-6}/yr. For the purpose of calculating the point estimate of risk for the postulated accident, the accident annual frequency of occurrence is assumed to be 1.0×10^{-5}/yr.

Table M.5.3.8.1–1. Advanced Boiling Water Reactor Evaluation Basis Accident Source Terms

	Accident Scenario			
Accident Parameter	**Failure of Small Primary Coolant Line Outside Containment**	**Steam System Piping Break Outside Containment**	**Cleanup Water Line Break Outside Containment**	**Fuel Handling**
Frequency of occurrence[a]	1.0×10^{-3}	1.0×10^{-5}	1.0×10^{-5}	1.0×10^{-5}
Isotope Released to Environment (Ci)				
I-131	4.2	43	2.4	130
I-132	34	410	5.5	160
I-133	25	260	5.9	120
I-134	49	720	8.3	6.0×10^{-6}
I-135	36	390	6.8	21
Xe-131m	0	2.9×10^{-4}	0	84
Xe-133m	0	5.5×10^{-3}	0	1.1×10^{3}
Xe-133	0	0.14	0	2.7×10^{4}
Xe-135m	0	0.47	0	220
Xe-135	0	1.3	0	1.9×10^{4}
Xe-137	0	2.0	0	2.1×10^{-10}
Xe-138	0	1.5	0	4.3×10^{-10}
Xe-139	0	0.70	0	0
Kr-83m	0	0.040	0	3.8
Kr-85m	0	0.078	0	55
Kr-85	0	1.9×10^{-4}	0	250
Kr-87	0	0.24	0	7.1×10^{-3}
Kr-88	0	0.23	0	14
Kr-89	0	1.6	0	8.1×10^{-11}
Kr-90	0	0.42	0	0

[a] Midpoint of the estimated frequency range.
Source: GE nda.

Fuel Handling Accident

This accident postulated a spent fuel assembly dropped into the reactor core. The analysis assumed that some rods in the dropped assembly and in the struck assembly fail. Table M.5.3.8.1–1 presents the source term released to the environment. The analysis did not estimate the accident annual frequency of occurrence. It is expected that the postulated annual frequency of occurrence would range from 1.0×10^{-4}/yr to 1.0×10^{-6}/yr. For the purpose of calculating the point estimate of risk for the postulated accident, the accident annual frequency of occurrence is assumed to be 1.0×10^{-5}/yr.

Beyond Evaluation Basis Accidents

Chapter 19 of the *Advanced Boiling Water Reactor Standard Safety Analysis Report* evaluated beyond design basis accidents that were initiated by either internal events (for example, a sequence of equipment failures) or external events (for example, severe natural phenomena such as beyond design basis earthquakes). The evaluation of external event-initiated accidents did not present accident frequency data, release fractions, or source term data that could be used to analyze the accident consequences and risks for this class of accident in this document.

Numerous internal event-initiated accidents were evaluated in Chapter 19 of the Advanced Boiling Water Reactor (BWR) Standard SAR. The accidents that had a common source term were grouped together and evaluated as a single accident, and a single total annual frequency of occurrence was defined for the group. Release fractions and the annual frequency of occurrence were defined for two accidents. The annual frequency of occurrence for the ten accidents in Chapter 19 of the Advanced BWR Standard SAR ranged from 7.0×10^{-8}/yr to less than 1.0×10^{-10}/yr. Two of the ten accidents had an annual frequency of occurrence greater than 1.0×10^{-8}/yr. These two accidents were selected for evaluation.

Anticipated Transient Without Scram and Loss of Core Cooling

The postulated accident is an anticipated transient without scram with the loss of core cooling. Due to the loss of core cooling, core damage results, the vessel fails in approximately 1 hour, and the containment fails in approximately 19 hours. The source term is presented in Table M.5.3.8.1–2. The annual frequency of occurrence for this accident is1.3×10^{-7}/yr.

Large Break Loss of Coolant Accident and Loss of Core Cooling

The postulated accident is represented by a source term that is common for a group of accidents. The group of accidents includes the following:

- Loss of all core cooling, vessel failure at high pressure, firewater addition system switched to drywell spray mode, containment overpressure protection system rupture disk ruptures, and release negligible—less than 0.1 percent volatile fission products.

- Loss of all core cooling, vessel failure at high pressure, passive flooder and drywell spray available, containment overpressure protection system rupture disk ruptures, and release negligible—less than 0.1 percent volatile fission products.

- Large break loss of coolant accident, loss of all core cooling, firewater addition system switched to drywell spray mode, containment overpressure protection system rupture disk ruptures, and release negligible—less than 0.1 percent volatile fission products.

- Station blackout with reactor core isolation cooling operating for 8 hours, offsite power restored at 8 hours, firewater addition system switched to drywell spray mode, containment overpressure protection system rupture disk ruptures, and release negligible—less than 0.1 percent volatile fission products.

- Loss of all core cooling, vessel failure at low pressure, passive flooder available, containment overpressure protection system rupture disk ruptures, and release negligible—less than 0.1 percent volatile fission products.

- Loss of all core cooling, vessel failure at low pressure, firewater addition system switched to drywell spray mode, containment overpressure protection system rupture disk ruptures, and release negligible—less than 0.1 percent volatile fission products.

The source term is presented in Table M.5.3.8.1–2. The annual frequency of occurrence for the group of accidents is 2.1×10^{-8}/yr.

Table M.5.3.8.1–2. Advanced Boiling Water Reactor Beyond Evaluation Basis Accident Source Terms

	Accident Scenario	
Accident Parameter	**Anticipated Transient Without Scram and Loss of Core Cooling**	**Large Break Loss of Coolant Accident and Loss of Core Cooling[a]**
Frequency of occurrence[b]	1.3×10^{-7}	2.1×10^{-8}
Isotope Released to Environment (Ci)		
Kr-85	2.3×10^{6}	5.2×10^{5}
Kr-85m	1.0×10^{6}	2.3×10^{7}
Kr-87	1.7×10^{6}	3.9×10^{7}
Kr-88	2.3×10^{6}	5.2×10^{7}
Rb-86	0.53	0.30
I-131	2.6×10^{3}	18
I-132	3.8×10^{3}	25
I-133	4.8×10^{3}	31
I-134	5.3×10^{3}	35
I-135	4.7×10^{3}	31
Xe-133	9.1×10^{6}	2.1×10^{8}
Xe-135	6.7×10^{6}	1.5×10^{8}
Cs-134	200	110
Cs-136	150	86
Cs-137	210	120

[a] Representative accident description for a group of accidents with the same source term.
[b] Total frequency for a group of accidents with same source term.
Source: Source term derived from accident release fractions (GE nda) and core inventory (TTI 1995b).

M.5.3.8.2 *Accident Impacts*

The estimated impacts of the postulated accidents at each site are provided in Tables M.5.3.8.2–1 through M.5.3.8.2–6. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.3.8.1–1 and M.5.3.8.1–2 using the MACCS computer code. [Text deleted.]

Table M.5.3.8.2–1. Evolutionary Light Water Reactor Accident Impacts at Hanford Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (person-rem)**	**Number of Latent Cancer/Prompt Fatalities[b]**	**Accident Frequency (per year)**
Failure of small primary coolant line outside containment	5.6×10^{-3}	2.2×10^{-6}/ 0	5.6×10^{-4}	2.8×10^{-7}/ 0	0.10	5.2×10^{-5}/ 0	1.0×10^{-3}
Scram system piping break outside containment	0.061	2.5×10^{-5}/ 0	6.2×10^{-3}	3.1×10^{-6}/ 0	1.11	5.5×10^{-4}/ 0	1.0×10^{-5}
Cleanup water line break outside containment	1.7×10^{-3}	6.7×10^{-7}/ 0	1.7×10^{-4}	8.4×10^{-8}/ 0	0.036	1.8×10^{-5}/ 0	1.0×10^{-5}
Fuel handling	0.071	2.8×10^{-5}/ 0	8.6×10^{-3}	4.3×10^{-6}/ 0	2.0	1.0×10^{-3}/ 0	1.0×10^{-5}
Anticipated transient with scram and loss of core cooling	40.4	0.021/ 0	5.4	2.7×10^{-3}/ 0	1,650	0.82/ 0	1.3×10^{-7}
Large break loss of coolant accident and loss of core cooling	422	0.089/ 0.40	85.6	0.08/ 9.7×10^{-3}	11,700	5.9/ 0	2.1×10^{-8}
[Text deleted.]							

[a] Increase likelihood (or probability) of cancer or prompt fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values. Advanced BWR data was used as surrogate data for the evolutionary LWR.

Source: Calculated using the source terms in Tables M.5.3.8.1–1 and M.5.3.8.1–2 and the MACCS computer code.

Table M.5.3.8.2–2. Evolutionary Light Water Reactor Accident Impacts at Nevada Test Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (person-rem)**	**Number of Latent Cancer/Prompt Fatalities[b]**	**Accident Frequency (per year)**
Failure of small primary coolant line outside containment	3.9×10^{-3}	1.5×10^{-6}/ 0	8.3×10^{-5}	4.1×10^{-8} 0	3.2×10^{-3}	1.6×10^{-6}/ 0	1.0×10^{-3}
Scram system piping break outside containment	0.043	1.7×10^{-5}/ 0	9.0×10^{-4}	4.5×10^{-7}/ 0	0.033	1.7×10^{-5}/ 0	1.0×10^{-5}
Cleanup water line break outside containment	1.2×10^{-3}	4.7×10^{-7}/ 0	2.5×10^{-5}	1.2×10^{-8}/ 0	1.1×10^{-3}	5.6×10^{-7}/ 0	1.0×10^{-5}
Fuel handling	0.048	1.9×10^{-5}/ 0	1.5×10^{-3}	7.6×10^{-7}/ 0	0.051	2.6×10^{-5}/ 0	1.0×10^{-5}
Anticipated transient with scram and loss of core cooling	27	0.014/ 0	1.0	5.0×10^{-4}/ 0	50.9	0.026/ 0	1.3×10^{-7}
Large break loss of coolant accident and loss of core cooling	279	0.087/ 0.25	17.8	0.15/ 2.4×10^{-4}	118	0.059/ 0	2.1×10^{-8}
[Text deleted.]							

[a] Increase likelihood (or probability) of cancer or prompt fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values. Advanced BWR data was used as surrogate data for the evolutionary LWR.

Source: Calculated using the source terms in Tables M.5.3.8.1–1 and M.5.3.8.1–2 and the MACCS computer code.

Table M.5.3.8.2–3. Evolutionary Light Water Reactor Accident Impacts at Idaho National Engineering Laboratory

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (person-rem)**	**Number of Latent Cancer/ Prompt Fatalities[b]**	**Accident Frequency (per year)**
Failure of small primary coolant line outside containment	5.5×10^{-3}	2.2×10^{-6}/ 0	5.0×10^{-5}	2.5×10^{-8}/ 0	0.043	2.2×10^{-5}/ 0	1.0×10^{-3}
Scram system piping break outside containment	0.061	2.4×10^{-5}/ 0	5.4×10^{-4}	2.7×10^{-7}/ 0	0.45	2.3×10^{-4}/ 0	1.0×10^{-5}
Cleanup water line break outside containment	1.6×10^{-3}	6.6×10^{-7}/ 0	1.5×10^{-5}	7.6×10^{-9}/ 0	0.015	7.6×10^{-6}/ 0	1.0×10^{-5}
Fuel handling	0.068	2.7×10^{-5}/ 0	1.0×10^{-3}	5.1×10^{-7}/ 0	0.65	3.3×10^{-4}/ 0	1.0×10^{-5}
Anticipated transient with scram and loss of core cooling	36.3	0.020/ 0	0.66	3.3×10^{-4}/ 0	689	0.35/ 0	1.3×10^{-7}
Large break loss of coolant accident and loss of core cooling	385	0.071/ 0.36	12.4	0.010/ 0	1,150	0.57/ 0	2.1×10^{-8}
[Text deleted.]							

[a] Increase likelihood (or probability) of cancer or prompt fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values. Advanced BWR data was used as surrogate data for the evolutionary LWR.

Source: Calculated using the source terms in Tables M.5.3.8.1–1 and M.5.3.8.1–2 and the MACCS computer code.

Table M.5.3.8.2–4. Evolutionary Light Water Reactor Accident Impacts at Pantex Plant

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (person-rem)**	**Number of Latent Cancer/Prompt Fatalities[b]**	**Accident Frequency (per year)**
Failure of small primary coolant line outside containment	2.2×10^{-3}	8.8×10^{-7}/ 0	1.6×10^{-3}	7.9×10^{-7}/ 0	0.063	3.2×10^{-5}/ 0	1.0×10^{-3}
Scram system piping break outside containment	0.024	9.7×10^{-6}/ 0	0.017	8.7×10^{-6}/ 0	0.68	3.4×10^{-4}/ 0	1.0×10^{-5}
Cleanup water line break outside containment	6.5×10^{-4}	2.6×10^{-7}/ 0	4.7×10^{-4}	2.3×10^{-7}/ 0	0.021	1.1×10^{-5}/ 0	1.0×10^{-5}
Fuel handling	0.027	1.1×10^{-5}/ 0	0.020	9.9×10^{-6}/ 0	0.97	4.9×10^{-4}/ 0	1.0×10^{-5}
Anticipated transient with scram and loss of core cooling	16.4	7.2×10^{-3}/ 0	12.0	6.5×10^{-3}/ 0	813	0.41/ 0	1.3×10^{-7}
Large break loss of coolant accident and loss of core cooling	162	0.095/ 0.080	129	0.10/ 0.047	4,660	2.3/ 0	2.1×10^{-8}
[Text deleted.]							

[a] Increase likelihood (or probability) of cancer or prompt fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values. Advanced BWR data was used as surrogate data for the evolutionary LWR.

Source: Calculated using the source terms in Tables M.5.3.8.1–1 and M.5.3.8.1–2 and the MACCS computer code.

Table M.5.3.8.2–5. Evolutionary Light Water Reactor Accident Impacts at Oak Ridge Reservation

	Worker at 665 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (person-rem)**	**Number of Latent Cancer/Prompt Fatalities[b]**	**Accident Frequency (per year)**
Failure of small primary coolant line outside containment	6.9×10^{-3}	2.8×10^{-6}/ 0	6.9×10^{-3}	3.5×10^{-6}/ 0	0.43	2.2×10^{-4}/ 0	1.0×10^{-3}
Scram system piping break outside containment	0.076	3.0×10^{-5}/ 0	0.076	3.8×10^{-5}/ 0	4.6	2.3×10^{-3}/ 0	1.0×10^{-5}
Cleanup water line break outside containment	2.1×10^{-3}	8.3×10^{-7}/ 0	2.1×10^{-3}	1.0×10^{-6}/ 0	0.15	7.4×10^{-5}/ 0	1.0×10^{-5}
Fuel handling	0.085	3.4×10^{-5}/ 0	0.085	4.3×10^{-5}/ 0	7.8	3.9×10^{-3}/ 0	1.0×10^{-5}
Anticipated transient with scram and loss of core cooling	50.6	0.028/ 0	50.6	0.035/ 0	6,250	3.1/ 0	1.3×10^{-7}
Large break loss of coolant accident and loss of core cooling	474	0.058/ 0.56	474	0.072/ 0.56	45,100	22.2/ 0	2.1×10^{-8}
[Text deleted.]							

[a] Increase likelihood (or probability) of cancer or prompt fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m or the site boundary [665 m for this facility at ORR], 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 the accident has occurred.

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

Note: All values are mean values. Advanced BWR data was used as surrogate data for the evolutionary LWR.

Source: Calculated using the source terms in Tables M.5.3.8.1–1 and M.5.3.8.1–2 and the MACCS computer code.

Table M.5.3.8.2–6. Evolutionary Light Water Reactor Accident Impacts at Savannah River Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (rem)**	**Probability of Latent Cancer/ Prompt Fatality[a]**	**Dose (person-rem)**	**Number of Latent Cancer/Prompt Fatalities[b]**	**Accident Frequency (per year)**
Failure of small primary coolant line outside containment	3.5×10^{-3}	1.4×10^{-6}/ 0	4.2×10^{-5}	2.1×10^{-8}/ 0	0.15	7.5×10^{-5}/ 0	1.0×10^{-3}
Scram system piping break outside containment	0.039	1.6×10^{-5}/ 0	4.5×10^{-4}	2.3×10^{-7}/ 0	1.6	7.9×10^{-4}/ 0	1.0×10^{-5}
Cleanup water line break outside containment	1.1×10^{-3}	4.2×10^{-7}/ 0	1.3×10^{-5}	6.3×10^{-9}/ 0	0.053	2.6×10^{-5}/ 0	1.0×10^{-5}
Fuel handling	0.045	1.8×10^{-5}/ 0	7.7×10^{-4}	3.9×10^{-7}/ 0	2.5	1.3×10^{-3}/ 0	1.0×10^{-5}
Anticipated transient with scram and loss of core cooling	27.2	0.013/ 0	0.52	2.6×10^{-4}/ 0	2,260	1.1/ 0	1.3×10^{-7}
Large break loss of coolant accident and loss of core cooling	279	0.095/ 0.21	8.4	4.9×10^{-3}/ 0	8,640	4.3/ 0	2.1×10^{-8}
[Text deleted.]							

[a] Increase likelihood (or probability) of cancer or prompt fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values. Advanced BWR data was used as surrogate data for the evolutionary LWR.

Source: Calculated using the source terms in Tables M.5.3.8.1–1 and M.5.3.8.1–2 and the MACCS computer code.

M.5.3.9 Plutonium Conversion

Studies of evaluation basis accidents and beyond evaluation basis accidents have been performed for a Pu conversion facility in the *Data Report for Plutonium Conversion Facility*. The studies postulated a set of accident scenarios that were representative of the risks and consequences for workers and the public that can be expected in the facility were constructed and operated. Although not all potential accidents were addressed, those that were postulated have consequences and risks that are expected to envelop the consequences and risks of an operating facility. In this manner, no other credible accidents with an expected frequency of occurrence larger than 1.0×10^{-7} per year are anticipated that will have consequences and risks larger than those described in this section.

M.5.3.9.1 *Accident Scenarios and Source Terms*

A wide range of hazardous conditions and potential accidents were identified as candidates to represent the risks to workers and the public of operating the facility. Through a screening process, four evaluation basis accidents and four beyond evaluation basis accidents were selected for further definition and analysis. Descriptive information on these accidents is provided in Tables M.5.3.9.1–1 and M.5.3.9.1–2. Accidents source term information is provided in Tables M.5.3.9.1–3 through M.5.3.9.1–5. Descriptions of accident scenarios are provided in Table M.5.3.9.1–6.

Table M.5.3.9.1–1. Evaluation Basis Accident Scenarios for the Plutonium Conversion Facility

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Fire on the loading dock	1.0×10^{-4} to 1.0×10^{-3}	18 g Pu	0.8 g Pu
Fire in a process cell	1.0×10^{-5} to 1.0×10^{-3}	24 g Pu	4.8×10^{-6} g Pu
Deflagration inside a glovebox	1.0×10^{-5} to 1.0×10^{-3}	10 kg Pu	1.0×10^{-3} g Pu
Forklift breach of containment	4.5×10^{-5}	4 kg PuO_2	1.7×10^{-9} g Pu

Source: LANL 1996c.

Table M.5.3.9.1–2. Beyond Evaluation Basis Accident Scenarios for the Plutonium Conversion Facility

Accident Scenario	Accident Frequency (per year)	Source Term at Risk	Source Term Released to Environment
Nuclear criticality	$<1.0 \times 10^{-7}$	5.0×10^{17} fissions; gaseous by-products released	[a]
Beyond design basis fire in a process cell	$<1.0 \times 10^{-7}$	24 g Pu	0.034 g Pu
Oxyacetylene explosion in a process cell	$<1.0 \times 10^{-7}$	10 kg Pu	50 g Pu
Beyond evaluation basis earthquake	$<1.0 \times 10^{-7}$	10 kg Pu	25 g Pu

[a] See Table M.5.3.9.1–3.

Source: LANL 1996c.

Table M.5.3.9.1–3. Plutonium Conversion Facility Criticality Source Terms

Nuclide	Produced (Ci)	Released (Ci)
Kr-83m	5.5	2.75
Kr-85m	3.55	1.75
Kr-85	4.05×10^{-4}	2.0×10^{-4}
Kr-87	21.5	11
Kr-88	11.5	6
Kr-89	650	325
Xe-131m	5.0×10^{-3}	2.5×10^{-3}
Xe-133m	0.11	0.05
Xe-133	1.35	0.75
Xe-135m	165	85
Xe-135	20.5	10
Xe-137	2,450	1,225
Xe-138	550	275
I-131	0.55	0.025
I-132	60	3
I-133	8	0.4
I-134	215	11
I-135	22.5	1.0

Source: LANL 1996c.

Table M.5.3.9.1–4. Plutonium Conversion Facility Evaluation Basis Source Terms

	Accident Scenario			
Accident Parameter	**Fire on Loading Dock**	**Fire in Process Cell**	**Deflagration Inside a Glovebox**	**Forklift Breach of Containment**
Frequency of occurrence (per year)	5.0×10^{-4}[a]	1.0×10^{-4}[a]	1.0×10^{-4}[a]	4.5×10^{-5}
Pu released to environment (g)	0.8	4.8×10^{-6}	1.0×10^{-3}	1.7×10^{-9}
Isotope Released to Environment (Ci)				
Pu-238	2.9×10^{-3}	1.74×10^{-8}	3.62×10^{-6}	6.15×10^{-12}
Pu-239	0.0448	2.69×10^{-7}	5.60×10^{-5}	9.57×10^{-11}
Pu-240	0.0147	8.83×10^{-8}	1.84×10^{-5}	3.13×10^{-11}
Pu-241	0.0606	3.63×10^{-7}	7.57×10^{-5}	1.29×10^{-10}
Pu-242	4.75×10^{-6}	2.85×10^{-11}	5.94×10^{-9}	1.01×10^{-14}
Am-241	3.19×10^{-4}	1.92×10^{-9}	3.99×10^{-7}	6.78×10^{-13}

[a] Midpoint of the estimated frequency range.
Source: Derived from Table M.5.1.3.4–3 and M.5.3.9.1–1.

Table M.5.3.9.1–5. Plutonium Conversion Facility Beyond Evaluation Basis Accident Source Terms

	Accident Scenario			
Accident Parameter	**Nuclear Criticality**	**Beyond Evaluation Basis Fire in Process Cell**	**Oxyacetylene Explosion in Process Cell**	**Beyond Design Basis Earthquake**
Frequency of occurrence (per year)	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}	1.0×10^{-7}
Pu released to environment (g)	NA	0.034	50	25
Fissions	5.0×10^{17}	NA	NA	NA
Isotope Released to Environment (Ci)				
Pu-238	0	1.23×10^{-4}	0.181	0.0905
Pu-239	0	1.90×10^{-3}	2.80	1.40
Pu-240	0	6.26×10^{-4}	0.920	0.460
Pu-241	0	2.57×10^{-3}	3.79	1.89
Pu-242	0	2.02×10^{-7}	2.97×10^{-4}	1.49×10^{-4}
Am-241	0	1.36×10^{-5}	0.02	9.98×10^{-3}
Kr-83m	2.75	0	0	0
Kr-85m	1.75	0	0	0
Kr-85	2.0×10^{-4}	0	0	0
Kr-87	11	0	0	0
Kr-88	6	0	0	0
Kr-89	325	0	0	0
Xe-131m	2.5×10^{-3}	0	0	0
Xe-133m	0.05	0	0	0
Xe-133	0.75	0	0	0
Xe-135m	85	0	0	0
Xe-135	10	0	0	0
Xe-137	1,225	0	0	0
Xe-138	275	0	0	0
I-131	0.025	0	0	0
I-132	3	0	0	0
I-133	0.4	0	0	0
I-134	11	0	0	0
I-135	1.0	0	0	0

Note: NA=not applicable.

Source: Derived from Tables M.5.1.3.4–3, M.5.3.9.1–2, and M.5.3.9.1–3.

Table M.5.3.9.1–6. Accident Scenario Descriptions for the Plutonium Conversion Facility

Accident Scenario	Accident Description
Evaluation Basis Accidents	
Fire on the loading dock	The fire is caused by welding, cleaning solvents, electrical shorts, or other miscellaneous causes. The scenario assumes an open garage door and that a single drum of combustible waste is involved in the fire.
Fire in the process cell	It is assumed that a process cell contains a glovebox used for final processing of Pu oxide powder. The gloves, stowed outside the glovebox, are coated with a layer of Pu dust. A flammable cleaning liquid such as acetone or isopropyl alcohol is brought into the process cell in violation of operating procedures, spills and ignites. The initial extent and intensity of the fire are sufficient to completely incinerate the gloves. The sprinkler system activates and protects the glovebox from further damage. The ventilation system with HEPA filters continues to function throughout the accident.
Deflagration inside a glovebox	The bounding evaluation basis explosion is a deflagration of a flammable gas mixture inside a glovebox. It is assumed that through some unforeseen set of failures, a combustible gas mixture accumulates inside a glovebox and is ignited, possibly by an electrical spark from an operating electrical device. The deflagration blows out the HEPA filter from the glovebox ventilation system exit. Gloves may also be blown out. The room volumes are sufficient to attenuate the pressure wave to levels below that needed to damage the building ventilation system HEPA filters.
Forklift breach of containment	The most catastrophic case of leak or spill or nuclear material would result from a forklift or other large vehicle running over a package or nuclear material, breaching the containment, and causing an airborne release to the room. Three-stage HEPA filtration is available for the facility exhaust to limit the release to the environment.
Beyond Evaluation Basis Accidents	
Nuclear criticality	There will not be sufficient quantities of Pu solutions at the facility to cause a criticality accident. The most likely cause of a criticality event involving Pu oxides would be improper stacking or handling of bulk nuclear material. Multiple operational errors in the material spacing, packing density, manner and type of containment, and maximum quantities of fissile materials permitted in the area would be required for the postulated criticality accident to occur.
Beyond evaluation basis fire in a process cell	A typical fire with coincident failures of two or more major safety systems constitutes a beyond evaluation basis fire. The evaluation postulated the fire in a process cell, discussed above, with the sprinkler system and ventilation system with HEPA filtration inoperative during the accident.
Oxyacetylene explosion in a process cell	The evaluation postulated the explosion of a welding rig oxyacetylene bottle in a process cell. The explosion is sufficient to blow out the HEPA filters and cause significant damage to the ventilation system and nearby equipment.
Beyond evaluation basis earthquake	The following assumptions were used in the evaluation: (1) the earthquake disables the ventilation system; (2) there is significant structural damage to the building but it does not totally collapse; (3) a ceiling slab falls on the glovebox; (4) the process cell with the most material at risk is located on an outside wall; (5) the outside wall cracks; and (6) the wind is blowing and the cracks are located in the lee side of the building.

