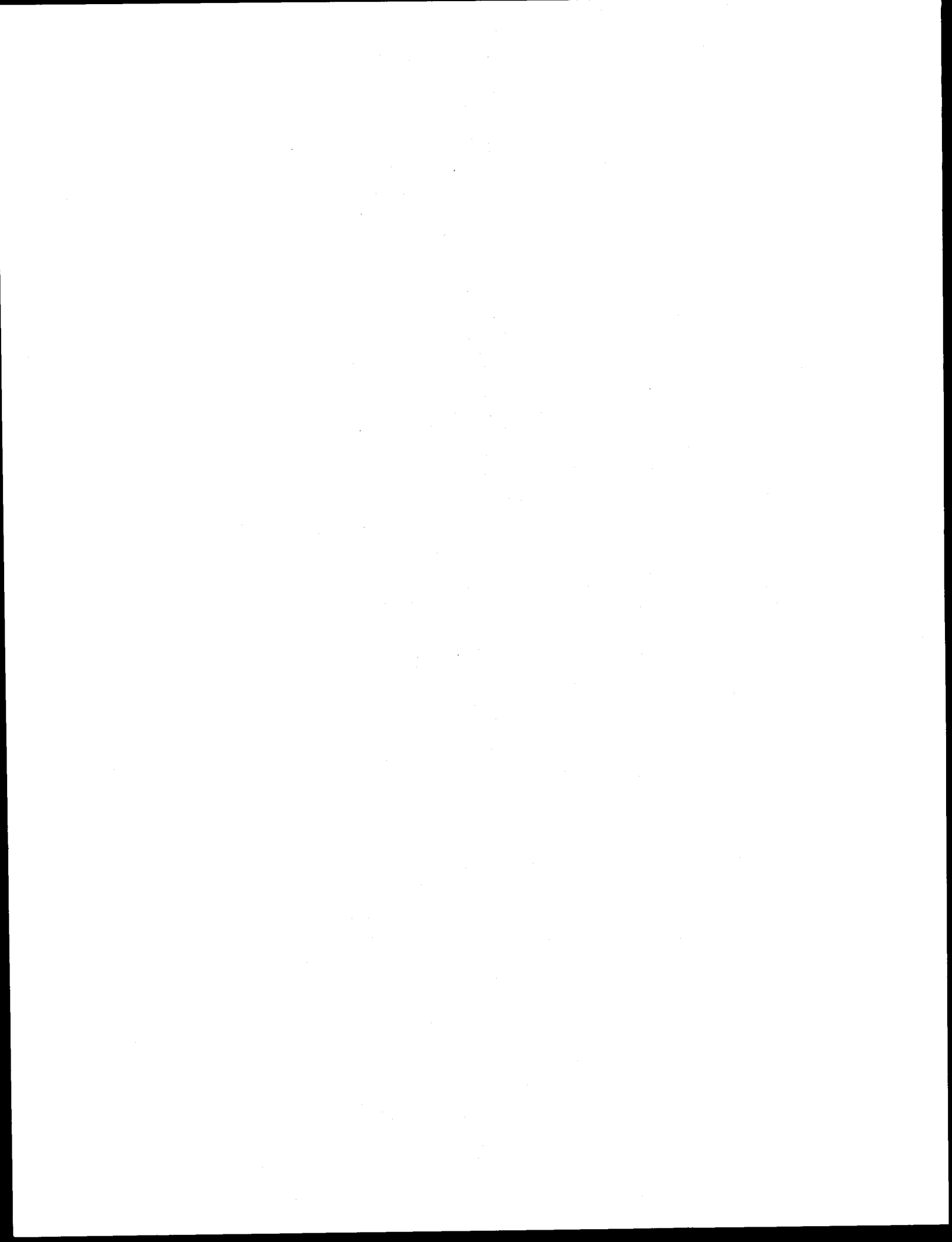


U.S. DEPARTMENT OF ENERGY
Office of New Production Reactors
Washington, DC 20585

**Draft
Environmental Impact Statement
for the Siting, Construction,
and Operation of
New Production Reactor Capacity**

Volume 4: Appendices D-R

April 1991



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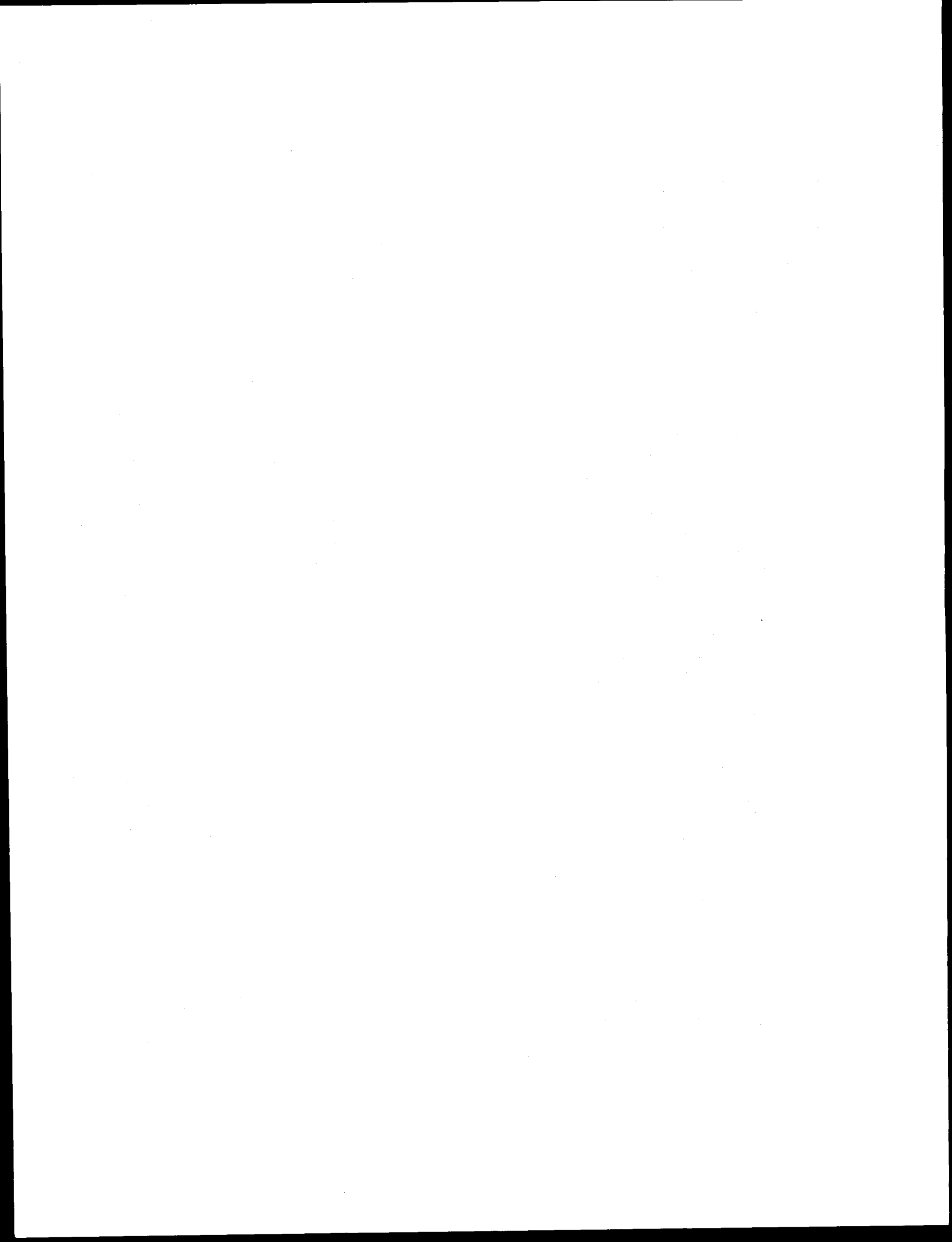
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APPENDIX D:
GEOLOGIC RESOURCES

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APPENDIX D: GEOLOGIC RESOURCES

Sections D.2, D.3, and D.4 of this appendix contain detailed descriptions of the geology of Hanford, INEL, and SRS. This material directly supports the summaries provided in Sections 4.1.1, 4.2.1, and 4.3.1. To aid in understanding the technical data presented, Section D.1 provides: (1) an excerpt from the Code of Federal Regulations that defines the term *capable* in the context of geologic faults; (2) an excerpt from the Code of Federal Regulations that defines the terms *prime* and *unique* in the context of farmlands; (3) the geologic time scale; and (4) the Modified Mercalli Seismic Intensity scale (Richter 1958), which describes the felt energy of earthquakes.