Source: LANL 1996c.

M.5.3.9.2 *Accident Impacts*

The estimated impacts of the postulated accidents at each site are provided in the Tables M.5.3.9.2–1 through M.5.3.9.2–6. The dose and cancer fatality estimates are based on the analysis of the accident source terms in Tables M.5.3.9.1–4 and M.5.3.9.1–5 using the MACCS computer code. [Text deleted.].

Table M.5.3.9.2–1. Plutonium Conversion Facility Accident Consequences at Hanford Site

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Fire on the loading dock	0.24	9.7×10^{-5}	7.7×10^{-3}	3.8×10^{-6}	14.0	7.0×10^{-3}	5.0×10^{-4}
Fire in a process cell	1.4×10^{-6}	5.8×10^{-10}	4.6×10^{-8}	2.3×10^{-11}	8.4×10^{-5}	4.2×10^{-8}	1.0×10^{-4}
Deflagration inside a glovebox	3.0×10^{-4}	1.2×10^{-7}	9.6×10^{-6}	4.8×10^{-9}	0.018	8.7×10^{6}	1.0×10^{-4}
Forklift breach of containment	5.1×10^{-10}	2.0×10^{-13}	1.6×10^{-11}	8.2×10^{-15}	3.0×10^{-8}	1.5×10^{-11}	4.5×10^{-5}
Nuclear criticality	5.2×10^{-4}	2.1×10^{-7}	1.7×10^{-5}	8.4×10^{-9}	3.4×10^{-3}	1.7×10^{-6}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	0.010	4.1×10^{-6}	3.3×10^{-4}	1.6×10^{-7}	0.59	3.0×10^{-4}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	15.0	6.8×10^{-3}	0.48	2.4×10^{-4}	873	0.44	1.0×10^{-7}
Beyond evaluation basis earthquake	7.5	3.1×10^{-3}	0.24	1.2×10^{-4}	436	0.22	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.9.1–4 and M.5.3.9.1–5 and the MACCS computer code.

Table M.5.3.9.2–2. Plutonium Conversion Facility Accident Impacts at Nevada Test Site

Accident Scenario	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		Accident Frequency (per year)
	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	
Fire on the loading dock	0.16	6.6×10^{-5}	3.0×10^{-3}	1.5×10^{-6}	0.32	1.6×10^{-4}	5.0×10^{-4}
Fire in a process cell	9.9×10^{-7}	4.0×10^{-10}	1.8×10^{-8}	9.1×10^{-12}	1.9×10^{-6}	9.5×10^{-10}	1.0×10^{-4}
Deflagration inside a glovebox	2.1×10^{-4}	8.2×10^{-8}	3.8×10^{-6}	1.9×10^{-9}	4.0×10^{-4}	2.0×10^{-7}	1.0×10^{-4}
Forklift breach of containment	3.5×10^{-10}	1.4×10^{-13}	6.4×10^{-12}	3.2×10^{-15}	6.7×10^{-10}	3.4×10^{-13}	4.5×10^{-5}
Nuclear criticality	3.9×10^{-4}	1.5×10^{-7}	6.5×10^{-6}	3.2×10^{-9}	6.6×10^{-5}	3.3×10^{-8}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	7.0×10^{-3}	2.8×10^{-6}	1.3×10^{-4}	6.4×10^{-8}	0.013	6.7×10^{-6}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	10.3	4.5×10^{-3}	0.19	9.4×10^{-5}	19.8	9.9×10^{-3}	1.0×10^{-7}
Beyond evaluation basis earthquake	5.1	2.1×10^{-3}	0.094	4.7×10^{-5}	9.9	4.9×10^{-3}	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.9.1–4 and M.5.3.9.1–5 and the MACCS computer code.

Table M.5.3.9.2–3. Plutonium Conversion Facility Accident Impacts at Idaho National Engineering Laboratory

	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	Accident Frequency (per year)
Fire on the loading dock	0.22	9.0×10^{-5}	1.9×10^{-3}	9.7×10^{-7}	4.2	2.1×10^{-3}	5.0×10^{-4}
Fire in a process cell	1.3×10^{-6}	5.4×10^{-10}	1.2×10^{-8}	5.8×10^{-12}	2.5×10^{-5}	1.3×10^{-8}	1.0×10^{-4}
Deflagration inside a glovebox	2.8×10^{-4}	1.1×10^{-7}	2.4×10^{-6}	1.2×10^{-9}	5.2×10^{-3}	2.6×10^{-6}	1.0×10^{-4}
Forklift breach of containment	4.8×10^{-10}	1.9×10^{-13}	4.1×10^{-12}	2.1×10^{-15}	8.9×10^{-9}	4.5×10^{-12}	4.5×10^{-5}
Nuclear criticality	5.0×10^{-4}	2.0×10^{-7}	3.9×10^{-6}	1.9×10^{-9}	8.5×10^{-4}	4.3×10^{-7}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	9.5×10^{-3}	3.8×10^{-6}	8.3×10^{-5}	4.1×10^{-8}	0.18	8.9×10^{-5}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	14.1	6.9×10^{-3}	0.12	6.1×10^{-5}	262	0.13	1.0×10^{-7}
Beyond evaluation basis earthquake	7.0	2.8×10^{-3}	0.061	3.0×10^{-5}	131	0.065	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.9.1–4 and M.5.3.9.1–5 and the MACCS computer code.

Table M.5.3.9.2–4. Plutonium Conversion Facility Accident Impacts at Pantex Plant

Accident Scenario	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		Accident Frequency (per year)
	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability of Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	
Fire on the loading dock	0.097	3.9×10^{-5}	0.031	1.5×10^{-5}	4.8	2.4×10^{-3}	5.0×10^{-4}
Fire in a process cell	5.8×10^{-7}	2.3×10^{-10}	1.9×10^{-7}	9.3×10^{-11}	2.8×10^{-5}	1.4×10^{-8}	1.0×10^{-4}
Deflagration inside a glovebox	1.2×10^{-4}	4.8×10^{-8}	3.9×10^{-5}	1.9×10^{-8}	5.9×10^{-3}	3.0×10^{-6}	1.4×10^{-8}
Forklift breach of containment	2.1×10^{-10}	8.2×10^{-14}	6.6×10^{-11}	3.3×10^{-14}	1.0×10^{-8}	5.1×10^{-12}	4.5×10^{-5}
Nuclear criticality	2.4×10^{-4}	9.7×10^{-8}	9.3×10^{-5}	4.6×10^{-8}	2.3×10^{-3}	1.1×10^{-6}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	4.1×10^{-3}	1.6×10^{-6}	1.3×10^{-3}	6.6×10^{-7}	0.20	1.0×10^{-4}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	6.1	2.5×10^{-3}	1.9	9.7×10^{-4}	297	0.15	1.0×10^{-7}
Beyond evaluation basis earthquake	3.0	1.2×10^{-3}	0.97	4.8×10^{-4}	149	0.074	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.9.1–4 and M.5.3.9.1–5 and the MACCS computer code.

Table M.5.3.9.2–5. Plutonium Conversion Facility Accident Impacts at Oak Ridge Reservation

	Worker at 772 m		Maximum Offsite Individual		Population to 80 km		
Accident Scenario	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (rem)**	**Probability of Cancer Fatality[a]**	**Dose (person-rem)**	**Number of Cancer Fatalities[b]**	**Accident Frequency (per year)**
Fire on the loading dock	0.28	1.1×10^{-4}	0.28	1.4×10^{-4}	52.3	0.026	5.0×10^{-4}
Fire in a process cell	1.7×10^{-6}	6.8×10^{-10}	1.7×10^{-6}	8.4×10^{-10}	3.1×10^{-4}	1.6×10^{-7}	1.0×10^{-4}
Deflagration inside a glovebox	3.5×10^{-4}	1.4×10^{-7}	3.5×10^{-4}	1.8×10^{-7}	0.065	3.3×10^{-5}	1.0×10^{-4}
Forklift breach of containment	6.0×10^{-10}	2.4×10^{-13}	6.0×10^{-10}	3.0×10^{-13}	1.1×10^{-7}	5.6×10^{-11}	4.5×10^{-5}
Nuclear criticality	5.9×10^{-4}	2.4×10^{-7}	5.9×10^{-4}	3.0×10^{-7}	0.035	1.8×10^{-5}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	0.012	4.8×10^{-6}	0.012	6.0×10^{-6}	2.2	1.1×10^{-3}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	17.6	7.9×10^{-3}	17.6	9.8×10^{-3}	3,270	1.63	1.0×10^{-7}
Beyond evaluation basis earthquake	8.8	3.5×10^{-3}	8.8	4.4×10^{-3}	1,630	0.82	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m or the site boundary (772 m for this facility at ORR), 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.9.1–4 and M.5.3.9.1–5 and the MACCS computer code.

Table M.5.3.9.2–6. Plutonium Conversion Facility Accident Impacts at Savannah River Site

Accident Scenario	Worker at 1,000 m		Maximum Offsite Individual		Population to 80 km		Accident Frequency (per year)
	Dose (rem)	Probability of Cancer Fatality[a]	Dose (rem)	Probability Cancer Fatality[a]	Dose (person-rem)	Number of Cancer Fatalities[b]	
Fire on the loading dock	0.16	6.3×10^{-5}	3.1×10^{-3}	1.5×10^{-6}	15.0	7.5×10^{-3}	5.0×10^{-4}
Fire in a process cell	9.4×10^{-7}	3.8×10^{-10}	1.9×10^{-8}	9.3×10^{-10}	9.0×10^{-5}	4.5×10^{-8}	1.0×10^{-4}
Deflagration inside a glovebox	2.0×10^{-4}	7.9×10^{-6}	3.9×10^{-6}	1.9×10^{-9}	0.019	9.4×10^{-6}	1.0×10^{-4}
Forklift breach of containment	3.3×10^{-10}	1.3×10^{-13}	6.6×10^{-12}	3.3×10^{-15}	3.2×10^{-8}	1.6×10^{-11}	4.5×10^{-5}
Nuclear criticality	3.5×10^{-4}	1.4×10^{-7}	5.7×10^{-6}	2.8×10^{-9}	4.6×10^{-3}	2.3×10^{-6}	1.0×10^{-7}
Beyond evaluation basis fire in a process cell	6.7×10^{-3}	2.7×10^{-6}	1.3×10^{-4}	6.6×10^{-8}	0.64	3.2×10^{-4}	1.0×10^{-7}
Oxyacetylene explosion in a process cell	9.9	4.4×10^{-3}	0.19	9.7×10^{-5}	936	0.47	1.0×10^{-7}
Beyond evaluation basis earthquake	4.9	2.1×10^{-3}	0.097	4.8×10^{-5}	468	0.23	1.0×10^{-7}
[Text deleted.]							

[a] Increased likelihood (or probability) of cancer fatality to a hypothetical individual (a single onsite worker at a distance of 1,000 m 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 the accident has occurred.

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

Note: All values are mean values.

Source: Calculated using the source terms in Tables M.5.3.9.1–4 and M.5.3.9.1–5 and the MACCS computer code.

M.5.3.10 Existing Commercial and Partially Completed Light Water Reactors

The use of MOX in place of Uranium Dioxide (UO_2) fuel in existing commercial and partially completed LWRs is an alternative being considered for the disposition of surplus Pu. The risks and consequences of potential accidents for commercial LWRs that use UO_2 fuel have been studied in detail and documented in reports submitted by nuclear plant owners to the NRC. These reports do not address the potential accident impacts associated with the use of MOX fuels.

The risks associated with existing commercial LWRs has been studied by the operating organizations in accordance with NRC guidance. The safety of these reactors has been analyzed and reported by plant operators in such documents as SARs and probabilistic risk assessments. In addition, the NRC has conducted probabilistic risk assessments on five existing reactors and issued the report *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants* (NUREG-1150). According to this report, the estimated mean core damage frequencies would range from 4.0×10^{-6} to 3.4×10^{-4} caused by internal events (not including natural phenomena). Should such core damage occur, the potential consequences would range from 71 to 1,022 latent cancer fatalities for the offsite population according to this NRC report. The estimated risks to the offsite population from these accidents would range from 9.6×10^{-4} to 2.4×10^{-2}.

Reactor safety issues regarding the use of MOX fuel in existing LWRs are addressed in a recent report by the National Academy of Science (NAS). The report indicates that the potential influences on the safety of the use of MOX fuel in LWRs has been extensively studied in the United States in the 1970s. These influences on safety have also been extensively studied in Europe, Japan, and Russia. Regarding effects of MOX fuel on accident probabilities, the report states,"... no important overall adverse impact of MOX fuel use on 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 that will involve factors not related to fuel composition and hence unaffected by the use of MOX fuel rather than LEU fuel" (NAS 1995a:352). Regarding the effects of MOX fuel on accident consequences, the report states, "... it seems unlikely that the switch from uranium-based fuel could worsen the consequences of a postulated (and very improbable) severe accident in an LWR by more than 10 to 20 percent. The influence on the consequence of less severe accidents, which probably dominate the spectrum value of population exposure per reactor-year of operation would be even smaller, because less severe accidents are unlikely to mobilize any significant quantity of plutonium at all" (NAS 1995a:355).

Analysis described has been performed using the MACCS code to determine the effects of the use of MOX fuel in an existing LWR. The assumed accident conditions include a large population distribution near the existing LWR and meteorology conditions for dispersal leading to large doses, and would not necessarily be reflective of actual site conditions. A sample of severe accident scenarios is illustrated in Table M.5.3.10–1. The data shown are derived from a range of severe accidents that make up the release scenarios. Some accidents have frequencies much less than 1×10^{-8} and large releases. These low frequency/high release accidents were included to reflect severe accident conditions leading to core damage and release of radioactive materials in order to obtain an estimate on the effects of using MOX fuel versus uranium fuel, as indicated by the ratios. To perform a comparison of existing commercial LWR impacts with other disposition alternatives, reactor accidents with frequencies less than 1×10^{-7}/yr would need to be done with site-specific meteorology, receptor, and population data.

Impacts are calculated in units of probability of cancer fatality for the MEI and the worker and the number of cancer fatalities for the offsite population. The fatality data shown does not reflect site conditions and would differ if site-specific meteorology and population were used. The ratios of accident fatalities for MOX and UO_2 fueled LWRs are given in Table M.5.3.10–2 only for the purpose of showing the relative impacts because the ratios would not be affected by meteorological or population data. Each scenario is based on releases taken from an existing commercial LWR probabilistic risk assessment of severe accidents. The releases were modeled using the MACCS code based on a large population distribution near a generic LWR and meteorological

Table M.5.3.10–1. Accident Impacts for Existing Light Water Reactor With Mixed Oxide Fuels

Accident Release Scenario[a]	Accident Scenario Frequency[b] (per year)	Worker at 1,000 m			Maximum Offsite Individual			Population Within 80 km		
		Probability of Latent Cancer Fatality	Risk of Latent Cancer Fatality (per year)	Risk of Latent Cancer Fatality per 17 yr (per 11 yr)	Probability of Latent Cancer Fatality	Risk of Latent Cancer Fatality (per year)	Risk of Latent Cancer Fatality per 17 yr (per 11 yr)	Number of Latent Cancer Fatalities	Risk of Latent Cancer Fatalities (per year)	Risk of Latent Cancer Fatalities[c] per 17 yr (per 11 yr)
Steam generator tube rupture	1.5×10^{-6}	1.0	1.5×10^{-6}	2.6×10^{-5} (1.7×10^{-5})	1.0	1.5×10^{-6}	2.6×10^{-5} (1.7×10^{-5})	5.9×10^{3}	0.0089	0.15 (0.098)
Large late containment failure-high RCS pressure	8.1×10^{-6}	0.79	6.4×10^{-6}	1.1×10^{-4} (7.2×10^{-5})	0.86	7.0×10^{-6}	1.2×10^{-4} (7.8×10^{-5})	1.3×10^{2}	0.0011	0.018 (0.012)
Large early containment failure-medium/low RCS pressure	7.0×10^{-7}	1.0	7.0×10^{-7}	1.2×10^{-5} (7.8×10^{-6})	1.0	7.0×10^{-7}	1.2×10^{-5} (7.8×10^{-6})	2.3×10^{3}	0.0016	0.027 (0.018)
Large early containment failure-high RCS pressure	8.7×10^{-7}	1.0	8.7×10^{-7}	1.5×10^{-5} (9.8×10^{-6})	1.0	8.7×10^{-7}	1.5×10^{-5} (9.8×10^{-6})	1.6×10^{3}	0.0014	0.024 (0.016)

Table M.5.3.10–1. Accident Impacts for Existing Light Water Reactor With Mixed Oxide Fuels—Continued

		Worker at 1,000 m			Maximum Offsite Individual			Population Within 80 km		
Accident Release Scenario[a]	Accident Scenario Frequency[b] (per year)	Probability of Latent Cancer Fatality	Risk of Latent Cancer Fatality (per year)	Risk of Latent Cancer Fatality per 17 yr (per 11 yr)	Probability of Latent Cancer Fatality	Risk of Latent Cancer Fatality (per year)	Risk of Latent Cancer Fatality per 17 yr (per 11 yr)	Number of Latent Cancer Fatalities	Risk of Latent Cancer Fatalities (per year)	Risk of Latent Cancer Fatalities[c] per 17 yr (per 11 yr)
Interfacing system loss of cooling accident	1.3×10^{-7}	1.0	1.3×10^{-7}	2.2×10^{-6} (1.4×10^{-6})	1.0	1.3×10^{-7}	2.2×10^{-6} (4.8×10^{-6})	7.3×10^{3}	0.00095	0.016 (0.010)
Small early containment failure	2.4×10^{-6}	1.0	2.4×10^{-6}	4.1×10^{-5} (2.7×10^{-5})	1.0	2.4×10^{-6}	4.1×10^{-5} (2.7×10^{-5})	1.2×10^{3}	0.0029	0.049 (0.032)

[a] Each release scenario is based on existing commercial LWR probabilistic risk assessment of severe accidents. The release scenarios were modeled using the MACCS code based on large population distribution near a generic site and meteorological conditions leading to large doses. The meteorological data used to estimate the fatalities are not for reactor sites, and therefore, the fatality data shown are not relevant to nor indicative of fatalities that would be calculated if site-specific meteorological and population data were used in the model.

[b] A release scenario typically will contain many accident sequences that have a common outcome (for example steam generator tube rupture). Each accident sequence has a frequency of occurrence that is derived from an event tree analysis and will include sequences with frequencies that are below those used in typical EISs.

[c] The population risk of latent cancer fatality, when compared with similar risk from the use of LEU reactor fuel, yields correct latent cancer fatality ratios of MOX-fuel relative to LEU-fuel. The accident conditions include a large population distribution near the existing LWR and meteorology conditions for dispersal leading to large doses and would not necessarily be reflective of actual site conditions. Further site-specific NEPA and safety documentation would be completed if the existing LWR alternative is selected.

Note: RCS=Reactor Coolant System. For the Preferred Alternative, for analysis purposes, approximately 70 percent of the Pu was assumed to be used for MOX fuel. The impacts projected for 50 t for the assumed 17-year existing LWR campaign would be proportionately reduced to those for an 11-year campaign; all values are mean values.

Source: HNUS 1996a.

Table M.5.3.10–2. Ratio of Accident Impacts for Mixed Oxide Fueled and Uranium Fueled Reactors for Typical Severe Accidents (Mixed Oxide Impacts/Uranium Impacts)[a]

Accident Release Scenarios[b]	Worker at 1,000 m	Maximum Offsite Individual	Population Within 80 km
Steam generator tube rupture	0.94	0.94	0.94
Large late containment failure-high RCS pressure	1.08	1.07	1.08
Large early containment failure-medium/ low RCS pressure	0.93	0.93	0.93
Large early containment failure-high RCS pressure	0.97	0.96	0.97
Interfacing system loss of cooling accident	0.93	0.92	0.93
Small early containment failure	0.96	0.95	0.96

[a] The ratio of accident fatalities for MOX-and UO_2-fueled LWRs are shown for the purposes of showing relative impacts. For example, 0.94 indicates that the impacts of the accident would be lower for a reactor with MOX fuel.

[b] Each release scenario is based on existing commercial LWR probabilistic risk assessment of severe accidents. The release scenarios were modeled using the MACCS code based on estimated population distribution near a generic site and meteorological conditions for dispersal leading to large doses. Therefore, the fatality data shown are not relevant to nor indicative of fatalities that would be calculated if site-specific meteorological and population data were used in the model.

Note: RCS=Reactor Coolant System.

Source: HNUS 1996a.

conditions for dispersal leading to large doses and would not necessarily be reflective of actual site conditions. Further site-specific NEPA and safety reviews would be performed should the Existing LWR Alternative be selected at the ROD.

The MACCS calculations for typical LWR reactors using a MOX core have shown that, for some risk measures, there is a reduction in risk. This must result from lower releases of some fission products, all else being equal. The main reasons for such lower releases are the following:

- The different spectrum of fission products resulting from Pu-239 fissions as compared to those from fissions of U-235.
- The lower flux level in a reactor with an increased loading of Pu-239 in place of U-235, both at the same power level.

The first effect will alter the amounts produced of individual radionuclides. The lower flux level will alter both the level at which saturating fission products will equilibrate and the rates of change of fission products with medium cross-sections.

Additional effects can result from different neutron flux spectrum shapes in MOX-fueled reactors, the effects of Pu-241 fission in MOX cores, and the change in U-238 fast fissions, as well as several other phenomena, but these are all of less importance.

Examination of typical core inventory isotopic ratios for MOX compared to UO_2 cores, reveals that the ratios are less than one for some risk-important radionuclides. In particular, these are the krypton, strontium, and tritium isotopes, and also cesium-134, iodine-134 and iodine-135. While there are others that are reduced, they all are not readily released to the environment (very small "release fraction") in a severe accident so that the change is not significant.

A series of sensitivity calculations has been performed to illustrate the effects of the changes is the isotopic core inventory ratios. These have been carried out using the MACCS code for one particular Pressurized Water Reactor reactor using as a basis the severe accident releases quoted in the Individual Plant Evaluation PRA submittal document to the NRC. Six risk calculations were performed: the base case (UO_2) for the Individual Plan and Evaluation submittal case, a MOX case using the isotopic factors, and four sensitivity cases in which some of the isotopic ratio factors were arbitrarily set to 1.0, while the remaining ratios were as for the MOX case. The sensitivity cases run were the following:

- Cs-134 ratio increased to 1.0 (from 0.65)
- All krypton (Kr) isotopes increased to 1.0 (from 0.57-0.62)
- All strontium (Sr) and yttrium (Y) isotopes increased to 1.0 (from 0.5-0.78)
- Iodine, I-134, and I-135 increased to 1.0 (from 0.96 and 0.92)

The results of these cases are shown in Table M.5.3.10–3. The column headed MOX is the percentage change in the risk factors in going to a MOX core from a UO_2 core.The columns headed Cs, Kr, Sr/Y, I-134/135 are the sensitivity cases listed immediately above, with the changes shown as percentage differences from the MOX case.

Table M.5.3.10–3. Accident Impact Sensitivity Analysis for Light Water Reactors Using Mixed Oxide Fuels

Risk Factor	MOX[a]	Cs -134[b]	Kr[b]	Sr/Y[b]	I-134/135[b]
Latent Cancer Fatalities to Regional Population					
Within 10 miles	-8%	11%	0%	0%	0%
Within 50 miles	4%	0%	0%	0%	0%
Within 500 miles	-6%	8%	0%	0%	0%
Population Dose					
Within 10 miles	-6%	9%	0%	0%	0%
Within 50 miles	4%	-1%	0%	0%	0%
Within 500 miles	-5%	8%	0%	0%	0%
Prompt Cancer Fatalities to Regional Population					
Within 10 miles	2%	1%	0%	0%	0%
Within 50 miles	2%	1%	0%	0%	0%
Within 500 miles	2%	1%	0%	0%	0%

[a] Values are the percent change in health impacts due to replacing UO_2 core with MOX core.
[b] Values are the percent change in health impacts due to changing the isotopes as described in the text.
Source: HNUS 1996a.

The MOX and Cs columns taken together show that the reduction in the latent fatalities and population exposures can be explained by the reduction in the amount of Cs-134 in the MOX core. For example, the MOX core substitution results in a 6-percent reduction (from the UO_2 base) in latent fatalities out to 800 km (500 mi). Arbitrarily increasing the Cs-134 core inventory back up to the UO_2 level results in an 8-percent increase in this risk factor (from the MOX base) back to the magnitude of the UO_2 case.

The other isotopes do not appear to have a significant effect. The strontium and tritium fission products have a very low release fraction in all of the accidents for this reactor. The two iodine isotopes would be expected to have the same effect as the cesium but do not appear to be significant as the relative change in isotopic ratio is small. The negative early fatality effect of the cesium, iodine and krypton isotope decreases appear to be outweighed by all the other fission products whose inventories are larger in the MOX core (relative to the UO_2 core).

It is not immediately obvious why, relative to the UO_2 case, the MOX latent fatalities decrease within both the 16-km (10-mi) and the 800-km (500-mi) radius but increase within the 80-km (50-mi) radius. These do appear to track with the cesium. It is postulated that this effect is due to the varying temporal relationships of the decay, the meteorology, the population distributions, the different paths, and other factors that have differing relative dose importances with distance and time.

These results are quantitatively applicable only to a typical set of accidents and environmental conditions. They should be qualitatively indicative for other reactor situations that are not too dissimilar.

M.5.3.11 Contribution of Americium, Curium, and Plutonium to Reactor Accident Consequences

The analysis of potential accidents and risks for an LWR that uses MOX fuel was performed using the MACCS computer program and the library of isotopes that was included with the program. The isotopes in the library are considered important contributors to accident risk and consequences by the program developers at Sandia National Laboratory. In addition to the analysis of potential reactor accident impacts discussed in this appendix, a study using the MACCS program has also been performed to provide information on the contribution to accident risks and consequences of radioisotopes associated with MOX fuel: (1) Americium (Am-241); (2) Curium (Cm-242 and Cm-244), and (3) Pu (Pu-238, Pu-239, Pu-240, and Pu-241). The study was based on several potential severe accidents for three operating commercial reactors. Using the MACCS program, the ratio of dose due only to Americium, Curium, and Pu to the total dose due to all isotopes in the MACCS library were estimated and are shown in Table M.5.3.11–1. The results indicate that the contribution of Americium, Curium, and Pu to accident risks and consequences in a MOX fueled LWR very low, less than 1 percent to 2 percent of the total. The only exception to this is the "large late containment failure high reactor coolant system" accident for which the contribution of Americium, Curium, and Pu is 10 percent of total.