D.1 GENERAL DEFINITIONS

D.1.1 Definition of Capable Fault

The following definition and discussion of a *capable fault* is excerpted from the Code of Federal Regulations, Title 10, Part 100 (10 CFR 100), Appendix A:

- (g) A "capable fault" is a fault which has exhibited one or more of the following characteristics:
 - (1) Movement at or near the ground surface at least once within the past 35,000 years or movement of a recurring nature within the past 500,000 years.
 - (2) Macro-seismicity instrumentally determined with records of sufficient precision to demonstrate a direct relationship with the fault.
 - (3) A structural relationship to a capable fault according to characteristics (1) or (2) of this paragraph such that movement on one could be reasonably expected to be accompanied by movement on the other.

In some cases, the geologic evidence of past activity at or near the ground surface along a particular fault may be obscured at a particular site. This might occur, for example, at a site having a deep overburden. For these cases, evidence may exist elsewhere along the fault from which an evaluation of its characteristics in the vicinity of the site can be reasonably based. Such evidence shall be used in determining whether the fault is a capable fault within this definition.

Notwithstanding the foregoing paragraphs III(g) (1), (2) and (3), structural association of a fault with geologic structural features which are geologically old (at least pre-Quaternary) such as many of those found in

the Eastern region of the United States shall, in the absence of conflicting evidence, demonstrate that the fault is not a capable fault within this definition.

D.1.2 Definition of Prime and Unique Farmlands

The following definitions of *prime farmland* and *unique farmland* are excerpted from 7 CFR 657.5.

a Prime Farmland

- (1) General Criteria. Prime farmland is land that has the best combination of physical and chemical characteristics for producing food, feed, forage, fiber, and oilseed crops and is also available for these uses (the land could be cropland, pastureland, rangeland, forest land, or other land but not urban built-up land or water). It has the soil quality, growing season, and moisture supply needed to economically produce sustained high yields of crops when treated and managed, including water management, according to acceptable farming methods. In general, prime farmlands have an adequate and dependable water supply from precipitation or irrigation, a favorable temperature and growing seasons, acceptable acidity or alkalinity, acceptable salt and sodium content, and few or no rocks. They are permeable to water and air. Prime farmlands are not excessively erodible or saturated with water for long periods of time, and they either do not flood frequently or are protected from flooding. Examples of soils that qualify as prime farmland are Palouse silt loam, 0 to 7 percent slopes; Brookston silty clay loam, drained; and Tama silty clay loam, 0 to 5 percent slopes.
- (2) Specific Criteria. Prime farmlands must meet all the following criteria. Terms used in this section are defined in these USDA publications: "Soil Taxonomy, Agriculture Handbook 436," "Soil Survey Manual, Agriculture Handbook 18," "Rainfall-Erosion Losses From Cropland, Agriculture Handbook 282," "Wind Erosion Forces in the United States and Their Use in Predicting Soil Loss, Agriculture Handbook 346," and "Saline and Alkali Soils, Agriculture Handbook 60."

(a) The soils have:

- 1 Aquic, udic, ustic, or xeric moisture regimes and sufficient available water capacity within a depth of 40 inches, or in the root zone (root zone is the part of the soil that is penetrated by plant roots) if the root zone is less than 40 inches deep, to produce the commonly grown cultivated crops (cultivated crops include, but are not limited to, grain, forage, fiber, oilseed, sugar beets, sugarcane, vegetables, tobacco, orchard, vineyard, and