Table M.5.3.11–1. Ratio of Accident Dose for Americium, Curium, and Plutonium in a Mixed Oxide Fuel Reactor to Total Dose for Uranium Fueled Reactor (Americium+Curium+Plutonium Dose/Total Dose)

	Offsite Population	Maximum Exposed Individual
Accident Release Category		
Steam generator tube rupture	1.7×10^{-4}	2.4×10^{-4}
Large late containment failure	1.0×10^{-1}	9.7×10^{-2}
Large early containment failure—low to medium RCS pressure	1.3×10^{-3}	1.8×10^{-3}
Large early containment failure—high RCS pressure	1.6×10^{-2}	2.1×10^{-2}
Interfacing system loss of cooling accident	2.1×10^{-3}	3.1×10^{-3}
Small early containment failure	1.4×10^{-2}	1.5×10^{-2}

Note: RCS=Reactor Coolant System.
Source: HNUS 1996a.

Appendix N
Multipurpose Reactor

A multipurpose reactor is a reactor that can produce tritium, use plutonium (Pu)-based fuel, and/or offset operating costs through revenues generated from the sale of electricity. In the past, Congress and commercial parties, including reactor vendors, have expressed interest in developing a multipurpose reactor that could both meet the nations tritium supply requirements and accommodate the disposition of surplus Pu. In the National Defense Authorization Act for Fiscal Year 1996, Congress directed that the Department of Energy (DOE) include a cost-benefit analysis in the *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic Environmental Impact Statement* (Storage and Disposition PEIS) of the multipurpose reactor using the Advanced Light Water Reactor (ALWR) and Modular Helium Reactor (MHR) technologies. DOE has also received comments on the Storage and Disposition PEIS concerning certain types of multipurpose reactors including the Fast Flux Test Facility (FFTF) at Hanford Site (Hanford), which would actually be a dual purpose reactor than a multipurpose reactor as characterized above.

The purpose of this appendix is to provide information on the costs and benefits of carrying out two separate projects for the performance of the tritium production and Pu disposition missions, versus the costs and benefits of carrying out one multipurpose project for both missions. Information in the appendix is not a proposal for action by the Department, nor is it an analysis of the multi-purpose reactor technology that would allow it to be selected in the Storage and Disposition PEIS Record of Decision (ROD). This information is presented in the interest of more fully informing the decision-maker and the public regarding discussions about the potential utility of multi-purpose reactors for overlapping or simultaneous Departmental missions. The cost-effectiveness of using FFTF is not known, beyond the common-sense proposition that using an existing facility could be an advantage. Furthermore, neither the multipurpose reactor nor the FFTF in a dual-purpose mode has been proposed or subjected an independent system analysis by the Department.

N.1 BACKGROUND

An evaluation of multipurpose reactors was performed as part of DOE's *Final Programmatic Environmental Impact Statement for Tritium Supply and Recycling* (TSR PEIS) (DOE/EIS-0161, October 1995). This evaluation was a part of the TSR PEIS because the multipurpose reactor could potentially offer the capability of producing tritium at a reduced cost compared to other tritium production options through the sharing of costs with the Pu disposition mission. The TSR PEIS evaluated the potential environmental impacts for multipurpose reactors for ALWR, Modular High Temperature Gas-Cooled Reactor (MHTGR), and existing commercial light water reactor (LWR) designs. Although the TSR PEIS also mentioned a MHR, an emerging design variation of the MHTGR, no detailed analyses of the MHR as a reasonable option for the tritium production mission were included.

All of these reactor types normally utilize uranium as fuel. Uranium fuel could continue to be used if one of these technologies were selected for tritium production; however, an all Pu fuel or a mixed oxide (MOX) fuel of Pu and uranium could also be used for a tritium production reactor. Both the LWR and gas cooled reactor technologies have been used for electrical power production but not for tritium production. While the technology exists in other countries for operating LWRs with MOX fuel, the gas cooled reactor has no MOX fuel operating experience.

The Department's TSR Record of Decision (ROD) was to pursue a dual track for tritium supply based on the two most promising tritium supply alternatives. This dual track consists of (1) initiating the purchase of an existing commercial reactor (operating or partially complete) or irradiation services with an option to purchase the reactor for conversion to a defense facility and (2) designing, building, and testing critical components of an accelerator system for tritium production. Within a 3-year period, DOE expects to select one of the tracks to serve as the primary source of tritium. The TSR ROD concluded that the use of LWRs, especially existing

commercial reactors, has the highest potential for delivering new tritium gas by the required start date of 2011. The MHTGR and MHR were judged to have a lower probability of meeting this date because they have the potential for major technical or regulatory delays. They were also judged to have a lower probability than the other alternatives considered of being able to operate with sufficient reliability, to meet the annual tritium production requirements. Furthermore, total life cycle costs and operation and maintenance costs for gas reactors were high, especially for the MHTGR.

The TSR PEIS also considered tritium production in the FFTF, an existing liquid metal cooled reactor located at Hanford near Richland, Washington. It was dismissed in the TSR ROD as a long-term tritium supply option because its remaining life is substantially less than the long-term mission requirements for tritium production. The FFTF is also limited in the amount of tritium that it could produce, which is only a percentage of the tritium requirement. It was therefore not considered reasonable to rely on operating the facility as a long-term tritium supply option. Nevertheless, the TSR ROD indicated that DOE would conduct further evaluations to determine whether the operation of the FFTF might be able to play any role in meeting future tritium requirements. Although the FFTF produces no electricity and like the light water and gas cooled reactors has never been used for tritium production, it has been used in experimental capacities. Therefore, its utility as a multipurpose reactor is included in the discussion below.

The sizes of the multipurpose reactors considered for tritium production use based upon meeting specific tritium supply requirements and not on the disposition of a prescribed quantity of Pu within a reasonable (25-year) time period. These reactors would have to be capable of producing tritium at a steady state production mode and at an increased production mode should the need arise.

Plutonium disposition using a single multipurpose reactor would take longer than the disposition goal of 25 years from project authorization. This disposition goal could only be achieved by using more reactors (or an additional disposition option such as immobilization). This would be required for tritium production alone. Furthermore, if new reactor(s) were provided, the Pu disposition mission would be completed before the end of their useful design life for the reactors, thereby resulting in unneeded plutonium disposition capacity after approximately 15 years of operation (assuming 50 metric tons [t] (55 short tons) [tons] of Pu). Thus, while plutonium disposition would be possible in a multipurpose reactor, the primary purpose of the reactor would be tritium production.

N.2 NEW MULTIPURPOSE REACTORS

A new ALWR or MHR multipurpose reactor could potentially offer savings to the Government by combining the tritium production and Pu disposition missions into a single reactor program. Although it may be technically feasible for a multipurpose reactor to perform these combined missions, additional development, demonstration, and testing would be required to address a number of technical issues. These technical issues relate primarily to fuel and tritium target development and demonstration, and to related reactor modification and licensing requirements. These issues would be expected to have greater impact on the MHR than on an ALWR because of its extensive developmental requirements and lack of operating experience. Also issues related to the different capacity requirements (number of reactors) for each mission, and different refueling and re-targeting schedules would need to be addressed. The use of a single reactor for tritium production is not compatible with the Pu disposition objective of satisfying the Spent Fuel Standard in 25 years or less from the time of project authorization. Furthermore, a lower reliability would be expected for combined missions versus separate missions in separate reactors, because the problems and delays with one mission could impact the timely implementation of the other. For example, problems with MOX fuel performance (a disposition mission factor) or tritium target performance such as repeated leaks (a tritium production mission factor) could delay both missions. Whether this potentially lower reliability would impact the effective implementation of both these critically important missions is still an open question.

The Nuclear Regulatory Commission (NRC) evaluated MOX-burning LWRs in an environmental impact statement issued in 1976.[1] They included extensive information on the changes and impacts required for an LWR in order to allow it to utilize MOX fuel. This document was reviewed as part of DOE's analysis of the multipurpose ALWR.

N.3 ADVANCED LIGHT WATER REACTOR

The TSR PEIS considered four ALWR options: a large 1300-megawatt electric (MWe) pressurized water reactor (PWR) plant, a large 1,100-MWe boiling water reactor (BWR) plant, a small 600-MWe PWR plant, and a small 600-MWe BWR plant. All ALWR options would use light (regular) water as the reactor coolant and moderator, and a steam cycle to remove heat from the light water coolant to generate electricity.

The ALWR concepts considered for tritium production are based on advanced commercial reactor designs developed by U.S. reactor vendors in conjunction with DOE. These commercial designs have been reviewed or are undergoing review by the NRC with the goal of simplifying their licensing process. They incorporate passive safety features and other advancements that would be expected in any new LWRs that might be built in the United States in the future. Advanced BWRs, similar to some of the designs being developed in the United States, are being built in Japan.

Although no ALWRs have been built in the United States, they are considered to have the lowest risk of any new reactor option evaluated for tritium production or Pu disposition because they are based on the light water technology and extensive operating experience of the 110 commercial reactors operating in this country today. A single multipurpose large ALWR (1,256 MWe) with a typical commercial reactor fuel cycle (18 to 24 months) could meet both the steady state and increased tritium production requirements. The steady state requirement could be achieved without having to displace any of the reactor's fuel rods with tritium target rods. However, the disposition of 50 t (55 tons) of surplus Pu in a single multipurpose reactor would take approximately 44 years from project authorization (12 years of plant design and construction and 32 years of operation). For the increased tritium production requirement, Pu disposition would take even longer since some of the reactor's fuel may have to be displaced by tritium targets to accommodate the higher tritium requirement. In order to produce the required amount of tritium and disposition 50 t (55 tons) of Pu in approximately 25 years from project authorization, two multipurpose ALWRs would be required. If a hybrid approach were chosen for Pu disposition in which reactors and immobilization were used, less than the full 50 t (55 tons) of Pu would be identified for disposition in reactors, and the disposition times and number of reactors would therefore be reduced.

N.4 MODULAR HELIUM REACTOR AND MODULAR HIGH TEMPERATURE GAS-COOLED REACTOR

The TSR PEIS considered the use of three 350-megawatt thermal (MWt) MHTGR modules for the production of tritium. The MHTGR concept uses a steam cycle to remove heat from the helium reactor coolant and to generate electricity. There has been some limited experience (with mixed success) in the United States and abroad with gas reactors using a steam cycle. However, the use of MOX fuel in MHTGR reactors designed to produce tritium would significantly reduce tritium production capability. Combining tritium production and Pu disposition in multipurpose gas reactors would require an increase to six 350-MWt MHTGR reactor modules in order to meet the increased tritium production requirement. However, this number of modules would still be insufficient to meet the 25 year Pu disposition objective. Approximately twenty-four 350-MWt reactor modules would be required to satisfy the Pu mission goal using the MHTGR technology. This number of reactors would be economically unattractive.

[1] *Final Environmental Statement on the Use of Recycled Plutonium Mixed Oxide Fuel in Light Water Cooled Reactors* (GESMO) (NUREG-0002), August 1976.

The MHR concept is economically more attractive than the MHTGR because it has the potential to achieve higher thermal efficiencies and its advanced design allows reactors to be packaged into larger modules (600-MWt) than the 350-MWt MHTGR. Fewer MHR modules would therefore be required to satisfy the same tritium or Pu mission, and the cost of reactor construction and operations would be less. Two MHR reactor modules would be required to meet the increased tritium production requirement alone, without Pu disposition. Eight multipurpose reactor modules using a 2-year reactor fuel cycle proposed by the MHR vendor, would be required to satisfy the increased tritium production requirement and disposition approximately 50 t (55 tons) of surplus Pu. With eight 600-MWt MHR reactor modules, the disposition of 50 t (55 tons) of surplus Pu would take approximately 26 years from project authorization (12 years of plant construction and 14 years of operation).

The MHR can achieve substantial cost savings when compared to the MHTGR because it uses the reactor's helium coolant in a high efficiency direct-cycle gas turbine to generate electricity rather than in a steam cycle like the MHTGR. While the direct cycle approach has cost advantages and higher plant efficiency, it also has a much larger risk associated with bringing it to a level of technical maturity that would allow the Government to proceed with confidence in the timeframe required to meet tritium production and Pu disposition objectives.

Since the direct-cycle turbine power conversion system is new and unproven, it must complete an extensive design, development, testing, and demonstration program. Further, the integration of the power conversion system with the reactor system has not been demonstrated, and will require extensive design and development. In addition to the direct cycle turbine, only very limited experience exists with Pu fuel for gas reactors. Both the fabrication process and operational performance of the fuel must be developed and demonstrated. The Pu fuel fabrication facility for the MHR would be a first-of-a-kind facility and therefore represents one of the highest risk items for maintaining the aggressive schedule that would be required for tritium production and Pu disposition. Direct cycle turbine and fuel development activities required for the MHR would therefore represent a major risk to meeting mission objectives in a timely, reliable, and cost-effective manner. It was for this reason that DOE did not consider the MHR to be a reasonable alternative for either tritium production or Pu disposition, or both missions combined in a multipurpose MHR reactor.

N.5 COST COMPARISONS

N.5.1 ADVANCED LIGHT WATER REACTOR COSTS

For purposes of cost analysis, reactors were assumed to be located at the Savannah River Site (SRS). Costs were calculated for separate tritium production and Pu disposition missions in separate reactors, as well as for combined missions in a multipurpose reactor.

The cost/benefit of the multipurpose ALWR reactor option was analyzed for two cases: Government ownership and private ownership. In the case of Government ownership, the front-end costs which include the reactor, tritium recovery facility, pit disassembly/conversion facility, MOX fuel fabrication facility, and all associated research and development, would all be paid for by the Government. In the case of private ownership, the reactor and MOX fuel fabrication plant would be financed, constructed, and operated by a private entity. All other front-end costs would be paid for by the Government. Cost comparisons, expressed in discounted 1996 dollars, are provided in Table N.5.1–1. Costs shown are for construction and operation of the reactor and common facilities (pit disassembly/conversion, MOX fuel fabrication, tritium target fabrication). An average revenue return of $.029 per kilowatt-hour is assumed for electricity production over the life of the reactor. This return reflects the potential effect of deregulation of the electric power industry. This value was also used in the *Reactor Alternative Summary Report Vol. 4—Evolutionary LWR Alternative* (ORNL/TM-13275/r4) and the *Technical Summary Report for Surplus Weapons—Usable Plutonium Disposition* (DOE/MD-0003).

The Government ownership cases highlight the potential cost advantages of a multipurpose reactor as compared to building new reactors for each mission separately. The front-end costs to the Government for deploying

Table N.5.1–1. Advanced Light Water Reactor Costs (1996 Dollars)

Mission	Reactor Size	Front-End Costs (Dollars in Billions)	Discounted Life Cycle Costs (Dollars in Billions)	Discounted Total Costs for Both Missions (Dollars in Billions)
Private[a]				
Pu disposition only	Two ALWRs 2,512 MWe	0.32	4.1	7.0[b]
Tritium production only	One ALWR 1,100 MWe	0.34	2.9	
Multipurpose	Two ALWRs 2,512 MWe	0.66	4.4	4.4
Government[c]				
Pu disposition only[d]	Two ALWRs 2,512 MWe	6.9	3.7	7.0[b]
Tritium production only	One ALWR 1,100 MWe	4.3	2.0	
Multipurpose	Two ALWRs 2,512 MWe	7.2	3.7	3.7

[a] Privately owned MOX fuel fabrication facilities and reactors. Government owned Pu processing and tritium facilities.
[b] Sum of Pu disposition and tritium production.
[c] All government owned facilities.
[d] Same case as Table 4.2 of October 1996 Technical Summary Report for Surplus Weapons Usable Plutonium Disposition (Rev 1).

separate reactors and associated facilities versus the costs for two multipurpose reactors and associated facilities are $11.2 billion and $7.2 billion, respectively. The front-end cost for the multipurpose reactor is therefore $4.0 billion less than that for separate new reactors. The discounted life-cycle costs are $2.6 billion less for the two multipurpose reactors compared to the three reactor separate projects case.

The private ownership option assumes a combination of industry-based financing for a substantial part of the undertaking, together with debt financing for the balance. However, the potential feasibility of private financing must be tempered with the recognition that for more than 15 years, the commercial nuclear power industry has not found it economically and politically feasible to build new nuclear power plants in the United States. It is unlikely that passive investors would be willing to "project finance" the construction phase of the project (that is, where the debt and equity investors take the project risk), for the following reasons: the size of the investment; the risks associated with completing the construction of such a facility; and the general anxiety within the investment community over nuclear projects. There is some possibility that passive investors might be willing to finance such a project once it is built and operating; after NRC approvals and associated risks are complete; and after technology issues, construction delays, cost overruns, permitting, and initial startup testing are behind the project. Government support of a privately owned reactor may overcome some of these concerns, as suggested by the interest expressed by some industry groups in both single purpose and multipurpose reactors for tritium production and Pu disposition.

The private ownership cases highlight the potential cost advantages of private ownership. The front-end costs to the Government are significantly reduced for the reactors in the Pu disposition only case ($6.6 billion reduction), the tritium production only case ($4.0 billion reduction), and the multipurpose reactor case ($6.6 billion reduction), when compared to Government ownership because the front-end costs of the reactor and MOX fuel fabrication facility would be financed by a private entity. For the multipurpose reactor, the discounted life-cycle cost is increased, however, by about $0.7 billion as a result of the higher financing costs for a private owner combined with tax effects and the desire by the private investor for a return on investment. These costs are

ultimately paid for by the Government as part of the irradiation service over the 40-year operating life of the reactor. Potential advantages of private ownership would be lower front-end costs and the fact that Government outlays could be spread over the life of the project.

The Department tasked a consultant (Putnam, Hayes & Bartlett, Inc. [PHB]) to provide an independent cost analysis of DOE's tritium production alternatives, including a preliminary assessment of the multipurpose ALWR option. For the multipurpose reactor option, the PHB report[2] analyzed the costs of a Government-owned multipurpose reactor and a privately financed and owned option for three cases: the purchase of an existing commercial LWR; the construction and operation of a new small ALWR; and the construction and operation of a new large ALWR.

The PHB analysis of cost impacts of privately financing the multipurpose reactor indicates that only the purchase of an existing reactor would result in a cost benefit to the Government. The PHB report shows that privatization of such a multipurpose reactor could provide savings to the Government of about $1 billion in discounted life-cycle cost compared to the cost of separate reactors for tritium production and Pu disposition. A privately financed and owned multipurpose reactor could cost the Government an additional $0.7 billion to $11.4 billion in discounted cost compared to the Government ownership option as a result of the higher financing cost for a private owner and the desire by the private investor for a return on investment. This result depends heavily on electricity unit revenues and the terms that private participants are willing to offer DOE. The PHB report concluded that private financing may provide benefits to the Government, but should be examined by DOE after a tritium supply option or strategy is chosen.

N.5.2 MODULAR HELIUM REACTOR COSTS

For purposes of MHR cost analysis, reactors were again assumed to be located at Savannah River Site. As in the case of the ALWR, costs were calculated for separate tritium production and Pu disposition missions in separate reactors, as well as for combined missions in a multipurpose MHR. The cost/benefit of the multipurpose MHR option was analyzed for two cases: Government ownership and private ownership. In the case of Government ownership, the front-end costs, which include the reactor, tritium recovery facility, pit disassembly/conversion facility, MOX fuel fabrication facility, and all associated research and development, would all be paid for by the Government. In the case of private ownership, the reactor and MOX fuel fabrication facility would be financed, constructed, and operated by a private entity. All other front-end costs would be paid for by the Government. Cost comparisons, expressed in discounted 1996 dollars, are provided in Table N.5.2–1. Costs shown are for construction and operation of the reactor and common facilities (pit disassembly/conversion, MOX fuel fabrication, tritium target fabrication). An average revenue return of $.029 per kilowatt-hour is assumed for electricity production over the life of the reactor.

For the Government ownership option, the Government's front-end costs for the multipurpose MHR plant and the sum of the costs for separate reactor plants for tritium production and Pu disposition, were calculated to be about $6.5 billion and $8.1 billion, respectively. The potential cost advantage of the multipurpose MHR would be about $1.6 billion in front-end costs.

With regard to private financing, the MHR would not be feasible until after a period of successful operation of a demonstration facility. Financing for a multipurpose MHR would be difficult in view of the poor operating history of the Fort Saint Vrain facility, a gas-cooled reactor plant in Colorado that experienced a multitude of operational problems. The revolutionary nature of its design, the development and demonstration requirements associated with the reactor fuel and direct cycle turbine, and the lack of NRC review would be further impediments to private financing.

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[2] DOE Tritium Production Options: Putnam, Hayes & Bartlett Final Report On Cost Analysis, September 1, 1995 (Text Revision October 15, 1995).

Table N.5.2–1. Modular Helium Reactor Costs (1996 Dollars)

Mission	Reactor Size[a]	Front-End Costs (Dollars in Billions)	Discounted Life Cycle Costs (Dollars in Billions)	Discounted Total Costs for Both Missions (Dollars in Billions)
Private[b]				
Pu disposition only	8 modules 2,288 MWe	0.32	3.5	5.0[c]
Tritium production only	2 modules 572 MWe	0.68	1.5	
Multipurpose	8 modules 2,288 MWe	1.0	4.1	4.1
Government[d]				
Pu disposition only	8 modules 2,288 MWe	5.8	3.1	5.0[c]
Tritium production only	2 modules 572 MWe	2.3	1.0	
Multipurpose	8 modules 2,288 MWe	6.5	3.7	3.7

[a] Each reactor module is designed to produce 286 MWe.
[b] Privately owned MOX fuel fabrication facilities and reactors. Government owned Pu Processing and tritium facilities.
[c] Sum of Pu disposition and tritium production.
[d] All government facilities.

For the privately owned multipurpose MHR, the front-end costs could be reduced by about $5.5 billion relative to the Government-owned multipurpose reactor option because the reactors and the fuel fabrication plant would be financed, constructed, and operated by private entities. The privately owned multipurpose reactor would cost the Government about $0.4 billion more in discounted life cycle costs compared to the Government-owned option. This increase, as in the case of the ALWR, is the result of higher financing costs for a private owner combined with tax effects and the desire of the private investor for a return on investment. These costs would ultimately be paid for by the Government as part of the irradiation service over the operating life of the reactor. Again, potential advantages of private ownership would be lower front-end costs and the fact that the Government outlays could be spread over the life of the project.

N.6 FAST FLUX TEST FACILITY

As part of the process of selecting Pu disposition technologies for evaluation in the Storage and Disposition PEIS, DOE considered the FFTF, a liquid metal reactor at Hanford, because it was an existing facility that would not require the large commitment of time and money that a new reactor would require for implementation of the Pu disposition mission. The FFTF, however, was eliminated because it was in a standby status awaiting shutdown and because it could not satisfy the Storage and Disposition PEIS criterion of completing the disposition mission within 25 years using the historic FFTF Pu fuel enrichment specifications.

It has been suggested by commentors in this PEIS and others that the use of the FFTF as an integral part of the nation's tritium production infrastructure might help ensure the tritium supply by 2005 and at the same time provide a way to begin the disposition of surplus weapons-usable Pu as well as provide a source of medical isotopes.

As noted in the DOE's December 5, 1995 ROD on Tritium Supply and Recycling, DOE is evaluating the operation of the FFTF to determine if it might have a role in meeting future tritium requirements. If DOE

proposes and subsequently decides to use the FFTF for tritium production, then, in order to accomplish this mission, a portion of the fuel-usable surplus Pu could be used to operate the FFTF.[3] Before the FFTF could begin to use this Pu for the production of tritium, 3 to 4 years would be required to develop and test a higher Pu enriched reactor fuel and to establish a MOX fuel fabrication capability. Under these conditions, it would take at least 35 years from a ROD to disposition the surplus weapons-grade Pu that is suitable for use in reactors, or a supplementary disposition approach (such as immobilization) would be needed for the unused balance.

At the time this Storage and Disposition PEIS went to print, DOE had not proposed to use FFTF for tritium production. If DOE proposes to consider the FFTF in detail for this purpose, appropriate *National Environmental Policy Act* review will be performed.

N.7 SUMMARY

A single multipurpose ALWR could perform the tritium mission at a steady state mode but it would not provide the capability to disposition surplus Pu in a timely manner. For a single new large multipurpose ALWR operating on a typical commercial fuel cycle, it would take approximately 12 years of design and construction and 32 years of operation (a total of 44 years) to disposition 50 t (55 tons) of Pu. The same reactor could meet an increased tritium production requirement, but Pu disposition would take longer since some of the reactor's fuel may have to be displaced by tritium targets to accommodate the higher production requirement. Two multipurpose ALWRs would be required to meet the increased tritium production requirement and complete Pu disposition within the goal of 25 years from project authorization.

Two MHR reactor modules would be required to meet the increased tritium production requirement alone, without Pu disposition. Eight MHR modules would be required to meet the same tritium production requirement and complete the Pu disposition mission in 26 years (12 years for construction and 14 years for operation).

The use of the FFTF, or new multipurpose reactors sized for tritium production alone, could provide cost advantages to the Government in front-end costs and overall life cycle costs relative to the use of separate new reactor facilities for tritium production and Pu disposition. However, these cost savings would need to be balanced against the impacts of the slower Pu disposition rate imposed by the limited reactor capacity of these options. The goal of completing the disposition mission within 25 years of project authorization could only be achieved if additional reactors were to be provided or other means of disposition employed, such as immobilization. In the case of multiple new reactors, the disposition mission would be completed before the end of the useful design life of the reactors, which would result in unneeded reactor capacity.

[3] The rate of utilization would be about one ton per year maximum. See Use of the Fast Flux Test Facility for Tritium Production (the "JASON Report"), S. Drell and D. Hoummer, Co-chairs, The MITRE Corporation, McLean, Virginia, September 11, 1996, page 22 (draft).

Appendix O
Can-in-Canister Variants

O.1 INTRODUCTION

This appendix presents descriptive information on variants to the vitrification and ceramic immobilization disposition alternatives described as "a can-in-canister approach at Savannah River Site (SRS)" in Table 2.4-1. Based upon comments from the public on the Draft Programmatic Environmental Impact Statement, there is substantial interest in the can-in-canister (CIC) concept for the disposition of surplus plutonium (Pu), and several requests have been made for the Department of Energy (DOE) to consider this concept in its decisionmaking process.