bush fruit crops) adapted to the region in 7 or more years out of 10; or

- 2 Xeric or ustic moisture regimes in which the available water capacity is limited, but the area has a developed irrigation water supply that is dependable (a dependable water supply is one in which enough water is available for irrigation in 8 out of 10 years for the crops commonly grown) and of adequate quality; or
 - 3 Acidic or torric moisture regimes, and the area has a developed water supply that is dependable and of adequate quality; and
- (b) The soils have a temperature regime that is frigid, mesic, thermic, or hyperthermic (pergelic and cryic regimes are excluded). These are soils that, at a depth of 20 inches, have a mean annual temperature higher than 32 degrees Fahrenheit. In addition, the mean summer temperature at this depth in soils with an O horizon is higher than 47 degrees Fahrenheit; in soils that have no O horizon, the mean summer temperature is higher than 59 degrees Fahrenheit; and
 - (c) The soils have a pH between 4.5 and 8.4 in all horizons within a depth of 40 inches or in the root zone if the root zone is less than 40 inches deep; and
 - (d) The soils either have no water table or have a water table that is maintained at a sufficient depth during the cropping season to allow cultivated crops common to the area to be grown; and
 - (e) The soils can be managed so that in all horizons within the depth of 40 inches or in the root zone if the root zone is less than 40 inches deep, during part of each year the conductivity of the saturation extract is less than 4 mmho/cm and the exchangeable sodium percentage is less than 15; and
 - (f) The soils are not flooded frequently during the growing season (less often than once in 2 years); and
 - (g) The product of K (erodibility factor) times the percent slope is less than 2.0, and the product of I (soils erodibility) times C (climatic factor) does not exceed 60; and
 - (h) The soils have a permeability rate of at least 0.06 inch per hour in the upper 20 inches, and the mean annual soil temperature at a depth of 20 inches is less than 50 degrees Fahrenheit; the permeability rate is not a limiting factor if the mean annual soil temperature is 59 degrees Fahrenheit or higher; and

- (i) Less than 10 percent of the surface layer (upper 6 inches) in these soils consists of rock fragments coarser than 3 inches.

b Unique Farmland

- (1) General Criteria. Unique farmland is land other than prime farmland that is used for the production of specific high-value food and fiber crops. It has the special combination of soil quality, location, growing season, and moisture supply needed to economically produce sustained high-quality and/or high yields of a specific crop when treated and managed according to acceptable farming methods. Examples of such crops are citrus, tree nuts, olives, cranberries, fruit, and vegetables.
- (2) Specific Characteristics. Unique farmland is used for a specific high-value food or fiber crop. It has a moisture supply that is adequate for the specific crop; the supply is from stored moisture, precipitation, or a developed irrigation system. It combines favorable factors of soil quality, growing season, temperature, humidity, air drainage, elevation, aspect, or other conditions, such as nearness to market, that favor the growth of a specific food or fiber crop.

- c Additional Farmland of Statewide Importance. This is land, in addition to prime and unique farmlands, that is of statewide importance for the production of food, feed, fiber, forage, and oilseed crops. Criteria for defining and delineating this land are to be determined by the appropriate State agency or agencies. Generally, additional farmlands of statewide importance include those that are nearly prime farmland and that economically produce high yields of crops when treated and managed according to acceptable farming methods. Some may produce as high a yield as prime farmlands if conditions are favorable. In some States, additional farmlands of statewide importance may include tracts of land that have been designated for agriculture by State law.

D.1.3 Definition of Major Geochronological and Chronostratigraphic Units

Figure D.1 depicts the components, relationships, and estimated age boundaries of major geochronologic and chronostratigraphic units.

D.1.4 Definition of Modified Mercalli Seismic Intensity Scale

Table D.1 defines the values of the Modified Mercalli Seismic Intensity scale.

Subdivisions Used by the U.S. Geological Survey					Age of Boundary (million years)	
					Estimate	Uncertainty Range*
Phanerozoic Eon	Cenozoic Era	Quaternary Period		Holocene Epoch	0.010	—
				Pleistocene Epoch		
		Tertiary Period	Neogene Subperiod	Pliocene Epoch	2	1.7-2.2
				Miocene Epoch	5	4.9-5.3
			Paleogene Subperiod	Oligocene Epoch	24	23-26
				Eocene Epoch	38	34-38
					55	54-56
				Paleocene Epoch	63	63-66
	Mesozoic Era	Cretaceous Period		Late Cretaceous Epoch or Upper Cretaceous Series	96	95-97
				Early Cretaceous Epoch or Lower Cretaceous Series	138	135-141
		Jurassic Period		205	200-215	
		Triassic Period		≈240	—	
		Paleozoic Era	Permian Period		290	290-305
			Carboniferous Periods	Pennsylvanian Period	≈330	—
				Mississippian Period	360	360-365
			Devonian Period		410	405-415
	Silurian Period		435	435-440		
	Ordovician Period		500	495-510		
	Cambrian Period		≈570 [†]	—		
	Proterozoic Eon	Proterozoic Z**			800	—
Proterozoic Y**			1,600	—		
Proterozoic X**			2,500	—		
Archean Eon						
				Oldest known U.S. rocks	3,600	

* Ranges reflect uncertainties of isotopic and biostratigraphic age assignments. Decay constants and isotope ratios employed are cited in Steiger and Jager (1977).