During the initial 12 scoping meetings held across the country from August to October 1994, there was public input on the potential reasonable alternatives and proposed screening process. One of the 37 disposition options under consideration at that time was Option I-2: Borosilicate Glass Immobilization using a Modified Defense Waste Processing Facility (DWPF). This option was eliminated as unreasonable because the DWPF was not designed for criticality control, and therefore would require extensive modifications and refitting of the facility and equipment during its present mission, potentially resulting in increased personnel radiation exposure, along with potential delays and cost escalation for its present mission (DOE 1995m:1-9).

The Department recognized that there could be potential cost savings for the immobilization category of disposition alternatives if the output product from the DWPF (that is, vitrified high-level waste [HLW]) could be used in the disposal of Pu. In response, SRS and the national laboratories developed a CIC concept, which was mentioned in the *Summary Report of the Screening Process* as a variant of Option I-3: Immobilization in Borosilicate Glass (DOE 1995m:4-6). This concept currently consists of two CIC variants that are included in the *Technical Summary Report for Surplus Weapons-Usable Plutonium Disposition* (DOE/MD–0003) dated July 17, 1996.

The CIC concept at SRS is described as an example of a technology variant at an existing facility. However, the Record of Decision (ROD) for this PEIS will only select broader technology strategies for disposition. Site specific decisions and variant-specific decisions will be made pursuant to subsequent *National Environmental Policy Act* (NEPA) review tiered from this initial PEIS.

O.2 CAN-IN-CANISTER CONCEPT AT SAVANNAH RIVER SITE

The CIC concept includes variations to the two Pu disposition alternatives for vitrification and ceramic immobilization. CIC could utilize existing facilities at SRS to house the processes for pit disassembly/conversion, Pu conversion, and vitrification or ceramic immobilization. These existing facilities include the 221-F Facility in F-Area where Pu would be immobilized into a ceramic or glass form and loaded in a can and the DWPF in S-Area where the immobilized Pu would be loaded into a canister with vitrified HLW. For vitrification CIC, Pu would be immobilized in a borosilicate glass matrix in small cans and the cans placed in stainless steel canisters, which are then filled with molten borosilicate glass containing HLW to serve as the radiation barrier. For ceramic CIC, Pu would be immobilized in a ceramic matrix in small cans in lieu of the borosilicate glass, and the cans placed in stainless steel canisters, which are then filled with molten borosilicate glass containing HLW to serve as the radiation barrier. In both cases, the stainless steel canisters would be filled at the DWPF and placed in interim onsite storage at SRS until shipment to a HLW repository is possible.

The CIC concept at SRS could offer the following advantages over the base case immobilization category alternatives:

- Maximize use of existing SRS facilities
- Use vitrified HLW already slated for disposal in a HLW repository
- Provide a simple, yet effective, means to control criticality (that is, small cans of Pu)
- Eliminate unnecessary packaging between front-end operations and immobilization operations
- Require fewer additional canisters in a HLW repository
- Potentially reduce cost and worker radiation exposure

O.3 VITRIFICATION CAN-IN-CANISTER VARIANT

The vitrification CIC variant would process Pu forms to oxide, immobilize Pu oxide in borosilicate glass, and fill individual stainless steel cans. The filled cans would be loaded onto a frame and placed inside an empty stainless steel canister. The canister would have a cylindrical shape with a diameter of 0.6 meters (m) (2 feet [ft]) and length of 3 m (10 ft), and be identical to the stainless steel canister currently used in the DWPF to hold vitrified HLW, with the exception that the canister head would not be welded to the body until after the canister is loaded with cans of Pu glass. The loaded canister would be transferred to the DWPF facility, where molten HLW glass would be poured inside the canister and around the small cans and allowed to harden. The filled canisters would then be decontaminated, welded closed, and stored on-site in the Glass Waste Storage Building until a HLW repository is available for final disposal. A process schematic and material flow diagram for the vitrification CIC process are presented in Figures O.3–1 and O.3–2, respectively.

Facility Description. The vitrification CIC variant could use part of the existing 221-F Canyon building including the Pu Storage Facility and the New Special Recovery facilities, and part of the DWPF including the Vitrification and Service Buildings and the Glass Waste Storage Building. Table O.3–1 lists the location of each process area. Pit disassembly/conversion and other front-end processing, pretreatment operations, and first stage vitrification could be performed in the existing 221-F facility in areas specifically modified to vitrify Pu. The current F-Area facilities are designed and built to handle large quantities of Pu and have systems to maintain criticality control and safeguard systems to maintain accountability and security. The F-Area site layout is shown in Figure O.3–3. The floor area required for the front-end Pu processing and the vitrification functions

Table O.3–1. Locations for Proposed Vitrification Can-in-Canister Process Equipment

Process	Location
Receiving, shipping, storage, sampling	221-F Pu Storage Facility
Pit disassembly, dehydrate/hydride/oxidation	221-F New Special Recovery Facilities
Oralloy decontamination	221-F New Special Recovery Facilities
Special recovery	221-F Canyon 3rd Level
Fuel decladding, halide material processing	221-F Canyon 3rd Level
Feed preparation (dry variants)	221-F Canyon 3rd Level
Oxide lag storage (dry variants)	221-F Canyon 3rd Level
1st stage melter (dry variants)	221-F Canyon 3rd Level
Off-gas treatment (dry variants)	221-F Canyon 3rd Level
Feed preparation (wet variants)	221-F Canyon 3rd Level
Oxide lag storage (wet variants)	221-F Canyon 3rd Level

Table O.3–1. Locations for Proposed Vitrification Can-in-Canister Process Equipment—Continued

Process	Location
1st stage melter (wet variants)	221-F Canyon 2nd Level
Off-gas treatment (wet variants)	221-F Canyon 2nd Level
Can decon (dry variants)	221-F Canyon 3rd Level
Can decon (wet variants)	221-F Canyon 2nd Level
Can weld and test (dry variants)	221-F Canyon 3rd Level
Can weld and test (wet variants)	221-F Canyon 2nd Level
Interim can storage	221-F Canyon 3rd Level
Place in canister	221-F Canyon 1st Level
Weld and test	221-F Canyon 1st Level
Interim canister storage	221-F Canyon 1st Level/DWPF Service Building Interim Vault
Blend tank	DWPF Vitrification Building Hot Cell
2nd stage melter	DWPF Vitrification Building Hot Cell
Canister decontamination	DWPF Vitrification Building Hot Cell
Weld and test	DWPF Vitrification Building Hot Cell
Off-gas treatment	DWPF Vitrification Building Hot Cell
Interim product storage	DWPF Glass Waste Storage Building

Source: LLNL 1996j.

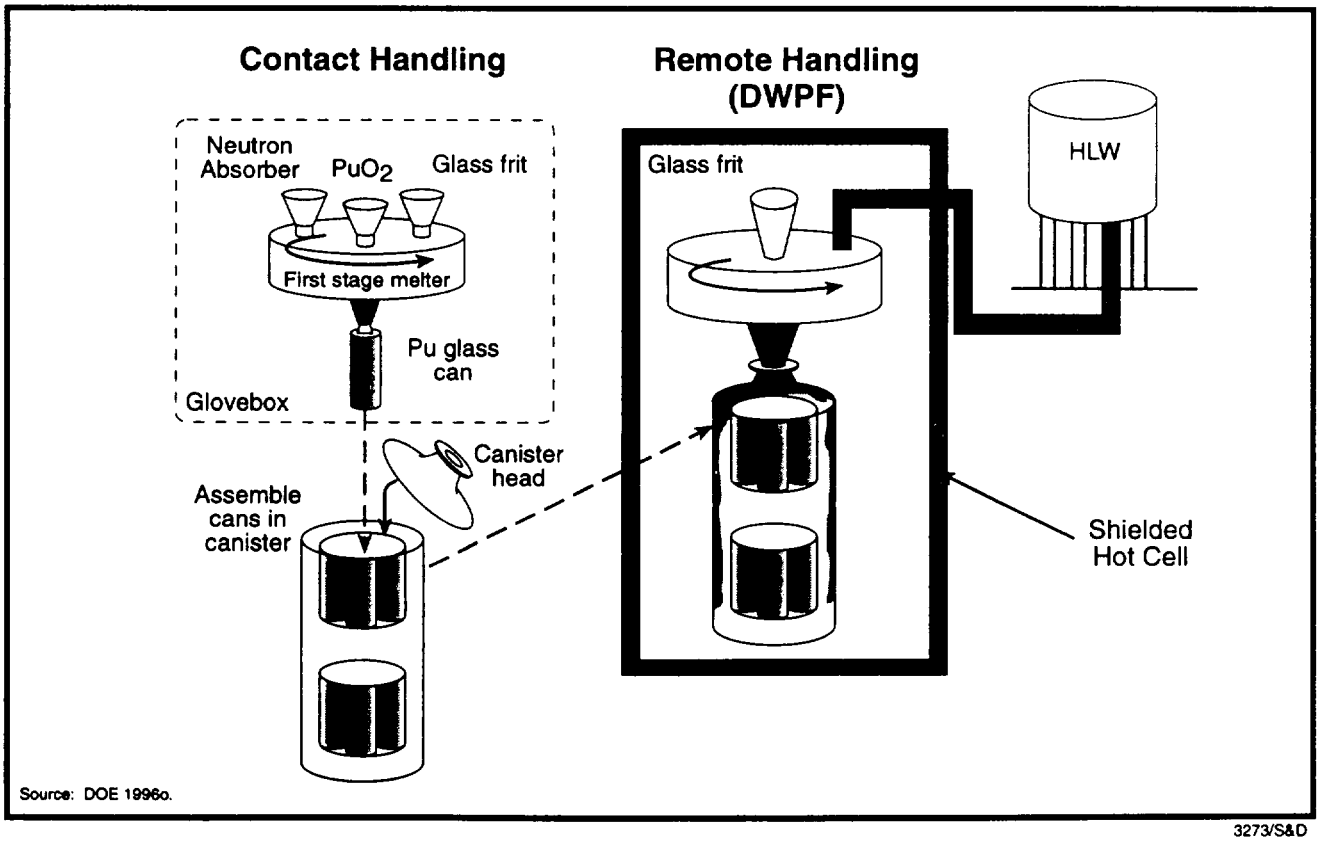
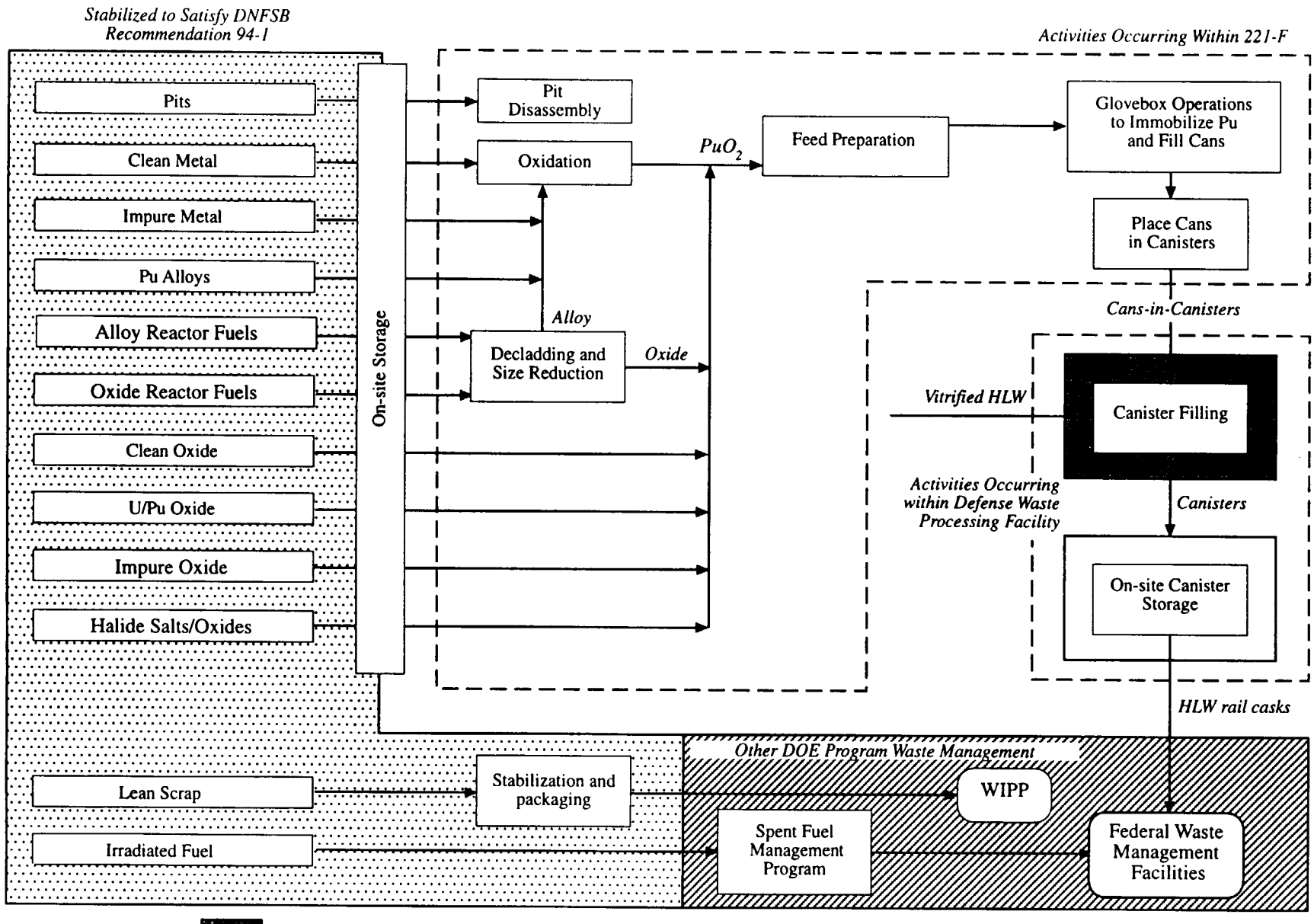


Figure O.3–1. Vitrification Can-in-Canister Variant Process.



Source: LLNL 1996j.

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Figure O.3–2. Can-in-Canister Material Flow Diagram for Operations at Savannah River Site.

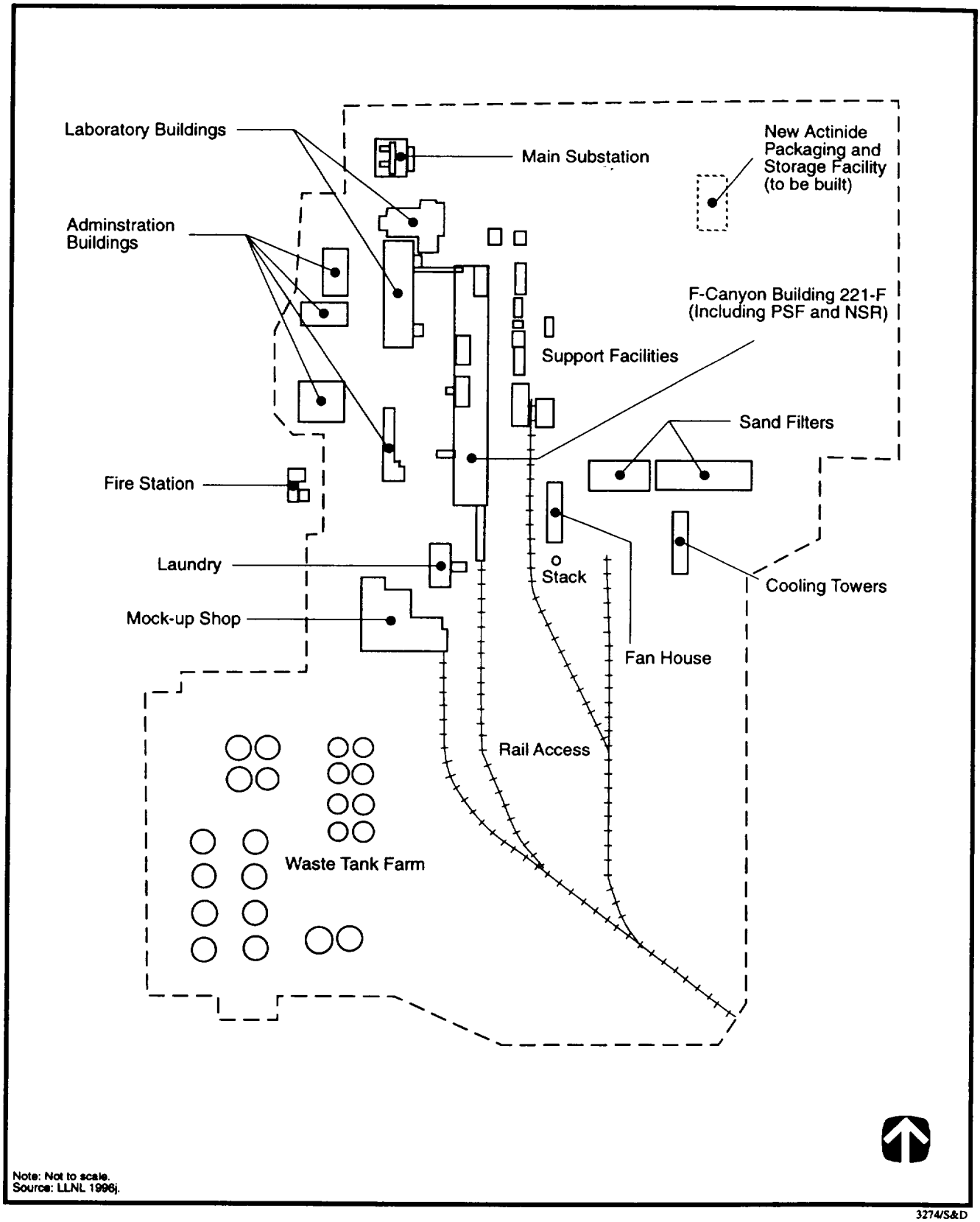

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Figure O.3–3. Savannah River Site F-Area Layout Showing Building 221-F.

would be approximately 2,080 square meters (m^2) (22,390 square feet [ft^2]). Total support area (heating, ventilation, and air conditioning, ingress, egress) would be an additional 6,500 m^2 (70,000 ft^2), approximately. The addition of the molten HLW as a radiation barrier in the canister would take place in the DWPF located in the S-Area. The S-Area site layout is shown in Figure O.3–4. The existing DWPF would be upgraded to meet safeguards and security requirements for the handling of stainless steel canisters containing Pu-glass cans during the canister filling operations. Disruptions to current DWPF operations would be minimized since Pu would not be introduced into the DWPF in other than a vitrified, criticality safe manner. Safeguards and security provisions would be upgraded in selected portions of the DWPF buildings.

Facility Operations. In the vitrification CIC variant, the initial step in the immobilization process would be the transportation by safe secure trailer (SST) of Pu feed materials (such as pits, metal, oxides, and unirradiated reactor fuels) from storage site(s) to the receiving facilities in the Pu Storage Facility and New Special Recovery facilities on top of the 221-F Canyon building at SRS. The shipping containers would be unpacked, and accountability measurements conducted. Pu pits would be disassembled and converted to oxide.[1] Other forms would go through the minimum necessary processing to be converted to oxides. The oxide feed materials would then enter into specially modified portions of the 221-F Canyon and undergo first stage immobilization in glove boxes as depicted in Figure O.3–2.

The feeds to this process would consist of glass formers, a neutron absorber, and the Pu oxide which would be combined in a melter to prepare a homogeneous borosilicate glass. This glass would be poured into small stainless steel cans with a Pu concentration of approximately 10 percent by weight. The outside dimensions of the can are a diameter of 13.4 centimeters (cm) (5.3 inches [in]), a length of 47.6 cm (18.7 in), and a shell thickness of 1 cm (0.4 in). After filling, the cans would then be capped, decontaminated, welded, tested, and transferred to lag storage in an onsite secured vault.

When Pu-glass cans are ready for processing, they will be decontaminated and transported to the canister loading area of the F-Canyon. The cans would be placed in a holding rack inside an open DWPF canister (head removed). After loading a canister with approximately 20 Pu-glass cans (5 cans in an array–4 arrays high), the head of the canister would be welded to the body and tested. Each canister would contain approximately 50 kilograms (kg) (110 pounds [lbs]) of Pu. The canisters may be placed in temporary storage in 221-F until they are shipped by rail or truck to the DWPF Service Building. A small storage vault, designed for special nuclear material and sized for about one week supply of canisters, would be provided in or adjacent to the DWPF Service Building to permit interim storage, if necessary, of canisters awaiting processing.

When a canister is ready to be filled, it would be transferred from the DWPF Service Building, via a controlled corridor, to the Melt Cell in the Vitrification Building. Utilizing a melt pour turntable, the HLW-borosilicate glass would be poured into the canister and around the Pu-glass cans. The cans would then become encapsulated in the HLW-glass within the stainless-steel canister. After filling, the canister would be decontaminated and sealed by plug welding and transported to the Glass Waste Storage Facility for interim onsite storage.

The Glass Waste Storage Building, Unit 1, was designed and constructed to hold HLW-glass canisters until a HLW repository is available. Since the capacity of Unit 1 is 2,286 canisters, a Unit 2 building is planned to be constructed as Unit 1 fills with HLW-glass canisters. Since Unit 2 is scheduled to be built in support of the DWPF mission, only upgrades to facility safeguards and security and increased capacity are necessary to accommodate additional canisters resulting from the Pu disposition mission. Alternatively, Unit 1 could be upgraded in the event Unit 2 was determined to be unnecessary. Because the Pu-glass cans displace volume that would normally contain HLW-glass in a separate DWPF operation, additional DWPF canisters would be needed to process all of the HLW in the SRS tank farm and all of the surplus Pu under scope of this PEIS. The number of additional DWPF canisters would be directly proportional to Pu loading in the Pu-glass. The total number of DWPF canisters containing Pu-glass cans would be about 1,000. Assuming 20 cans per canister, the volume of

[1] Based upon the Preferred Alternative, Pu pits would be converted to MOX fuel and would not be immobilized.

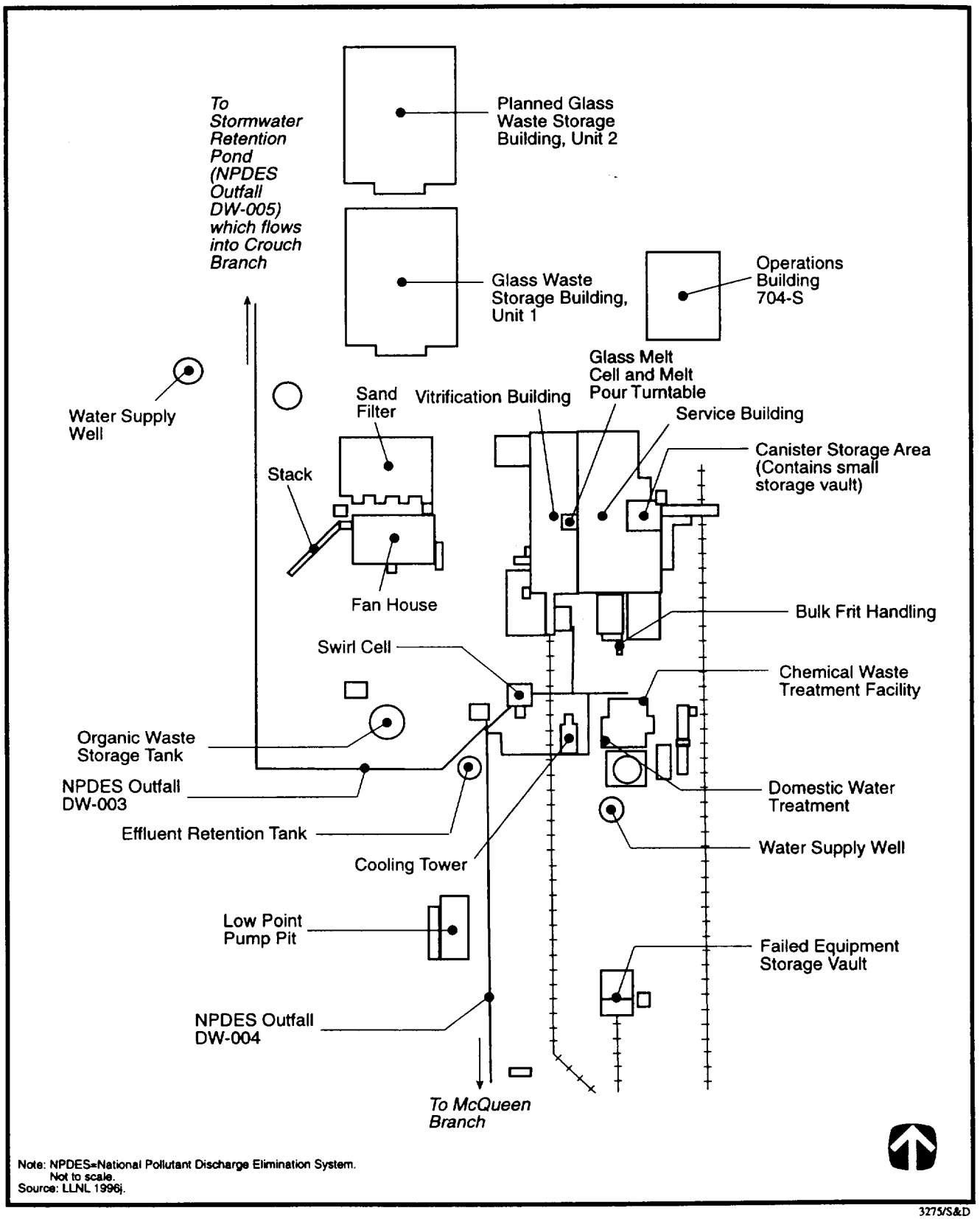

Figure O.3–4. Savannah River Site S-Area Layout Showing Defense Waste Processing Facility.

HLW-glass displaced would be approximately 20 percent, or about 200 additional canisters to those required for the SRS tank farm HLW program.

This variant would process 5,000 kg (11,000 lb) of surplus Pu annually. A normal operating year for the facility would be 200 days with a nominal throughput of 25 kg (55 lb) of Pu per operating day. This operating schedule assumes three shifts per day, 7 days per week. Remote maintenance, accountability, criticality control, and other functions would be performed during the 165 days per year the plant would not be expected to operate.

Construction. The vitrification CIC variant utilizes existing SRS facilities to house the pit disassembly/conversion, Pu conversion, and immobilization operations. For this variant, no major new construction would be required at SRS. However, some of the existing facilities would be modified and upgraded. Facilities in F-Area are designed and built to handle large quantities of Pu and have systems in place to maintain criticality control and safeguards systems to maintain accountability and security. However, the DWPF Vitrification Building, selected portions of the Service Building, and the Glass Waste Storage Building, Unit 1 or 2, would have to be upgraded to meet safeguards and security requirements to support storage and handling of vitrified Pu in accordance with DOE O 470.1 *Safeguards and Security Program.*

Radiological. With the CIC concept, the radiation source used to satisfy the spent fuel standard is the vitrified HLW outside the can. This eliminates the need for introducing radioactive cesium-137 (Cs-137) into the immobilization process, thereby reducing radiation shielding/hot cell requirements and the potential radiation exposures to operating personnel and the public. The dose to workers and the public would be smaller for the CIC variant than the dose to workers and the public for the new facility analyzed in Section 4.3.4.1.9.

Waste Management. Since the Pu conversion and immobilization processes would be similar to those otherwise described for the vitrification alternative, implementation of the vitrification CIC variant would result in similar waste streams. The wastes generated as a result of the operation of this variant would consist of transuranic (TRU), mixed TRU, low-level, mixed low-level, hazardous, nonhazardous (sanitary), and nonhazardous (other) materials. This variant could offer the potential for significant reductions in the quantity of pollutants emitted and wastes generated compared to the construction and operation of the separate disposition facilities as detailed in this PEIS. Significant emphasis would be placed on the minimization of both liquid and solid wastes. Vitrified HLW would be used to surround the cans and fill the canisters; however, no HLW would be generated as a result of the Pu disposition operations. In addition, any criteria pollutants, hazardous air pollutants, and other toxic compounds and gases that may be emitted as a result of the process would be within permit requirements.

Transportation. Fissile material located at various DOE facilities would be transported by SST, in compliance with applicable regulatory requirements, to SRS and placed in onsite temporary storage. Intrasite transport of radiological materials that are not immobilized would be limited to the secure movements within the 221-F Canyon building. Canisters containing the vitrified Pu in cans would be transported from 221-F to the DWPF Service Building via rail or truck. The filled canisters would be stored in the Glass Waste Storage Building until shipment to a HLW repository is possible.

The Department is developing a rail shipping cask for DWPF canisters. This rail shipping cask would hold five DWPF canisters. Based on the use of this transport cask, it is estimated that over the life of the project 40 additional shipments from SRS to a HLW repository would be required.

O.4 CERAMIC CAN-IN-CANISTER VARIANT

The ceramic CIC variant would be similar to the vitrification CIC variant in that both could use existing SRS facilities, produce cans of immobilized Pu, and fill DWPF canisters with Pu and vitrified HLW. The major difference between the two variants is that the Pu inside the stainless steel can would be immobilized in a titanate-based ceramic matrix, rather than a glass matrix. The ceramic product would be formed using a dry

feed, cold press, and sintering (heating) process without including Cs-137 in the ceramic matrix, rather than the wet feed, hot press process with the added Cs-137 radiation barrier as described for the ceramic immobilization alternative in Section 2.4.4.2. The advantages of using the cold press and sintering (heating) process would include increased throughput, simplicity, and proven production experience as used in the MOX fuel industry. Cold pressing would be an option for the ceramic CIC because the volatility of Cs-137 in the sintering process is not an issue with external radiation barrier variants. Using HLW as the radiation barrier, in lieu of the Cs-137 from cesium chloride capsules, would offer the advantages of process simplification and cost reduction, reduction in the potential for worker radiation exposure, and improvements in facility operations and maintenance requirements. If desired, the wet feed and cold press and sintering process could be used in the ceramic CIC variant.

A process schematic for the ceramic CIC process is presented in Figure O.4–1; the material flow diagram for this CIC process is the same as that shown in Figure O.3–2. As in the vitrification CIC variant, many of the feed materials require conversion to oxide form. Such treatment, conversion processing, and oxidation would take place in glove boxes. The resulting Pu oxide product would be fed to the ceramic process where Pu oxide would be blended with ceramic precursors and neutron absorbers. This mixture would be calcined, cold pressed, and sintered to produce densified pellets to be loaded into the small stainless steel cans. The cans of immobilized Pu in ceramic forms would be placed on a frame which would fit inside a DWPF canister and be transferred to the DWPF. These canisters would be filled with vitrified HLW at the DWPF to provide a radiation barrier for the final product.

Facility Description. The ceramic CIC variant could use part of the existing 221-F Canyon building including the Pu Storage Facility and New Special Recovery facilities, and part of the DWPF including the Vitrification and Service Buildings and the Glass Waste Storage Building. Table O.4–1 lists the location of each process area. Pit disassembly/conversion and other front-end processing, treatment operations, and immobilization (feed preparation, calcine and fill, press and package, and can filling) could be performed in the existing 221- F facility in areas specifically modified for Pu ceramic immobilization. The current F-Area facilities are designed and built to handle large quantities of Pu and have systems to maintain criticality control and safeguard systems to maintain accountability and security. The F-Area site layout was previously shown in Figure O.3–3. The floor area required for the front-end Pu processing and immobilization functions would be approximately 2,080 m^2 (22,390 ft^2). Total support area (heating, ventilation, and air conditioning, ingress, egress) would be an additional 6,500 m^2 (70,000 ft^2). The addition of the molten HLW as a radiation barrier in the canister would take place in the DWPF located in the S-Area. The S-Area site layout was previously shown in Figure O.3–4. The existing DWPF would be upgraded to satisfy the safeguards and security requirements for the handling of the stainless steel canisters containing Pu ceramic cans during the canister filling operations. Disruptions to current DWPF operations would be minimized since Pu would not be introduced into the DWPF in other than an immobilized, criticality safe manner. Safeguards and security provisions would be upgraded in selected portions of the DWPF buildings.

Facility Operations. In the ceramic CIC variant, the initial step in the immobilization process would be the transportation by SST, in Department of Transportation shipping containers, of Pu feed materials (such as pits, metal, oxides, and unirradiated reactor fuels) from storage site(s) to receiving facilities in the Pu Storage Facility and New Special Recovery facilities on top of the 221-F Canyon building at SRS. The shipping containers would be unpacked and accountability measurements conducted. Pu pits would be disassembled and converted to oxide.[2] Other forms would go through the minimum necessary processing to be converted to oxides. The oxide feed materials would then enter into specially modified portions of the 221-F Canyon and undergo ceramic immobilization in glove boxes as depicted in Figure O.3–2.

The feeds to this process would consist of ceramic precursors with a neutron absorber, dried titanate ion exchanger, and size-reduced Pu oxide powders which would be dry blended. Dry blending would be conducted

[2] Based upon the Preferred Alternative, Pu pits would be converted to MOX fuel and would not be immobilized.

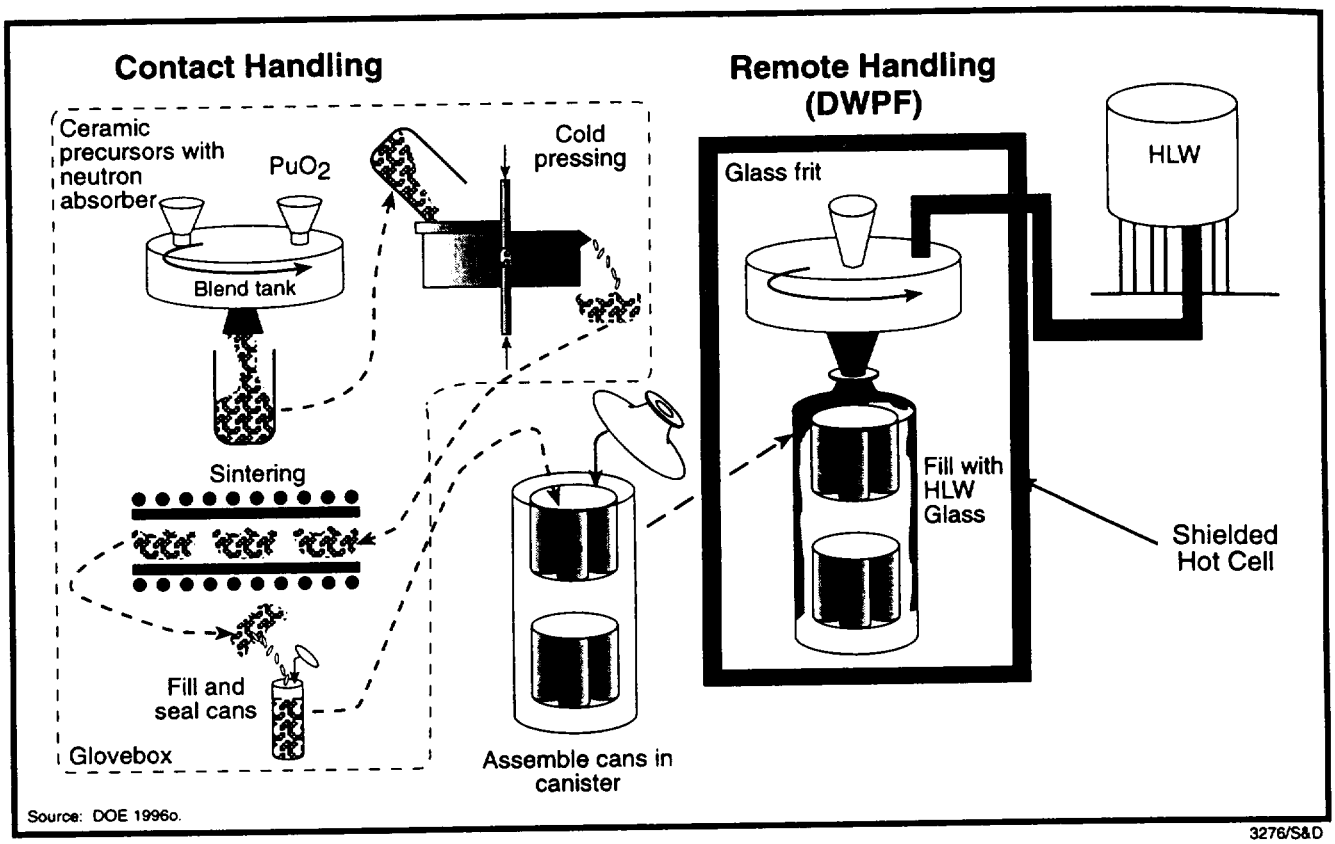

Figure O.4–1. Ceramic Can-in-Canister Variant Process.

Table O.4–1. Locations for Proposed Ceramic Can-in-Canister Process Equipment

Process	Location
Receiving, shipping, storage, sampling	221-F Pu Storage Facility
Pit disassembly, dehydride/hydride/oxidation	221-F New Special Recovery Facilities
Oralloy decontamination	221-F New Special Recovery Facilities
Special recovery	221-F Canyon 3rd Level
Fuel decladding	221-F Canyon 3rd Level
Feed preparation (dry feed)	221-F New Special Recovery Facilities
Oxide lag storage (dry feed)	221-F Canyon 3rd Level
Ceramic press and sinter (dry feed)	221-F Canyon 3rd Level
Off-gas treatment (dry feed)	221-F Canyon 3rd Level
Feed preparation (wet feed variant)	221-F Canyon 3rd Level
Oxide lag storage (wet feed variant)	221-F Canyon 3rd Level
Ceramic press and sinter (wet feed variant)	221-F Canyon 2nd Level
Off-gas treatment (wet feed variant)	221-F Canyon 2nd Level
Can decon (dry)	221-F Canyon 3rd Level
Can decon (wet feed variant)	221-F Canyon 2nd Level
Can weld and test (dry)	221-F Canyon 3rd Level
Can weld and test (wet feed variant)	221-F Canyon 2nd Level
Interim can storage	221-F Canyon 3rd Level
Place in canister	221-F Canyon 1st Level

Table O.4–1. Locations for Proposed Ceramic Can-in-Canister Process Equipment—Continued

Process	Location
Weld and test	221-F Canyon 1st Level
Interim canister storage	221-F Canyon 1st Level/DWPF Service Building Interim Vault
Blend tank	DWPF Vitrification Building Hot Cell
DWPF melter	DWPF Vitrification Building Hot Cell
Canister decontamination	DWPF Vitrification Building Hot Cell
Weld and test	DWPF Vitrification Building Hot Cell
Off-gas treatment	DWPF Vitrification Building Hot Cell
Interim product storage	DWPF Glass Waste Storage Building

Source: LLNL 1996k.

in a standard blending device such as a V-blender. The drying and calcining would be conducted in a rotary calciner that is a rotating tank inside a high temperature furnace. Following the drying and calcining process, the material would undergo milling and granulation to optimize size and morphology for the pressing and sintering operations. The dried and calcined ceramic precursor material loaded with Pu would then be poured into a feeder hopper which would deliver the oxide material into an automated pressing machine. The size of the pellets would be about 7 cm (2.75 in) in diameter by 2.8 cm (1.1 in) high. Pressed pellets would be transferred by a conveyer belt to the sintering oven and heated to 1,200 degrees Centigrade (°C) (2,200 degrees Fahrenheit [°F]) for several hours. After sintering, the pellets would be approximately 6.4 cm (2.5 in) in diameter by 2.5 cm (1 in) high. Any cracked or deformed pellets would be recycled. The Pu-ceramic pellets would then be loaded into 7.6 cm (3 in) diameter by 0.6 m (2 ft) high cans. The Pu loading in the ceramic form would not exceed 12 percent by weight. Cans loaded with the Pu-ceramic would be stored in storage racks in a vault on the third level of the 221-F Canyon building until ready to be placed in the DWPF canister.

When Pu-ceramic cans are ready for processing, they will be decontaminated and transported to the canister loading area of the F-Canyon. The cans would be placed in a holding rack inside an open DWPF canister (head removed). After loading a canister with approximately 20 Pu ceramic cans (5 cans in an array–4 arrays high), the head of the canister would be welded to the body and tested. Each canister would contain approximately 50 kg (110 lb) of Pu. The canisters may be placed in temporary storage in 221-F until shipped by rail or truck to the DWPF Service Building. A small storage vault, designed for special nuclear material and sized for about one week supply of canisters, would be provided in or adjacent to the DWPF Service Building to permit interim storage, if necessary, of canisters awaiting processing.

When a canister is ready to be filled, it would be transferred from the DWPF Service Building via a controlled corridor, to the Melt Cell in the Vitrification Building. Utilizing a melt pour turntable, the HLW-borosilicate glass would be poured into the canister and around the Pu-ceramic cans. The cans would then become encapsulated in the HLW-glass within the stainless-steel canister. After filing, the canister would be decontaminated and sealed by plug welding and transported to the Glass Waste Storage Facility for interim onsite storage.

The Glass Waste Storage Building, Unit 1, was designed and constructed to hold HLW-glass canisters until a HLW repository becomes available. Since the capacity of Unit 1 is 2,286 canisters, a Unit 2 building is planned to be constructed as Unit 1 fills with HLW-glass canisters. Since Unit 2 is scheduled to be built in support of the DWPF mission, only upgrades to facility safeguards and security and increased capacity would be necessary to accommodate the additional canisters resulting from the Pu disposition mission. Alternatively, Unit 1 could be upgraded in the event Unit 2 was determined to be unnecessary. Because the Pu-ceramic cans displace volume that would normally contain HLW-glass in a separate DWPF operation, additional DWPF canisters would be needed to process all of the HLW in the SRS tank farm and all of the surplus Pu. The number of additional

DWPF canisters would be directly proportional to Pu loading in the Pu-ceramic. The total number of DWPF canisters containing Pu-ceramic cans would be expected to be about 1,000. Assuming 20 cans per canister, the volume of HLW-glass displaced would be approximately 20 percent, or about 200 additional canisters to those required for the SRS tank farm HLW program.

This variant would process 5,000 kg (11,000 lb) of surplus Pu annually. A normal operating year for the facility would be 200 days with a nominal throughput of 25 kg (55 lb) of Pu per operating day. This operating schedule assumes 3 shifts per day, 7 days per week. Remote maintenance, accountability, criticality control, and other functions would be performed during the 165 days per year the facility would not be expected to operate.

Construction. The ceramic CIC variant utilizes existing SRS facilities to house the pit disassembly/conversion, Pu conversion, and immobilization operations. Under this variant, no major new construction would be required at SRS. However, some of the existing facilities would be modified and upgraded. Facilities in F-Area are designed and built to handle large quantities of Pu and have systems in place to maintain criticality control and safeguard systems to maintain accountability and security. However, the DWPF Vitrification Building, selected portions of the Service Building, and the Glass Waste Storage Building, Unit 1 or 2, would have to be upgraded to meet the safeguards and security requirements to support storage and handling of immobilized Pu, in accordance with DOE O 470.1, *Safeguards and Security Program.*

Radiological. With the CIC concept, the radiation source used to satisfy the spent fuel standard is the vitrified HLW outside the can. This eliminates the need for introducing radioactive Cs-137 into the immobilization process, thereby reducing radiation shielding/hot cell requirements and the potential radiation exposures to operating personnel and the public. The dose to workers and the public would be smaller for the CIC variant than the dose to workers and the public for the new facility analyzed in Section 4.3.4.2.9.

Waste Management. Since the Pu conversion and immobilization processes would be similar to those otherwise described for the ceramic immobilization alternative, implementation of the ceramic CIC variant would result in similar waste streams. The wastes generated as a result of the operation of this variant would consist of TRU, mixed TRU, low-level, mixed low-level, hazardous, nonhazardous (sanitary), and nonhazardous (other) materials. The ceramic CIC variant could offer the potential for significant reductions in the quantity of pollutants emitted and wastes generated when compared to the construction and operation of the other separate disposition facilities as detailed in this PEIS. Significant emphasis would be placed on the minimization of both liquid and solid wastes. Vitrified HLW would be used to surround the cans and fill the canisters; however, no HLW would be generated as a result of the Pu disposition operations. In addition, any criteria pollutants, hazardous air pollutants, and other toxic compounds and gases that may be emitted as a result of immobilization activities would be within permit requirements.

Transportation. Fissile material located at various DOE facilities would be transported by SST, in compliance with applicable regulatory requirements, to SRS and placed in onsite temporary storage. Intrasite transport of radiological materials that are not immobilized would be limited to the secure movements within the 221-F Canyon. Canisters containing the Pu-ceramic cans would be transported from 221-F to the DWPF Service Building via rail or truck. The filled canisters would be stored in the Glass Waste Storage Building, Unit 2, until shipment to a HLW repository is possible.

The Department is developing a rail shipping cask for DWPF canisters. This rail shipping cask would hold five DWPF canisters. Based on the use of this transport cask, it is estimated that, over the life of the project, 40 additional shipments from SRS to a HLW repository would be required.

O.5 REPOSITORY ACCEPTANCE

An analysis has been completed that examines the feasibility of introducing Pu-loaded glass into a HLW repository, using a CIC concept where the DWPF glass is poured into the space between the Pu glass cans and the DWPF canister (DOE 1996d:8-8). The DWPF glass acts as a radiation barrier to theft and diversion; gadolinium is added to the Pu glass cans as a neutron absorber. The analysis presented represents a case where the CIC concept for Pu disposition is a supplement to the disposition of the defense HLW already planned for the repository. The conclusions presented here for the vitrification variant are expected to apply to the ceramic CIC variant.

Regulatory. Any waste form that is accepted for disposal in a geologic repository must comply with the provisions of the *Nuclear Waste Policy Act* (NWPA), as amended. Under Section 2(12)B of the NWPA, the Nuclear Regulatory Commission (NRC) has the authority to certify this waste as eligible for the NWPA geologic repository. Such NRC action or legislative clarification in authorizing legislation will be necessary before this waste form can be considered for disposal in an NWPA repository. The final disposal of this waste form will have to conform to the licensing provisions of the NRC. Further, it is current DOE policy not to accept into the first HLW repository any wastes that include components regulated as hazardous under the *Resource Conservation and Recovery Act* (RCRA) (DOE 1995a:6). The absence of any RCRA-regulated hazardous materials in the final form would have to be demonstrated prior to acceptance into the HLW repository.

Criticality. The effective neutron multiplication factor (k_{eff}) for the intact glass form, assuming credit for the neutron absorbers during the post-closure period, is calculated to be 0.3 which is well below the 0.95 maximum value of k_{eff} allowed (10 CFR 60).

Thermal. The initial heat release from this CIC waste form is mostly from the HLW glass component. Temperatures peak between 30 and 60 years. By 100 years, the radiolytic heat generated by Pu will exceed the thermal output from the HLW. These predicted temperatures are far lower than, and therefore safely away from, the glass transition temperature of 400°C (750°F). The temperature and thermal output from these canisters are unlikely to materially affect the thermal balance of the repository.

Radiation. A comparison of the radiation exposure emanating from a repository waste package containing DWPF HLW glass canisters versus a package containing Pu cans and HLW glass shows that the radiation dose at the waste package surface is 81 roentgen equivalent man (rem)/hour (hr) for the package containing DWPF glass compared to the 30 rem/hr for one with the Pu cans with HLW glass. The radiation level for the package incorporating Pu CIC is below the threshold value for radiolytic corrosion, so no additional thickness of the copper-nickel alloy waste package outer barrier would be required to reduce the radiation to an acceptable level (100 radiation absorbed dose/hr) to protect the waste package from radiolysis-induced corrosion. However, additional shielding would be required to protect workers. Doses at a distance of 2 m (6.6 ft) from the waste package surface show values of 12.5 rem/hr for the DWPF glass and 4.7 rem/hr for the CIC. For emplacement in the repository, only 7 cm (2.75 in) of lead thickness must be added to the CIC waste package underground transporter to reduce the radiation doses to meet the standard allowable dose of 10 millirem/hr at 2 m (6.6 ft) from lateral outer surfaces (49 CFR 173.441) to ensure worker protection versus 10 cm of lead for repository waste package containing DWPF HLW glass canisters.

Releases. The calculated dose contribution to the accessible environment from all the CIC waste packages would be nearly 100 times lower than the calculated peak dose from a repository that contains only commercial spent fuel and HLW. This is to be expected because the repository release would be dominated by the greater quantity of commercial, uranium-based spent nuclear fuel.

O.6 TESTS AND DEMONSTRATIONS

The department has received comments on the Draft PEIS requesting that "pilot plant" studies of the CIC concept be performed. In this regard, DOE has initiated research, development, and testing of various aspects of the CIC concept for both the vitrification and ceramic immobilization alternatives. For example, in January 1996, a cold (without radionuclides) demonstration of the CIC variant was successfully conducted at the DWPF. Small cans containing a high temperature glass with a Pu surrogate were loaded into two full-size DWPF canisters (one canister contained 8 cans and the other 20 cans) which were subsequently filled with surrogate HLW-glass in the DWPF as part of the cold stratup qualification tests of that facility. Other tests are being conducted to demonstrate techniques that could enhance the nonproliferation properties of the final product. Ceramic waste forms have been under development for HLW for many years; however, the application of this technology to the immobilization of Pu is currently developmental.

The DOE plans to continue research to determine whether glass or ceramic is the preferred form for Pu disposition and to establish the optimum Pu concentration and chemical composition of a waste form that can be readily processed and satisfy nonproliferation concerns and perform well after the emplacement in a geologic repository. In addition, developmental efforts are underway to design acceptable processing equipment and controls and to demonstrate on a pilot scale the integration of the individual processing steps. As part of this effort, the Pacific Northwest National Laboratory, Argonne National Laboratory, Lawrence Livermore National Laboratory, and SRS are each contributing to a cooperative, integrated program of testing and evaluation of forms and processes.

Should DOE select either the vitrification or ceramic immobilization disposition alternative or the CIC variant, these activities will provide the information needed to demonstrate technical viability and practicality of the Preferred Alternative for disposition as well as provide information useful for DOE's tiered NEPA reviews (and RODs) regarding the selection of a specific technology variant, location, and/or facilities.

Appendix P
Manzano Storage Facility

The *Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapon Components* (Pantex EIS) analyzed the storage of plutonium (Pu) in a pit form. Under the Pit Storage Relocation Alternative in the Pantex EIS, the pit storage function currently carried out at Pantex Plant (Pantex) would be transferred to another site. The Manzano Weapons Storage Area (WSA) at the Kirtland Air Force Base (KAFB) near Albuquerque, NM, is one of the candidate sites for the storage of pits. The WSA represented a reasonable alternative for this mission because the pits to be potentially relocated from Pantex exist in a sealed and stable metal form and will be packaged in specially designed containers that will ensure a very low probability of breach, metal oxidation, and dispersion of oxidation products. Therefore, the Manzano WSA was considered in the Draft Programmatic Environmental Impact Statement (PEIS) but eliminated as a reasonable alternative primarily because Manzano WSA could not accommodate storage of both pit and non-pit materials.

Since the issuance of the Draft PEIS, the Department of Energy (DOE) has developed a Preferred Alternative for storage that would separate storage of most Pu pits from storage of non-pit Pu material. Specifically, the Preferred Alternative would store Pu pits from Pantex and Rocky Flats Environmental Technology Site (RFETS) at Pantex, and would store non-pit Pu at Savannah River Site, Hanford Site, and Idaho National Engineering Laboratory. Since DOE's Preferred Alternative would separately locate storage of pits and non-pit Pu from RFETS, the option to store pits at Manzano WSA no longer appears unreasonable. Therefore, DOE has added this appendix to the Final PEIS, which discusses potential storage of Pantex and RFETS pits at Manzano WSA.

For a number of reasons, the Preferred Alternative would store the pits from Pantex and RFETS at Pantex, rather than Manzano WSA. Pantex is the proposed site for interim storage of pits under the Preferred Alternative in the Pantex EIS.[1] The majority of the pits that require storage are surplus to U.S. defense needs and are already located at Pantex. The number of pits that would be relocated from RFETS would be small by comparison. Since the majority of pits are already in storage at Pantex, it would be prudent for DOE to consolidate all pits there for storage. Assembly and disassembly operations would continue at Pantex even if pit storage did not occur there. Selecting Manzano WSA would require DOE to create another site where Pu would be located with the risk of contamination and the associated costs for site infrastructure and security. In addition, other missions that could be added to Pantex (for example, pit disassembly/conversion or mixed oxide fuel fabrication) could not be added to Manzano.

Storage at Manzano WSA would involve the transportation risk of moving these materials from Pantex to Manzano WSA. Furthermore, two shipment campaigns would be required for disposition for most of the pits (those already at Pantex) if Manzano WSA were chosen, whereas only one shipment campaign of those same pits would be required if the pits were stored at Pantex. For the Manzano case, pits at Pantex would require relocation to Manzano and then a second shipment campaign to a disposition site. Leaving the pits in storage at Pantex would result in only one shipment campaign from Pantex to the disposition site.[2] This appendix incorporates applicable information from the Pantex EIS.

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[1] The disposition of these surplus pits would begin within the next 10 years and would be completed within the next 25 years. The time period required for the storage of the pits is therefore close to that considered in the Pantex EIS for pit storage and the reasons for not using Manzano WSA are the same.

[2] Two shipment campaigns of pits would be required for those pits currently stored at RFETS for both Pantex and Manzano.

P.1 MANZANO WEAPONS STORAGE AREA

The KAFB is an Air Force Materiel Command base sharing base facilities and infrastructure with a number of major tenants, including DOE, Sandia National Laboratories (SNL), the Defense Special Weapons Agency, and Phillips Laboratory (Figure P.1–1). The base covers an area of 21,320 hectares (ha) (52,600 acres) on the southeast boundary of Albuquerque, New Mexico. Approximately 8,300 ha (20,500 acres) of this area is withdrawn public lands (USAF 1993a:1-3). Major Air Force units at KAFB include the 377th Air Base Wing, 58th Special Operations Wing (which performs helicopter crew training and pararescue training) and Phillips Laboratory (which performs research and development for space systems, ballistic missiles, geophysics, and directed energy systems). SNL conducts research and development for space systems, testing, stockpile surveillance, and the transportation of nuclear materials (USAF 1993a:3-2).

The Manzano WSA at KAFB consists of four plants inside Manzano Mountain (used primarily for research activities) and 122 magazines, of which 81 are earth covered and 41 are tunneled into the mountainside (KAFB 1993a:13) (Figure P.1–2). Construction began in June 1947, and the facility became operational in April 1950. In June 1992, the Manzano WSA was deactivated, and Phillips Laboratory assumed responsibility for its maintenance. SNL continues to provide minimum security, although the Perimeter Intrusion Detection and Alarm System was deactivated with the termination of the main mission in 1992. The Manzano WSA has enough magazine space to store the pits from Pantex and RFETS. The proposed location for the storage of pits is the set of 41 magazines that are tunneled into the mountainside. As many as 35 magazines have overburden greater than 9 meters (m) (30 feet [ft]) of earth and granite. The existing fence would be reactivated to the extent necessary, and no new fence or security systems would be required. If in the Pantex EIS, DOE chooses to do storage of pits in the Manzano WSA at KAFB, a pit placement, retrieval, and inventory system would have been implemented. The storage areas at the Manzano WSA are well suited for the Stage Right equipment and techniques successfully implemented at Pantex Plant.

P.2 AFFECTED ENVIRONMENT AND ENVIRONMENTAL IMPACTS

The environmental resources discussed below have been assessed for KAFB. Analyses have shown that the impacts to some resources from the potential storage of pits at the Manzano WSA are small enough to warrant only limited discussion. Therefore, the resources are discussed commensurate with their impacts.

P.2.1 LAND RESOURCES

The Manzano WSA is currently being used in part for storage of a variety of items such as furniture and document boxes. These items would easily be removed and space made available for storage of surplus pits. The use of some storage magazines for the storage of pits would not change the array of potential storage materials for which these weapon storage magazines are designed. Additionally, no land disturbance is projected as it pertains to the Manzano WSA. Impacts to land use would not be expected.

P.2.2 SITE INFRASTRUCTURE

The KAFB infrastructure is managed by the 377th Air Base Wing and includes support to all tenants. As the Manzano WSA is not a DOE site, the exact breakdown of infrastructure support activities that would be performed by KAFB and DOE personnel has yet to be worked out in detail. Should this site be selected for pit storage, a Memorandum of Understanding between the Department of the Air Force and DOE would be developed detailing these duties. The infrastructure operations at KAFB that could be affected by or be expected to directly support pit storage operations include security, vehicle and building maintenance, safety and health protection, utilities, administration, and general support (for example, cafeteria, general stores).

The direct impacts from the implementation of pit storage would include a small increase in the site's security force. Electrical usage due to long-term pit storage (estimated to be 4,110 megawatt-hours per year [MWh/yr])

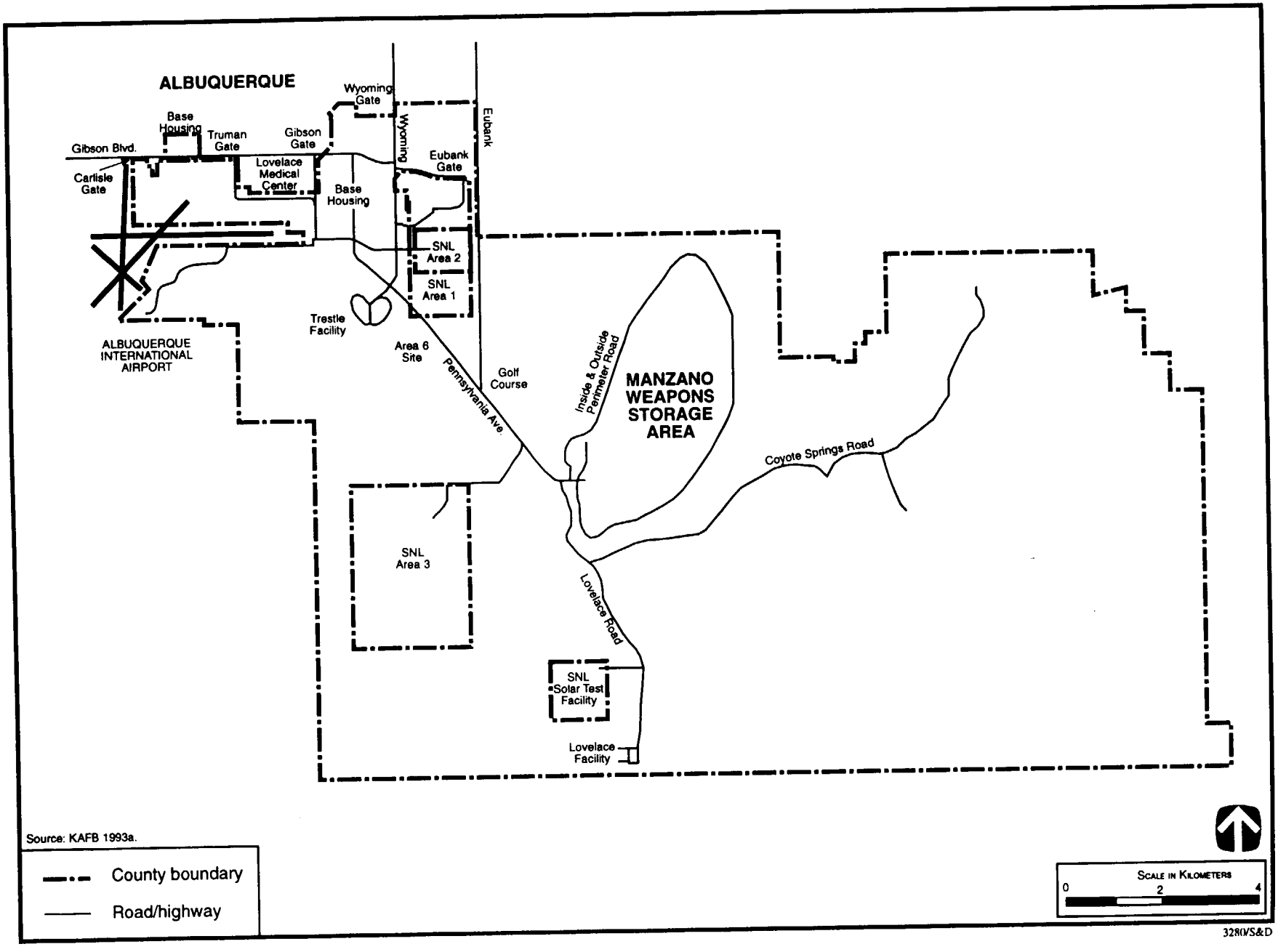
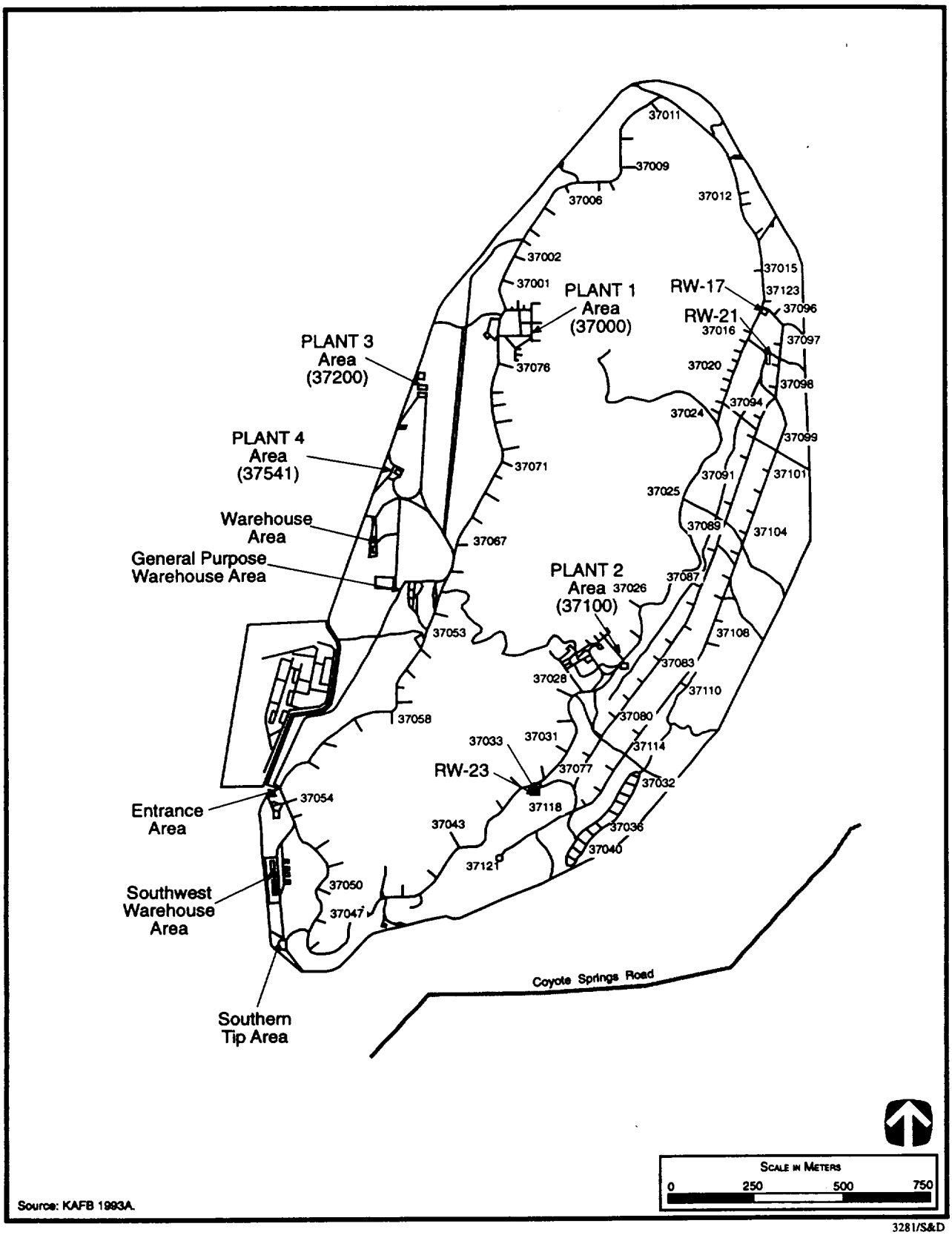


Figure P.1–1. The Manzano Weapons Storage Area at Kirtland Air Force Base, New Mexico.



Source: KAFB 1993A.

3281/S&D

Figure P.1–2. Layout of the Manzano Weapons Storage Area at Kirtland Air Force Base.

represents a 0.8 percent increase over the site's fiscal year 1993 usage of 490,000 MWh/yr and 0.4 percent of the fiscal year 1993 system capacity of 1,095,000 MWh (USAF 1993a:3-17). Maintenance support and indirect impacts resulting from pit storage worker requirements (for example, water, wastewater treatment, and fuel) would increase minimally in comparison to the current and historical onsite infrastructure support levels and system capacities. The Manzano WSA is not currently being utilized at historical or design levels; therefore, the utility systems supporting this facility generally have excess capacity available to support pit storage activity.

P.2.3 AIR QUALITY AND NOISE

Since Manzano WSA is in a moderate nonattainment area for carbon monoxide (CO), air quality analysis is performed at a greater detail for this site. This section provides a more detailed analysis.

P.2.3.1 Air Quality Affected Environment

The Manzano WSA at KAFB is located in Bernalillo County, which is situated in the Albuquerque-Mid Rio Grande Intrastate Air Quality Control Region 152. The Manzano WSA lies outside the City of Albuquerque and is classified as better than national standards for sulfur dioxide, unclassifiable/attainment for ozone (O_3), unclassifiable for particulate matter less than or equal to 10 microns in diameter (PM_{10}), cannot be classified or better than national standards for nitrogen dioxide (NO_2), attainment for CO, and not designated for lead. For CO, Bernalillo County has not had a violation during the past 3 years. As of July 15, 1996, the Environmental Protection Agency (EPA) redesignated Bernalillo County from nonattainment to attainment for CO. The nearest Prevention of Significant Deterioration (PSD) Class I area to the Manzano WSA is the Bandelier Wilderness, approximately 80 kilometers (km) (50 miles [mi]) to the north. The Manzano WSA has no emission sources subject to PSD requirements.

P.2.3.2 Air Quality Environmental Impacts

There are no direct criteria pollutant emissions from the pits during storage. Indirect pollutant emissions would be produced from the exhausts of the vehicles used by employees used to commute to and from work. Also, exhaust emissions from the tractor-trailers used to transport the pits from Pantex to the Manzano WSA would contribute a small amount of pollution to the overall pollutant burden in Bernalillo County, NM.

The calculation of emission rates of exhaust pollutants from employee and pit delivery vehicles was made based on emission factors obtained from the EPA Mobile Source Emission Factor Model (MOBILE 5a). The following assumptions were used in calculating the exhaust pollutant emissions:

- 120 vehicles would be used by security employees (365 days/yr)
- 30 vehicles would be used by operations staff employees (255 days/yr)
- Average roundtrip commute distance: 48 km (30 mi)
- Pit delivery truck roundtrip distance from Pantex in Bernalillo County: 80 km (50 mi)

Table P.2.3.2–1 presents the estimated annual pollutant emissions from employee and pit delivery vehicles. A comparison of these emissions with those in Bernalillo County is also provided in the table. Table P.2.3.2–1 shows that the resulting increase in the CO emission due to storage of pits at the Manzano WSA would be 0.08 percent. Also, these emissions from mobile sources would be distributed over a relatively large area. The increases in the ambient concentrations would, therefore, probably not be detectable and would not cause an increase in the violations of the CO ambient air quality standard (Bernalillo County is currently an attainment area for CO). Nor would these negligible increases cause any violations of the National Ambient Air Quality Standards (NAAQS) for the other criteria pollutants. Also, these small emission increases would not slow

progress in attaining the CO standard. The air quality impacts resulting from the storage of pits at the Manzano WSA would therefore be negligible.

Table P.2.3.2–1. Pollutant Emission Rates Related to Storage of Pits at the Manzano Weapons Storage Area

Source	Pollutant Emission Rate		
	CO (kg)	NO_2 (kg)	VOC (kg)
Employee Vehicles	19,940	3620	2,080
Pit Delivery Vehicles	40	50	10
Total	19,980	3,670	2,090
Bernalillo County (1993)	26,303	NA	NA
Percent of County Emission	0.08	NA	NA

Note: NA=emission factors not available.
Source: PX DOE 1996b.

P.2.3.3 General Conformity Determination

The EPA published the General Conformity Rule 40 Code of Federal Regulations (CFR) parts 6, 51, and 93 on November 30, 1993, to implement section 176(c) of the *Clean Air Act* (CAA) as amended in 1990. This section requires that Federal action conform to the appropriate State Implementation Plan. Conformity, as defined in the CAA, is conformity to the State Implementation Plan's purpose of eliminating or reducing the severity and number of violations of the NAAQS and achieving expeditious attainment of such standards. A formal conformity determination is required for Federal actions occurring in nonattainment areas when the total direct and indirect emissions of nonattainment pollutants (or their precursors) exceed specified annual de minimis (threshold) values. Because O_3 is a secondary pollutant, the conformity determination for O_3 uses the precursor emissions of volatile organic compounds and NO_2 as surrogate pollutants. The threshold values are presented in Table P.2.3.3–1. Since the Manzano WSA is in a maintenance for attainment area for CO, the threshold value for CO is 90.7 metric tons (100 short tons) per year (Table P.2.3.2–1). As shown in Table P.2.3.3–1, the emission rate for CO is well below the threshold value. Therefore, a general conformity analysis is not required for the Manzano WSA.

Table P.2.3.3–1. Threshold Values

Criteria Pollutant	Degree of Nonattainment	Emission Rate (kg)
Ozone (VOCs and NO_2)	Serious	45,400
	Severe	22,700
	Extreme	9,100
	Other ozone nonattainment areas (outside of ozone transport region)	90,700
VOCs	Marginal/moderate nonattainment (within ozone transport region)	45,400
NO_2	Marginal/moderate nonattainment (within ozone transport region)	90,700
CO	All	90,700
PM_{10}	Moderate	90,700
	Serious	63,500
SO_2/NO_2	All	90,700
Pb	All	22,700

Source: PX DOE 1996b.

P.2.3.4 Noise

The major sources of noise within KAFB include blasting and explosives testing, aircraft operations, and equipment and machine operations. The only additional sources of noise associated with pit storage operations would be from transportation vehicles. These impacts would be minimal.

P.2.4 WATER RESOURCES

Because of the nature of the pit storage activities, operations at the Manzano WSA would not affect surface water or groundwater. The pit storage activities would not use surface waters at the Manzano WSA. The Manzano WSA has several springs and seeps. Four springs are located on the mountains that make up the Manzano WSA (USCOE 1995b:22,23,27). Some magazines show evidence of water intrusion (KAFB 1993a:48). These magazines were designated as unsuitable for pit storage and would not be used. The sanitary sewer waste from the Manzano WSA would be discharged to approved septic systems. The wastewater would not have a measurable affect on groundwater quality because of the combined effects of a deep water table (15 to 30 m [50 to 100 ft]), low additional discharge volumes, high evaporation rates, and a composition and concentration consistent with treated and sanitary wastewater. The water demands of pit storage operations are solely due to use by storage personnel. The water demands would be less than historical usage at the Manzano WSA and negligible in comparison to the 6.4 billion liters (1.7 billion gallons) used annually at KAFB (USAF 1994a:3-20). The Manzano WSA is located outside of the 100-year and 500-year floodplains (USAF 1979a).

P.2.5 GEOLOGY AND SOILS

The only aspects of geology and soils resource area that could be affected by or have an effect on the implementation of long-term pit storage at the Manzano WSA are the risks associated with earthquakes. The earthquake risk was assessed and found to be bounded by other accidents, as discussed in Section P.2.9. The Manzano WSA is not anticipated to require upgrades that would involve land disturbance; therefore, impacts to soils are not anticipated.

P.2.6 BIOTIC RESOURCES

No Federally listed threatened or endangered plant and animal species have been reported from the Manzano WSA, although the peregrine falcon and the bald eagle may be occasional KAFB migrants (USAF 1994a:3-8, 3-9). However, three specimens of the grama grass cactus, a species of concern, were noted just west of the perimeter fence near the Manzano WSA administrative complex. The western burrowing owl, another species of concern, has been reported 1.6 km (1 mi) west of the Manzano WSA perimeter fence, but not within that facility's boundary. Additionally, two State endangered plants, the viznagita cactus and Wright's fishhook cactus, were found together on gravelly or rocky slopes at nine sites within the Manzano WSA (NM NHP 1995a:15,C-176,C-177). Further, four springs were identified within the perimeter of the Manzano WSA (USCOE 1995b:ES,15-27). However, the long-term storage of pits does not include any action that would disturb the animal or plant species noted above or any of the four springs. Therefore, no impacts to biotic resources would be expected.

P.2.7 CULTURAL RESOURCES

Twenty-seven historic and prehistoric archaeological sites have been found in the Manzano WSA. Of these sites, 8 have been recommended for inclusion in the National Register of Historic Places (NRHP) and 14 others are considered to be potentially eligible for inclusion (ANL 1995c:1-1,1-2,8-2–8-6). The storage of pits would not include any action that would disturb these resources. No storage facilities identified have been nominated to the NRHP. Therefore, no impacts to cultural and paleontological resources would be expected.

To identify areas of potential concern and locations, DOE has, in the past, sought consultations with Native American groups with traditional ties to the area. Two of these groups, the Sandia and Isleta Pueblos, expressed a general concern about the Manzano WSA. Isleta Pueblo considers the Four Hills area that comprises the Manzano WSA to be within their traditional area of cultural activities. They have requested that KAFB inform them of any archaeological finds at the Manzano WSA, specifically in regards to human remains and ritual objects (ANL 1995c:1-1,1-2). The long-term storage of pits is not expected to affect these concerns.

P.2.8 SOCIOECONOMIC RESOURCES

Approximately 150 additional personnel (including 120 security personnel) would be required to operate the storage magazines at the Manzano WSA if pit storage activities were moved to this facility. This number represents less than a 0.8-percent increase in the total Federal workforce at KAFB. Most of these workers could be hired locally; therefore, the increase to the KAFB workforce or the regional population would not be significant. According to the 1990 Census, 150 workers represent 0.06 percent of the of the workforce employed within the KAFB region of influence (ROI) (Census 1993m:202-205). No socioeconomic impacts would be anticipated.

P.2.9 OCCUPATIONAL AND PUBLIC HEALTH

The basic approach used in assessing human health concerns is to first identify the affected environments and establish a baseline that represents the risk from current operations. Changes in this baseline risk resulting from the long-term storage of pits are then examined for both normal operations and potential accidents.

In the Pantex EIS, the assessment of the human health risk impact from potential accidents that results from storing the pits in the Manzano WSA involved a risk screening process. The first step in this process was to identify a broad spectrum of potential accident scenarios. The second step in the process used screening techniques to identify the specific scenarios that dominate risk. Rigorous consequence evaluations are only performed for the identified risk-dominant scenarios.

Two types of accident consequences are examined:

- Worker and public exposure
- The probability of the accident causing fatal cancer in a worker or the public

If DOE chooses to relocate pits to KAFB, two aspects of this relocation contribute to a potential for environmental impacts. They are the impacts associated with the following:

- Transferring pits from the transporter to their storage location inside the facility
- Storage itself (for example, potential impacts resulting from having the pits reside inside the facility)

Each time pits are transferred from the transporter to their storage location inside the facility, there is a small probability that an accidental release could occur due to a handling accident. In addition, the transfer of pits from the transporter to their storage location would result in radiological exposures to involved workers.

P.2.9.1 Affected Environment

The release of radioactivity and toxic chemicals to the environment from a DOE facility is an important issue for onsite workers and the public. Since the human environment contains many sources of radioactivity and toxic chemicals, it is essential to understand the sources of these substances and how effectively they are controlled.

Table P.2.9.1–1 summarizes the major sources of radiation exposure in the vicinity of the Manzano WSA. The average annual probability of contracting a fatal cancer in the State of New Mexico is 1.4×10^{-3}. Using a nominal

Table P.2.9.1–1. Major Sources of Radiation Exposure in the Vicinity of the Manzano Weapons Storage Area at Kirtland Air Force Base

Source of Exposure	Dose to Average Individual (mrem/yr)	Percentage of Total Exposure
Natural Background Radiation		
Cosmic and external terrestrial	119	84.8
Internal terrestrial	39	
Radon in home	200	
Total natural	358	
Medical Radiation		
Diagnostic x rays	39	12.6
Nuclear medicine	14	
Total medical	53	
Other Sources		
Weapons test fallout	<1	2.6
Consumer and industrial products	10	
Air travel	1	
Nuclear facilities (other than transportation of radioactive materials)	<1	
Manzano/Sandia-environmental radioactivity	4×10^{-8}	
Total other	11	
Total (All Sources)	**422**	**100**

Source: NCRP 1987a.

fatal cancer risk factor of 5×10^{-4} cancer fatalities per person rem and the environmental radioactivity data for Manzano/Sandia in Table P.2.9.1–1, it is calculated that fatal cancers attributable to environmental radioactivity released in the vicinity of the Manzano WSA and SNL constitute an extremely small fraction (<0.01 percent) of the average yearly fatal cancer probability in the State of New Mexico (NM DOH 1995a:1).

Figure P.2.9.1–1 depicts the offsite population within an 80-km (50-mi) radius of the Manzano WSA. Windspeeds and directions in the Manzano WSA vicinity are presented in Figure P.2.9.1–2. Winds are predominantly southerly during the summer and northerly during the winter.

P.2.9.2 Environmental Impacts

Human health impacts from pit storage activities could potentially result from normal operations and accident scenarios. Impacts from normal operations would be confined to onsite workers. Normal operational impacts result from the unloading of pits from safe, secure trailers (SSTs) at the Manzano WSA. Unloading operations would result in radiological exposure to cargo handlers. Based on conservative calculations made for handling of pits at Pantex Plant, the worker doses from unloading of 2,000 pits per year are estimated to be 27 person-roentgen equivalent man (rem) per year or 270 person-rem for the unloading of pits. Once removed from the SSTs, pits would be transferred into the Manzano WSA for storage. Pit transfers within the Manzano WSA would result in radiological exposures to onsite workers handling the pits. The transfer of pits would result in worker doses of less than 2 person-rem per year for handling 2,000 pits and 13 person-rem for the placement of pits. The combined worker dose from unloading and storage of pits at the Manzano WSA would be 283 person-rem distributed over the 30 people directly involved in material movement. Over a period of

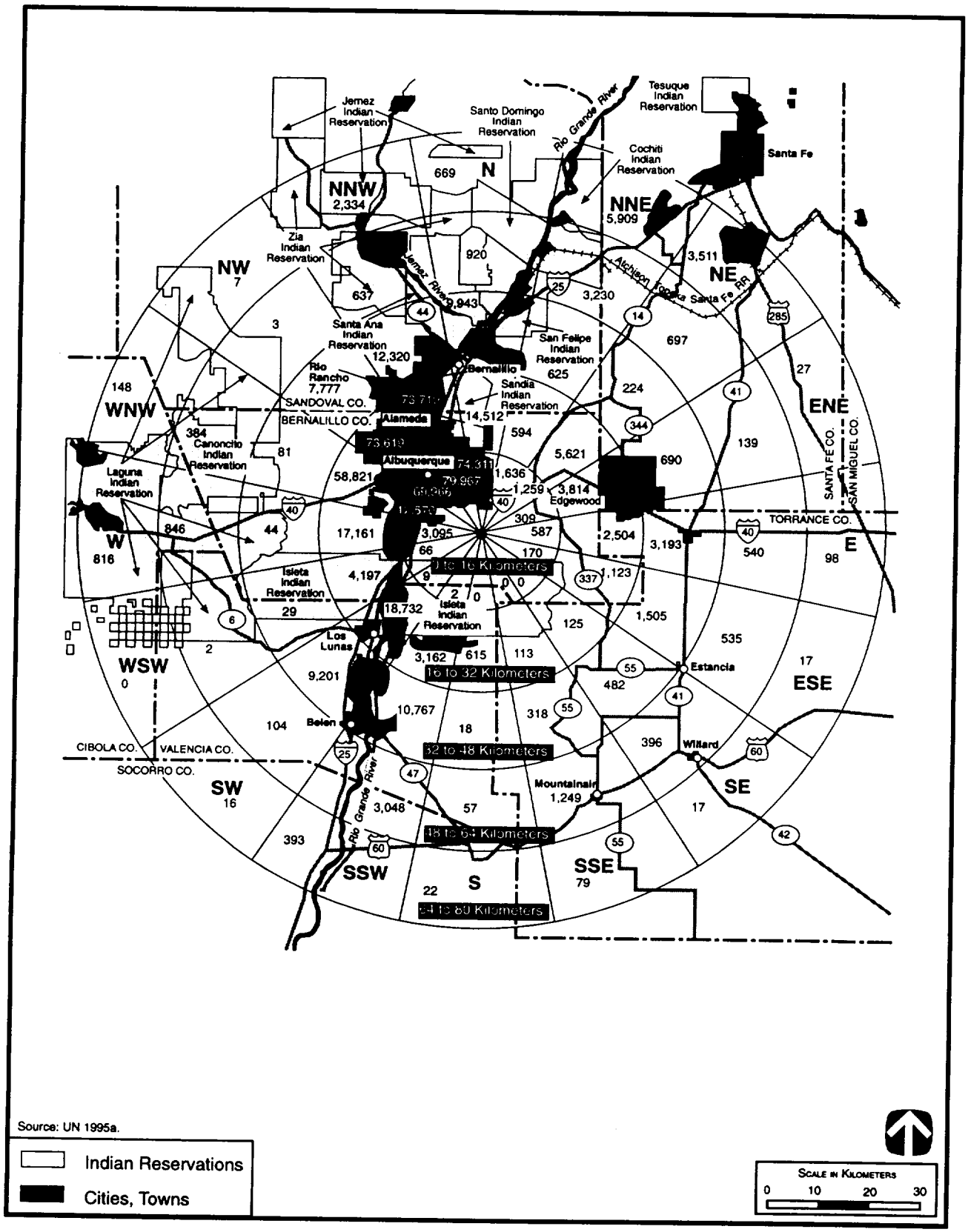
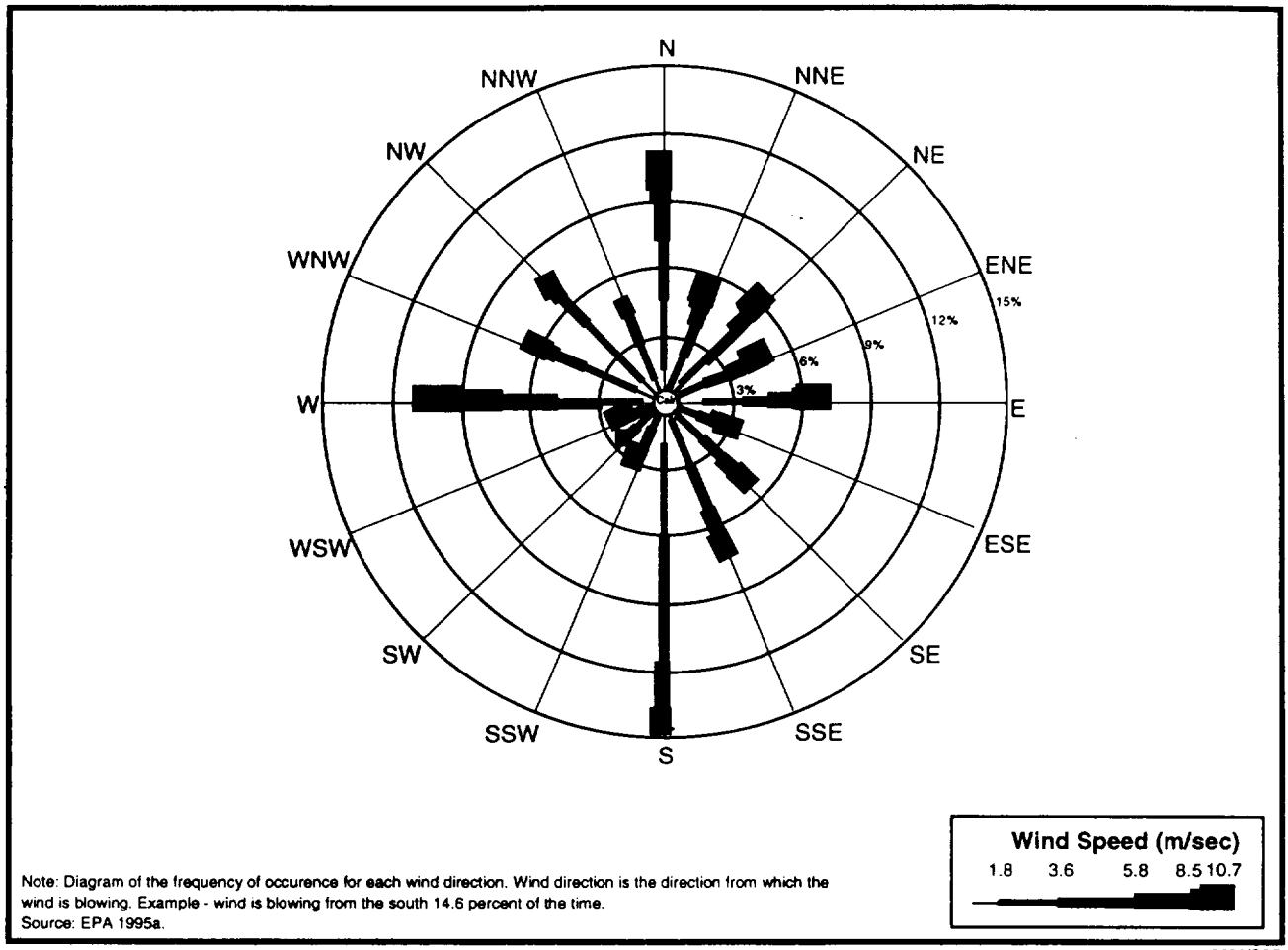


Figure P.2.9.1–1. Offsite Populations in the Vicinity of the Manzano Weapons Storage Area.



Note: Diagram of the frequency of occurence for each wind direction. Wind direction is the direction from which the wind is blowing. Example - wind is blowing from the south 14.6 percent of the time.
Source: EPA 1995a.

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Figure P.2.9.1–2. Wind Direction and Speed at Albuquerque International Airport.

10 years and using a dose-to-risk conversion factor of 4×10^{-4} latent cancer fatality (LCF) per person-rem, there would be an additional 0.11 LCF experienced by this group due to radiological exposure from pit handling. For long-term storage, over a period of 50 years, assuming the handling of 2,000 pits per year and no movement to or from the WSA, there would be an addition 0.04 LCF experienced by the workers.[3]

Some operational accidents could result in impacts to both onsite workers and the offsite general population. Radiological exposures and the resultant risk of latent cancers have been evaluated. The probability of an onsite worker or an offsite member of the general public contracting a fatal cancer resulting from accidental radiological exposure was calculated using the Melcor Accident Consequence Code System (MACCS) computer code.

The radiological health risk from accidents associated with the storage of pits is dominated by handling accidents that could occur when the pits are being transferred from the transporter. A standard tine forklift is likely to be used to remove pit containers from an SST. The probability of a standard tine forklift causing a puncture during a single handling operation is in the range 10^{-4} to 10^{-6} (that is, extremely unlikely as defined by DOE orders). It is estimated that a forklift puncture of a pit container would release 9.2×10^{-5} curies of Pu.

[3] For the Preferred Alternative, surplus Pu material would not be stored for 50 years but until disposition occurs.

This is a conservative estimate of the respirable, airborne release caused by a puncture of one shipping container (PX DOE 1992f:7-39).

Given such a release, an involved worker (the forklift driver) would receive a dose of 6.6 rem, corresponding to an incremental increase in an LCF of 2.6×10^{-3}. In addition, a noninvolved worker 100 m (328 ft) downwind along the center line of the Pu dispersion plume would receive a 5.2×10^{-2} rem exposure, corresponding to an incremental increase in an LCF of 2.1×10^{-5}. The maximally exposed individual of the public would be expected to receive an exposure of 6.7×10^{-3} rem, corresponding to an incremental increase in an LCF of 3.4×10^{-6}.

This event would result in an exposure of 4.0×10^{-2} person-rem to the public within 50 km (80 mi). Considering the likelihood and consequence of this event, on the average, a member of the public will have an increased annual risk of developing a fatal cancer from this potential accident of 2.6×10^{-14} fatal cancers per year. The annual fatal cancer risk to a person in the State of New Mexico from all other causes is 1.4×10^{-3} fatal cancers per year.

Pit container inventories at the Manzano WSA are expected to be performed using either shielded or automated techniques and equipment. Consequently, these normal operations are not expected to result in any significant radiological exposure to workers.

P.2.9.3 Aircraft Accidents

The Manzano WSA is located in the foothills of the Manzano Mountains, approximately 6.5 km (4 mi) southeast of the main (east-west) runway of the Albuquerque International Airport. Figure P.1–1 shows the locations of the Manzano WSA relative to the two runways for the Albuquerque International Airport, one of three airports in the vicinity of the Manzano WSA. The Albuquerque International Airport is the major commercial airfield in the State of New Mexico; it is the only airport with regular commercial jet service. In addition to its role as a commercial airfield, the Albuquerque International Airport is utilized by military aircraft stationed at Kirtland Air Force Base. In 1994, the Albuquerque International Airport had 220,914 aircraft operations (takeoffs and landings) (FAA 1996a:1). Table P.2.9.3–1 summarizes the total number of airfield operations at the Albuquerque International Airport.

In addition to the Albuquerque International Airport, there are two other airports within the Albuquerque area. Coronado Airport is located approximately 19 km (12 mi) to the north-northwest, has two runways, and is used by general aviation aircraft. Similarly, Alameda Airport is located approximately 24 km (15 mi) to the northwest, has two runways, and is also used by general aviation aircraft. Both of these airports are outside the boundary for general aviation aircraft and were therefore not included in the aircraft crash analysis. Only the Albuquerque International Airport and nonairport (in-flight) aircraft were included in the analysis.

Table P.2.9.3–1. Albuquerque International Airport Operations for 1994

Aircraft Type	Number of Operations
Air Carrier	77,978
Air Taxi	41,349
Military	29,929
General Aviation	71,658
Total Airfield Operations	220,914

Source: FAA 1996a:1.

In the history of the Manzano WSA, there have been three aircraft crashes. One crash involved an F-100C; the crash site is located east of the Manzano administration area. Another crash involved a B-29 in the northern portion of the site. This aircraft departed from Kirtland Air Force Base and crashed after approximately

3 minutes in flight, killing the crew. The third crash also occurred in the northern portion of the site and involved an EC-135 (KAFB 1993a:69,73,74). None of these crashes affected the storage facilities.

If DOE chooses to relocate pits to the Manzano WSA, the pits would be stored in Type D magazines. Type D magazines (as shown in Figure P.2.9.3–1) have access tunnels that vary in length from 20 m to over 30 m (65 ft to over 100 ft). The main chambers are approximately 19 m (61 ft) long and have the capacity to store up to 800 pit containers each in a Stage Right configuration using a shielded forklift to stack containers. In addition, the main chambers are protected by two vaultlike steel doors at both ends of the access tunnel.

Type D facilities are tunneled into the mountainside, which provides significant earth overburden protection from penetrating aircraft. As many as 35 magazines have overburden greater than 9 m (30 ft) and are potentially available for pit storage. The frequency of an aircraft impacts at the Manzano WSA is relatively high compared with other potential storage sites. However, the earth overburden of Type D magazine provides complete protection against potential damage from aircraft impacts.

At Manzano, the potential exists for airplanes overflying the area to be carrying conventional bombs. An analysis was performed to determine whether expected bomb loads (one to four 909-kg [2,000-lb] bombs) could damage the Manzano storage magazines in the event of an airplane crash. With the minimum overburden cover of 9 m (30 ft) of granite and earth, the magazines cannot be damaged by any foreseeable aircraft events (Army 1986a: 3-19).

Using the Final DOE Standard for determining the probability of aircraft crashes and 1994 data from the FAA, the frequency of hitting one of the 25 Type D magazines was calculated as 8.8×10^{-5} for all types of aircraft (*DOE Standard, Accident Analysis for Aircraft Crash into Hazardous Facilities, SAFT-0030*). It should be noted that the frequency calculation represents a conservative upper bound. Since this frequency is greater than 1×10^{-7}, the Final DOE Standard stated that further analysis was needed. A structural analysis was done according to the Final DOE Standard for the facility with a 9-m (30-ft) overburden. The analysis was done for the maximum penetrator missile for each of the aircraft categories except for helicopters. None of the aircraft missiles penetrated the facility. Since this frequency is 0, the DOE Standard stated no further analysis was needed.

P.2.10 WASTE MANAGEMENT

For the purpose of this assessment, it is assumed that DOE's SNL would manage the wastes from pit storage at the Manzano WSA. Waste management figures from SNL are used for comparison. SNL manages mixed transuranic waste, transuranic waste, mixed waste, low-level waste, hazardous waste, and nonhazardous wastes in accordance with the requirements of a number of Federal and State regulations, permits obtained under these regulations (for example, New Mexico unilateral FCC order) and DOE Orders. These requirements are primarily under the authority of the EPA, DOE, and the New Mexico Environment Department. SNL generated an estimated 90 cubic meters (m^3) (110 cubic yards [yd^3]) of low-level waste and an estimated 1.7 m^3 (2 yd^3) of mixed waste in 1994. In addition, SNL currently stores approximately 70 m^3 (90 yd^3) of mixed waste onsite (DOE 1995cc:6-4; DOE 1993j:3-71). The new Radioactive Mixed Waste Management Facility for handling these wastes is due to become operational in the near future. SNL generated 751 m^3 (198,450 gal) of liquid and 127 m^3 (166 yd^3) of solid hazardous waste in 1991 (DOE 1993j:3-71). The pit storage operations would generate less than 1 m^3 (1.3 yd^3) of mixed, low-level, and hazardous wastes. Compared to the amounts of waste generated and stored at SNL, the wastes generated by the pit storage activities would be minimal and would not impact the current waste management at SNL.

P.2.11 INTRASITE TRANSPORTATION

Interstate 40 and Interstate 25 provide access to the Albuquerque metropolitan area. Access to KAFB from Interstate 40 is provided from either the Wyoming or Eubank gate entrances (Figure P.1–1). Access to KAFB from Interstate 25 is via Gibson Boulevard.

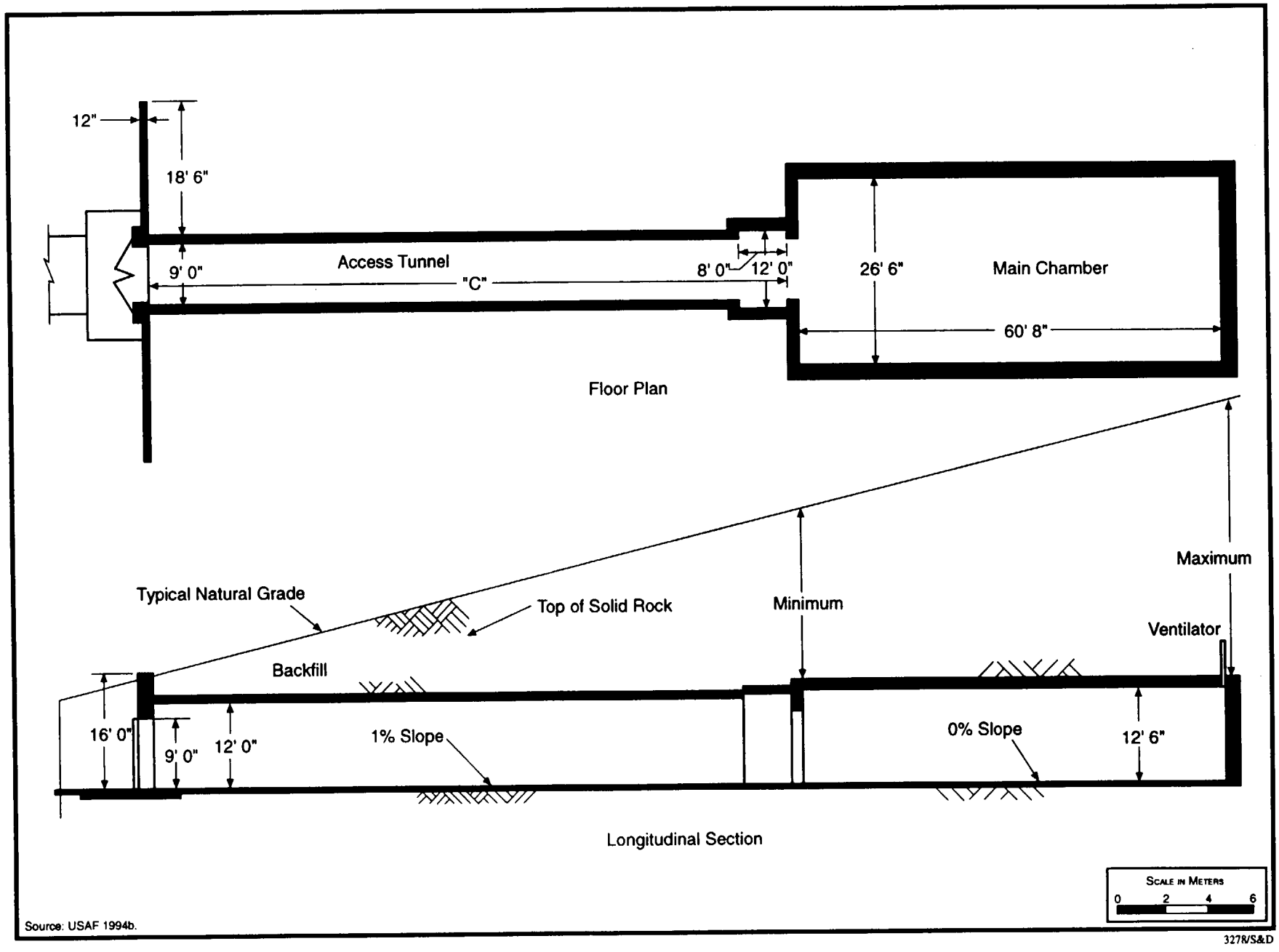


Figure P.2.9.3–1. Type D Storage Facility at the Manzano Weapons Storage Area.

The onsite road system at KAFB consists of paved streets and access roads. The Manzano WSA is located on the east side of KAFB. Access to the Manzano WSA is via Pennsylvania Avenue. The Manzano WSA is surrounded by fencing. Access to facilities within the area is provided via a ring road that encircles the mountain (Figure P.1–2). Traffic within the KAFB boundaries is strictly controlled, and the roads are not open to public traffic. Base personnel traffic would be controlled as SST convoys pass through the base roads. Because a release of Pu from an intersite pit shipment would require a severe accident (for example, an accident with a fuel tanker or a train), the controlled transportation environment at KAFB does not pose a significant threat to pit shipments. Consequently, the contribution to overall intersite transportation accident risk from onsite transport is negligible.

Two high speed transportation corridors (Gibson and Tijeras Arroyo corridors) that would traverse KAFB have been proposed. Of these, the Tijeras Arroyo Corridor would come in closest proximity to the Manzano WSA. Both transportation routes have been discussed for a number of years. However, *National Environmental Policy Act* documentation has not been completed on either project.

P.2.12 ENVIRONMENTAL JUSTICE

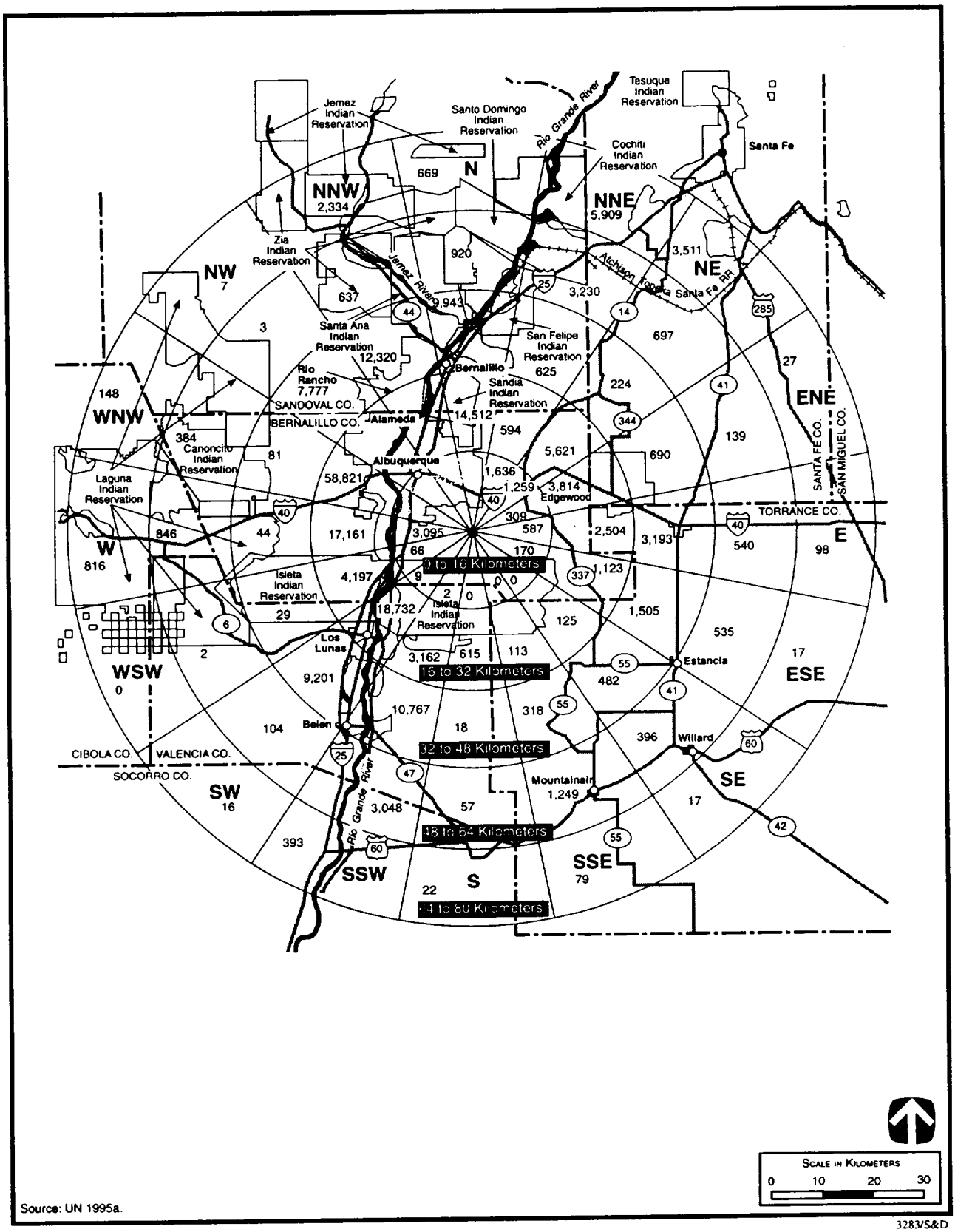
P.2.12.1 Affected Environment

The Manzano WSA is located on KAFB, which is adjacent to the southeastern city limits of Albuquerque, in central New Mexico. Besides the Air Force and other Department of Defense facilities, KAFB is also the location of various DOE operations, including SNL. Nearly 20,000 military and civilian personnel work on the base (KAFB 1995a:14). In order to identify the populations covered by Executive Order 12898, an 80-km (50-mi) radius circle centered on the Manzano WSA was overlaid on 1990 Census tract maps. The communities that lie within the 80-km (50-mi) ROI are shown in Figure P.2.12.1–1.

According to the 1990 Census, there were 606,446 persons within the Manzano ROI. White persons comprised 55 percent of the population, Hispanics were the second largest group with 37 percent, and Native Americans accounted for just over 4 percent of the total population. Native American reservations and trust lands belonging to 10 Native American tribes are located within the Manzano ROI, and approximately half of the Native Americans counted in the area in 1990 resided on Native American land. Blacks, Asians, Pacific Islanders and other racial groups totaled less than 4 percent of the total population in 1990 (UN 1995a).

Most of the population in the Manzano ROI resides in various cities, towns, and Census Designated Places. Albuquerque is the most populous community, with 384,736 persons or 63 percent of the total population within the Manzano ROI in 1990. An unincorporated area known as the South Valley, located immediately southwest of Albuquerque and due west of KAFB, is the second largest community in the area, with a 1990 population of 35,701. More than 70 percent of residents in the South Valley were Hispanic. Rio Rancho, northwest of Albuquerque in south-central Sandoval County, is third largest, with 32,505 persons in 1990. North and south of Albuquerque, along the Rio Grande River, are a number of towns and villages, most with primarily Hispanic populations: Belen (5,960 persons in 1990, 67 percent Hispanic), Bernalillo (5,960 persons, 75 percent Hispanic), Bosque Farms (3,791 persons, 25 percent Hispanic), Corrales (5,453 persons, 27 percent Hispanic), Los Chaves (3,872 persons, 49 percent Hispanic), Los Lunas (6,013 persons, 58 percent Hispanic), Tome-Adelino (1,695 persons, 65 percent Hispanic), and Valencia (3,917 persons, 47 percent Hispanic) (Census 1992b:11-21). Most of these communities are also characterized by fairly large low-income populations. For example, Belen had 28 percent of its population below the poverty level, Bernalillo had 24 percent below the poverty level, Los Chaves had 19 percent below the poverty level, Los Lunas had 25 percent below the poverty level, and Valencia had 15 percent below the poverty level (Census 1993m:516-520).

There are also nine primarily Native American communities in the Manzano ROI. A major portion of the northern boundary of the Isleta Indian Reservation borders the southern boundary of KAFB, but the Isleta people (2,699 in 1990) primarily live near the Rio Grande River, several miles from the KAFB boundary. In the



Source: UN 1995a.

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Figure P.2.12.1–1. The Manzano Weapons Storage Area Region of Influence.

Sandoval County portion of the Manzano ROI are seven additional Native American reservations with persons residing in dense settlements known as Pueblos: Sandia Pueblo with 358 Native American residents in 1990; Santa Ana Pueblo with 481 Native American residents; San Felipe Pueblo with 1,859 Native American residents; Santo Domingo Pueblo with 2,947 Native American residents; Cochiti Pueblo with 666 Native American residents; Zia Pueblo with 637 Native American residents; and Jemez Pueblo with 1,738 Native American residents. In the northwest corner of Bernalillo County is the Canoncito Navajo Reservation, a satellite of the main Navajo Reservation, with 1,060 Native American residents counted in 1990 (Census 1991h:60,61). The most notable socioeconomic characteristic of these communities is their large numbers of low-income persons. The percentage of persons below the poverty level based on 1989 incomes found on these reservations were: Isleta, 27 percent; Sandia, 19 percent; Santa Ana, 13 percent; San Felipe, 42 percent; Santo Domingo, 34 percent; Cochiti, 25 percent; Zia, 33 percent; Jemez, 37 percent; and Canoncito, 60 percent (Census 1993m:622-625).

Figure P.2.12.1–2 shows 1990 Census tracts within the Manzano ROI. The tracts are shaded if minority populations comprised 25 percent or more of the populations in 1990 or if 25 percent or more of the persons in a tract were below the poverty level based on their incomes in 1989. The 25 percent threshold levels for minority or low-income persons are based on the working definitions contained in the notice of the EPA's Office of Environmental Justice (59 FR 50757).

Virtually every tract in the Manzano ROI had a population in 1990 in which at least 25 percent of persons were minority or non-Whites. The major exceptions were the southernmost tract in Santa Fe County, 4 tracts in Rio Rancho in southcentral Sandoval County, and 25 tracts located primarily in the northeastern quadrant of Albuquerque, including the Four-Hills Tract located just north of the Manzano WSA.

Low-income persons were not nearly as prevalent in the Manzano ROI in 1990 as were minority persons. High levels of poverty found in Native American communities account for the shaded tracts in rural Sandoval County, eastern Cibola County, and western and southern Bernalillo County. The tracts shaded for low-income persons in rural Socorro, Valencia, Torrance and San Miguel Counties are also areas with largely Hispanic populations. In the Albuquerque area, high poverty levels were found primarily in the southern half of the city, with the greatest concentration of low-income persons situated in the southwest quadrant, in the unincorporated area known as the South Valley, with its 73-percent Hispanic population (Census 1992b:11-21).

P.2.12.2 Environmental Impacts

Because the long-term storage of pits at KAFB would not require any construction activities and because all facility modifications would take place inside existing facilities, impacts to the natural environment would be minimal. Under normal operating conditions, a minor increase in PM_{10} concentrations would be expected from the operation of forklifts that are used to move the pits from the unloading area to the storage area. These impacts are not likely to affect the surrounding population. Radiological releases from normal pit storage operations would have no measurable effect on an individual occupying a position near the KAFB boundary for an entire year. Levels at the site boundary would be indistinguishable from natural background radiation. No health effects would be expected among the general public, including minority and low-income populations, as a result of normal storage operations.

An abnormal event, such as accidental puncture of a storage container by a forklift, has the potential of exposing the general public to radiation. The analysis in Section P.2.9, indicates that the risk to the public from such an accident would be negligible. With no measurable impacts on the general population, the minority and low-income populations would not be disproportionately impacted.

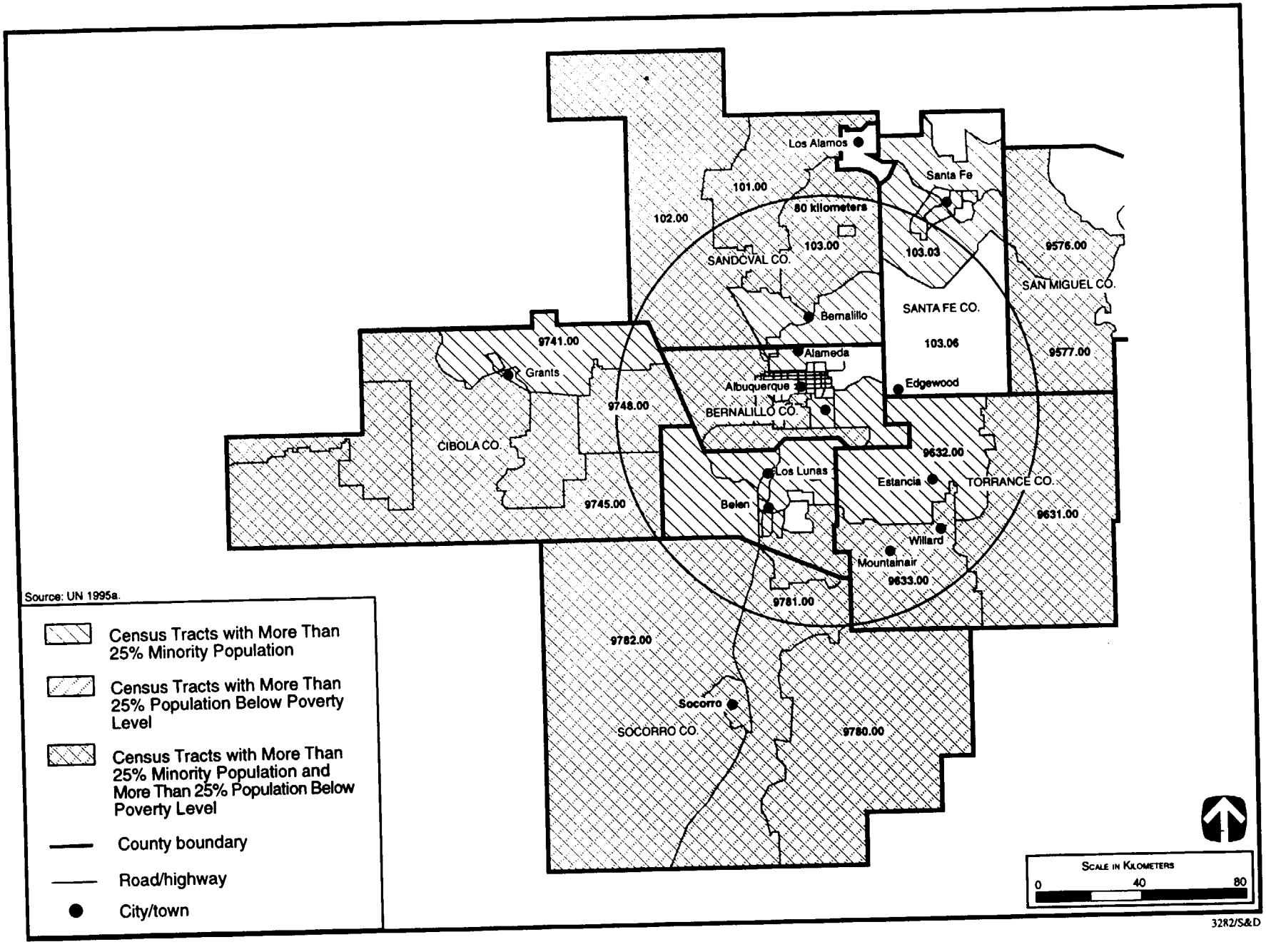

Figure P.2.12.1–2. Minority and Low-Income Populations in the Manzano Weapons Storage Area Region of Influence.

Appendix Q
Storage of Rocky Flats Environmental Technology Site Plutonium Pits at Pantex Plant

Q.1 INTRODUCTION

The Preferred Alternative for storage identified in this Programmatic Environmental Impact Statement (PEIS) calls for the transfer of plutonium (Pu) pits from Rocky Flats Environmental Technology Site (RFETS) (both strategic and surplus) to Pantex Plant (Pantex) with storage of surplus pits continuing until disposition. Pits to be transferred would be packaged in FL (Type B) containers at RFETS before shipment and, upon receipt at Pantex, would be repackaged into AL-R8 containers in Zone 12 South and placed into storage in Zone 4 West pending availability of AT-400A containers and relocation to upgraded storage facilities in Zone 12 South. The transportation of pits between Zone 4 and Zone 12 and the repackaging of the pits from AL-R8 to AT-400A containers is analyzed in the *Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapon Components* (Pantex EIS). The environmental analysis of intersite transportation for shipment of the RFETS Pu to Pantex is given in Section 4.4 and Appendix G of this PEIS for both workers and the public. Storage of Pantex pits at Zone 4 West is analyzed in the Pantex EIS and incorporated into this appendix as follows.

There are a small number of pits at RFETS that are surplus to national security needs but are still needed for on-going, non-weapons-related research and development projects at Los Alamos National Laboratory (LANL) and Lawrence Livermore National Laboratory. Therefore, these pits will not come within the scope of this PEIS until the research and development projects are completed. It is expected that this work will result in the conversion of the Pu into metal or oxide and the return of the material to RFETS. At that point, the materials will come within the scope of this PEIS and be stored and dispositioned in accordance with the decisions reached on the storage and disposition of surplus Pu in metal or oxide form.

The majority of the surplus pits at RFETS are already in a shippable form. However, there are a small number of pits that are not currently in a shippable condition. If a decision is reached to store surplus pits to Pantex, actions will be taken at RFETS, in accordance with existing procedures and in accordance with decisions based on existing environmental analyses, to place them in a shippable condition prior to shipment to Pantex. All of these pits are types which are currently stored at Pantex or have been stored there in the past.

Q.2 PIT TRANSFER AND STORAGE

Currently at Pantex, pits removed from weapons are swipe-tested to ensure that there is no surface radioactive contamination. Pits are then placed in AL-R8 containers and sealed with a tamper-indicating device. A pit within a container presents an external radiological hazard that is weapons-system-specific. A typical dose rate is 3 millirem per hour at 1 meter (m) (3.3 feet [ft]) from the AL-R8 container.

All pits at Pantex are expected to be stored in Zone 4 West using Stage Right techniques and equipment by December 1996. Stage Right techniques and equipment would enable the stacking of pit containers in a safe configuration, simplify pit transfers, reduce the need for entrance into magazines by personnel, and limit their exposure to radiation. Pits being transferred to Zone 4 West for staging are placed at the disassembly point within a Stage Right pallet (either 4 or 6 containers per pallet) and, using an electric forklift, loaded into a pallet trailer or a hardened trailer, which can carry 24 pit containers. The pallet trailer is driven to Zone 4 West, and the pit pallets are then unloaded by the shielded Stage Right forklift into the storage magazine. Pits are retrieved from a storage magazine in a similar manner.

Q.3 ZONE 4 WEST STAGING OPERATIONS

The bulk of the storage activity associated with the Zone 4 West staging magazines involves material movements to and from the production areas in Zone 12, as discussed in Section 4.12.1.1 of the Pantex EIS.

The Department requires safeguards and security programs at facilities handling special nuclear material. At Pantex, this program requires the periodic inventory of the magazines containing nuclear explosives and nuclear explosive components. The frequency of inventory for pit-staging magazines is dictated by the staging class of the magazine. Pit-staging magazines are divided into two classes: normal and exempt magazines. Normal staging magazines contain reserve components for existing weapon systems. For these magazines, the maximum time between inventories would be 12 months and 1 week. Exempt magazines contain components that are not expected to be reused. For these magazines, the maximum time between inventories would be 21 months and 1 week.

When pit containers are placed in Stage Right configuration, inventories in the magazines are performed with automatic bar code readers and video cameras attached to an inventory pallet. No movement of the pit containers is required for the inventory.

Q.4 RADIOLOGICAL EXPOSURE FROM ZONE 4 STORAGE

Pits from both dismantlement and stockpile management activities are transported and handled within the Pantex boundary as part of the Pantex operations. The transfer of pits between Zone 4 and Zone 12 would require repackaging of pits from original containers (for example, the FL [Type B] containers for pits from the RFETS) into AL-R8 containers for storage in Zone 4. After the AT-400A containers are available, the pits would be repackaged into AT-400A containers for either long-term storage or transportation to a disposition site. Pantex's current schedule indicates a repackaging rate of approximately 2,000 pits /year(yr) starting in 1997. No public exposure to radiation would occur from onsite, incident-free transport of pits.

Under the Proposed Action described in the Pantex EIS, as many as 20,000 pits could be stored in Zone 4. From the repackaging of 2,000 pits per year, it is estimated that an additional worker exposure of less than 30 person-roentgen equivalent man (rem) will be incurred. Similarly, an additional worker exposure of less than 300 person-rem for the repackaging of 20,000 pits will be incurred. Using a normal operations dose-to-risk conversion factor of 4×10^{-4} cancer fatalities per rem, less than 0.12 cancer fatalities would be incurred in the workforce from repackaging 20,000 pits. Repackaging all of the RFETS pits would yield less than an additional 10^{-2} cancer fatalities.

Table Q.4–1 presents the estimated exposures to the 50 people (based on current operation levels) who are directly involved with transportation and staging operations at Pantex. Workers who are not directly involved are not allowed in the vicinity of material transfer operations. These exposures were estimated using historical dosimetry information on site. No public exposure to radiation occurs from nonincident onsite material transfers. Assuming a 2,000 weapons operations maximum activity level, and assuming that the same 50 people remain involved in material handling for the 10 years under evaluation of the Pantex EIS, there should be 0.024 additional latent cancer fatalities in this group due to this exposure.

Assuming that a maximum exposed worker receives less than 0.3 rem/yr over the timeframe evaluated in the Pantex EIS, the incremental increase in lifetime fatal cancer probability from the projected exposure period of 10 yr is approximately 1.2×10^{-3}.

Although the worker exposure will vary depending on the weapons-system-specific pit, the dose to workers from the movement of a single pit container between Zone 4 and Zone 12 or Zone 12 and Zone 4 is 6.5×10^{-4} person-rem, as analyzed in the Pantex EIS. Transferring all of the RFETS pit containers from storage in Zone 4 to Zone 12 for repackaging and returning them to Zone 4 for storage would yield less than 10^{-3} cancer fatalities.

Table Q.4–1. Estimated Transportation and Staging Worker Exposures for the Proposed Action

Number of Additional Weapons Operations	10-Year Worker Exposure[a] (person-rem)	Yearly Average Exposure[a] (person-rem)	Latent Cancer Risk For 10-Year Exposure[a]
2,000	61	6	0.024
1,000	48	5	0.019
500	41	4	0.016

[a] Includes a baseline of 2,000 Pantex pit repackaging operations per year.
Source: PX DOE 1996b.

Therefore, the total exposure to workers from both the transfer of pit containers and the repackaging of pits from FL (Type B) to AL-R8 containers would be less than 10^{-2} additional latent cancer fatalities to workers due to this exposure. No additional exposure to the public from these activities for normal operations would occur.

Potential consequences to the Ogallala Aquifer from an accidental plutonium release were investigated in conjunction with a Safety Analysis Report and an Environmental Assessment by LANL (LANL 1992e:1,2,10). The hypothetical accident leading to dispersal of Pu to the environment around Pantex was assumed to be a high-temperature fire caused by a jet plane impact into a Zone 4 storage magazine containing Pu pits, and subsequent ignition of jet fuel. LANL envisioned that the hypothetical jet fuel fire could disperse fine particulate plutonium downwind of Pantex for a maximum distance of 80 kilometers (50 miles). Prompt decontamination efforts could reduce radiation levels to 0.2 microcuries per square meter ($\mu Ci/m^2$), but surface runoff and wind transport could concentrate contamination at playa lakes, where surface soil radiation levels could be as high as 2.0 $\mu Ci/m^2$. Surface water infiltrating through this contaminated soil could carry plutonium and decay products down toward the Ogallala Aquifer. The model assumed an average recharge rate of 3 centimeters (1 inch)/yr (10 times the High Plains average), and that recharge water would reach the Ogallala Aquifer at a depth of 20 to 100 m (65 to 330 ft).

The analyses conclude that the hypothetical plutonium dispersal accident does not pose a significant threat to the Ogallala Aquifer. The assumptions of the analyses were conservative because the "worst-case" scenarios were based on a depth to the water table of 20 m (50 ft) whereas, at Pantex, the typical depth to the top of perched groundwater is approximately 82 m (270 ft), and the depth to the main Ogallala Aquifer rangers from 104 to 140 m (340 to 460 ft). For water table depths of 60 and 100 m (200 and 330 ft), LANL calculated plutonium travel times of 305,000 and 610,000 years, respectively. Interactions with both surficial materials and the unsaturated portion of the Ogallala Formation would be expected to retard the movement of Pu relative to the infiltrating water (that is Pu would move at a rate slower than the infiltrating water). During the transport time, radioactive decay would be expected to further reduce Pu concentrations (LANL 1992e:10,12). Where the perched aquifer is present, the downward movement of plutonium would be further reduced, because the low-permeability fine-grained zone would impede downward flow and potential contamination would be more likely to move horizontally and follow the course of buried channel sands and gravels.

The likelihood of an aircraft crashing into a critical area at Pantex is extremely unlikely. The likelihood of an aircraft crashing into a critical area at Pantex and affecting (by fire or direct hit) an intransit pit shipment is considered not reasonably foreseeable (frequency of occurrence is less than 10^{-6} /yr) and at least several orders of magnitude less likely than an aircraft impact to a Pantex facility (because of the limited target area of the trailer compared with facilities and the limited time that weapons are contained in a trailer). In terms of risk the potential environmental impacts from an airplane crash into a Zone 4 weapons magazine are much more significant than those associated with a crash into a trailer.

Other potential onsite transportation accidents are much more likely to occur but have less potential for environmental impacts. A characteristic high-probability/low-consequence accident involving radioactive

material is a forklift puncture causing a release of plutonium from a pit container in the onsite transportation environment.

An individual worker would be expected to receive no more than a 7-rem exposure from Scenario 6 (PX DOE 1994e:ES-9). This corresponds to an upper bound incremental increase in fatal cancer probability of 2.8×10^{-3}, given that the release occurs. The maximally exposed non-involved worker would be expected to receive an exposure of 4×10^{-3} rem. This corresponds to an incremental increase in fatal cancer probability of 1.6×10^{-6}.

Q.5 OTHER IMPACTS FROM ZONE 4 STORAGE

The storage of RFETS pits planned under the Preferred Alternative would use existing storage space in Zone 4. Therefore, no land resources, including floodplains and wetlands, would be affected. Currently, all existing Modified Richmond and Steel Arch Construction magazines have the necessary utility support and material access control, and are supported by existing plant facilities and infrastructure. Steel Arch Construction magazines are used to stage nuclear weapons and pits, and require similar levels of infrastructure support as the Modified Richmond magazines. No new construction of storage magazines is required as a result of increasing storage for the RFETS pits. Therefore, current levels of infrastructure and utility support are expected to continue.

Only indirect pollutant emissions would result from pit storage activities. Pollutant exhaust emissions from moving pits between Zone 4 and Zone 12 would be the principal source. Onsite pit transfers are accomplished with electric forklifts which do not emit any pollutants. Pollutant emissions from these indirect sources are a small fraction of the total emissions from Pantex. Therefore, air quality impacts resulting from pit storage activities would be negligible. The principal noise source from pit storage activities would be from the vehicles moving pits between Zone 4 and Zone 12. The number of vehicles would be a very small fraction of the total traffic generated by Pantex. Overall noise level increases resulting from pit storage activities would not be detectable by the human ear. Therefore, noise impacts from pit storage activities would be negligible.

Under the Preferred Alternative, the RFETS number of pits would be stored at Pantex. The increased number of pits would be within the capacity of Zone 4. Storage activities do not pose additional impacts to water quality or availability currently for the operations occurring. Impacts to pit storage activities due to potential erosion, subsidence, and seismic hazards are the same or less than those for other operations at Pantex. Storage activities do not pose additional impacts to soil and sediment quality under normal conditions because the pits would not come in contact with the soil or sediment. Accident release scenarios are discussed in Section Q.4.

The storage of RFETS pits would utilize existing facilities and would not pose impacts to biotic or cultural resources. Only 30 employees are directly involved in current pit storage activity at Pantex. The addition of RFETS pits would use only a portion of these employees. Since they are part of the total employment of 3,800 workers at Pantex, the socioeconomic effects would be minimal.

Pit storage activities generate low-level waste, mixed low-level waste, hazardous waste, and nonhazardous waste. The amount of waste generated (less than 1 cubic meter [1.3 cubic yards] /yr) for each waste category is small compared to the volume of waste routinely handled at the plant, and would not affect waste management activities. The activities generating waste include radiation safety operations (for example, labels, security seals, and personnel protective equipment) and minor maintenance.

Appendix R
Aircraft Crash Accident at Pantex Plant

Aircraft accidents are a concern at Pantex Plant (Pantex) because of the large volume of local air traffic, the proximity of Pantex to flight paths to and from the Amarillo International Airport, and overflights. Pantex is a unique Department of Energy (DOE) facility because the location of nuclear weapons and weapon components, Zone 4 West, is situated on a direct line off the centerline of Runway 04/22. The airport is used by commercial air carriers and air taxis, large and small military aircraft flying touch and go training exercises, and general aviation aircraft. Even though the likelihood of an aircraft crash at Pantex is small, a crash could have the potential of causing radioactive releases from Pantex facilities.

In the past, the aircraft crash analysis was done using the Solomon model, which is over 20 years old. However, this method has been determined to be obsolete because it does not consider aircraft altitude in the model, does not incorporate recent data for crashes that occur near airports in the United States, and does not account for recent changes for aircraft that do not fly on designated airways. DOE has created a new set of models that replace the Solomon model. This DOE standard, *Accident Analysis for Aircraft Crash Into Hazardous Facilities*, was used to estimate the aircraft impacts in the *Environmental Impact Statement for the Continued Operation of the Pantex Plant and Associated Storage of Nuclear Weapon Components* (Pantex EIS).

The Pantex EIS analyzed the probability of hitting those nuclear facilities where plutonium (Pu), highly enriched uranium (HEU), and tritium are located. This analysis separated the 60 storage magazines in Zone 4 West from the nuclear facilities in Zone 12 South, where nuclear operations occur. The probability of an aircraft hitting a facility in Zone 4 or Zone 12 is 1.3×10^{-5} and 1.8×10^{-5}, respectively, for a total probability of 3.1×10^{-5}. The likelihood of an aircraft hitting a Zone 12 South nuclear facility is slightly greater than that of an aircraft hitting a Zone 4 West magazine, primarily because the total Zone 12 South target footprint is slightly larger than that of Zone 4 West.

After determining the probability of an aircraft crashing into a facility, the potential for sufficient building damage to cause a release was then determined. For an aircraft impact, building damage is defined as perforation (when a missile [flying object] generated by an aircraft penetrates into a facility) or scabbing (when an impact of an aircraft missile on a facility generates a secondary missile inside the facility). For magazines or building containing pits not in weapons, the only release mechanism possible is a perforation followed by a fire from spilled aircraft fuel. The Pantex EIS assumed that a perforation would lead to a release. This assumption is conservative for several reasons: 1) a fire may not occur, 2) the magazine contents may not be involved in the fire if the fuel material does not get into the facility, and 3) pits are currently stored in AL-R8 containers, which provide thermal and impact resistance. For Zone 4 and Zone 12 where Pu, HEU, and tritium are stored, the probability of facility perforation leading to a fire is 5.3×10^{-7} and 4.7×10^{-7}, respectively, for a total of 9.9×10^{-7}.

If any of the three storage alternatives (upgrade, consolidate, or collocate) of the *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic Impact Statement* (Storage and Disposition PEIS) were implemented, there would be a change in the aircraft crash probability. These alternatives would transport Pu material from other existing Pu storage sites, including Rocky Flats Environmental Technology Site (RFETS). Since the result of any of these alternatives would be the removal of all Pu pits not in weapons from Zone 4, aircraft crash and release probabilities would be reduced. If either the Preferred Alternative (Upgrade With RFETS Pu Pits Subalternative) or the Upgrade With All or Some RFETS Pu and Los Alamos National Laboratory Pu Subalternative is selected, all Pu would be moved to existing buildings in 12-66 and 12-82. This would reduce the aircraft crash and release probabilities almost proportionally to the number of Zone 4 West magazines no longer used. The aircraft crash and release probabilities in Zone 4 would only be for those magazines where nuclear weapons are staged. The impact of additional Pu in Zone 12 South buildings would

be minimal because Buildings 12-66 and 12-82 are existing and adjacent to where Pu is currently stored. Therefore, the aircraft crash probability would be approximately the same.

The Consolidation Alternative to the Storage and Disposition PEIS has two options at Pantex: 1) build a new facility and modify existing facilities in Zone 12 South or 2) build a new facility in Zone 12 South. Under the first option, there would be a reduction in the aircraft crash and release probabilities due to a reduction in target footprint in Zone 4 and a minimal increase in Zone 12 South as discussed for the Upgrade Alternative. The new facility in Zone 12 South would have a smaller target footprint compared to the Zone 4 West pit storage magazines no longer used. There would be an increase in the aircraft crash and release probabilities because of the addition of a new facility in Zone 12 South and a decrease because of the closing of some magazines in Zone 4 West. Because the overall Pantex footprint would decrease, the overall aircraft accident probabilities for all of Pantex would be reduced. The impacts from the second consolidation option, building a new facility, or the Collocation Alternative, would be similar to the first consolidation option since material would be moved from Zone 4 West to Zone 12 South.

Pantex is a potential site for locating two disposition facilities - pit disassembly/conversion and mixed oxide (MOX) fuel fabrication. Should new facilities be built at Pantex, there would be an increase in the aircraft crash and release probabilities depending on locations. If existing buildings where Pu operations are currently occurring were used, there would be no increase in the aircraft crash and release probabilities. The aircraft crash and release probabilities are highly dependent on the size of the building (target footprint) and the location of the building (whether or not it is shielded by other buildings). Based on the current size and location of the buildings, for the Preferred Alternative, if either the pit disassembly/conversion facility or MOX fuel fabrication facility were built at Pantex, either building would increase by no more than 10 percent of the current Zone 12 crash probability, 1.8×10^{-5}, or release probability, 4.7×10^{-7}. The crash and release probabilities for either building would be assessed in subsequent, tiered *National Environmental Policy Act* review.