† Rocks older than 570 million years are also called Precambrian, a time term without specific rank.

** Time terms without specific rank.

FIGURE D.1 Major Geochronologic and Chronostratigraphic Units (Source: Modified from American Geological Institute 1982)

TABLE D.1 Modified Mercalli Seismic Intensity Scale

Value	Description of Effects
I	Not felt. Marginal and long-period effects of large earthquakes.
II	Felt by persons at rest, on upper floors, or favorably placed.
III	Felt indoors. Hanging objects swing. Vibration like passing of light trucks. Duration estimated. May not be recognized as an earthquake.
IV	Hanging objects swing. Vibration like passing of heavy trucks, or sensation of a jolt like a heavy ball striking the walls. Standing motor cars rock. Windows, dishes, doors rattle, glasses clink; crockery clashes; wooden walls and frame creak.
V	Felt outdoors; direction estimated. Sleepers wakened. Liquids disturbed, some spilled. Small unstable objects displaced or upset. Doors swing, close, open. Shutters, pictures move. Pendulum clocks stop, start, change rate.
VI	Felt by all. Many frightened and run outdoors. ^a People walk unsteadily. Windows, dishes, glassware broken. Knickknacks, books, etc., off shelves. Pictures off walls. Furniture moved or overturned. Weak plaster and masonry D (weak masonry) cracked. Small bells ring (church, school). Trees, bushes shaken visibly, or heard to rustle.
VII	Difficult to stand. Noticed by drivers of motor cars. Hanging objects quiver. Furniture broken. Damage to masonry D, including cracks. Weak chimneys broken at roof line. Fall of plaster, loose bricks, stones, tiles, cornices, unbraced parapets, and architectural ornaments. Some cracks in masonry C (ordinary masonry). Waves on ponds; water turbid with mud. ^a Small slides and caving in along sand or gravel banks. ^a Large bells ring. Concrete irrigation ditches damaged. ^a
VIII	Steering of motor cars affected. Damage to masonry C; partial collapse of masonry D. Some damage to masonry B (good masonry); none to masonry A (excellent masonry). Fall of stucco and some masonry walls. Twisting, fall of chimneys, factory smokestacks, monuments, towers, elevated tanks. Frame houses moved on foundations if not bolted down; loose panel walls thrown out. Decayed pilings broken off. ^a Branches broken from trees. ^a Changes in flow or temperature of springs and wells. ^a Cracks in wet ground and on steep slopes. ^a
IX	General panic. ^a Masonry D destroyed; masonry C heavily damaged, sometimes with complete collapse; masonry B seriously damaged. General damage to foundations. Frame structures shifted off foundations, if not bolted. Frames rocked. Serious damage to reservoirs. ^a Underground pipes broken. ^a Conspicuous cracks in ground. ^a In alluviated areas sand and mud ejected, earthquake mountains, sand craters. ^a
X	Most masonry and frame structures destroyed with their foundations. Some well-built wooden structures and bridges destroyed. ^a Serious damage to dams, dikes, embankments. ^a Large landslides. ^a Water thrown on banks of canals, rivers, lakes, etc. ^a Sand and mud shifted horizontally on beaches and flat lands. ^a Rails bent slightly. ^a
XI	Rails bent greatly. ^a Underground pipelines completely out of service. ^a

TABLE D.1 (Cont'd)

Value	Description of Effects
XII	Damage nearly total. Large rock masses displaced. ^a Lines of sight and level distorted. ^a Objects thrown into the air. ^a

^aThese criteria may be misleading as measures of the strength of shaking.

Source: American Geological Institute 1982.

D.2 HANFORD SITE

D.2.1 Geologic Setting

The Hanford Site lies within the Columbia Basin subprovince of the Columbia Intermontaine physiographic province. The Columbia Basin is bounded by the Cascade Mountains on the west, the northern Rocky Mountains on the north and east, and the Blue Mountains on the south. Hanford is located in the structural basin known as the Pasco Basin (Figure D.2). Basalt flows exceeding 3,050 m in cumulative thickness dominate the geology of the Pasco Basin (Figure D.3). This thick sequence of Miocene tholeiitic continental flood basalts is called the Columbia River Basalt Group. The lavas erupted from north-northwest-trending fissures in Oregon, Washington, and Idaho between about 17 and 6 million years ago (DOE 1988).

Late Miocene to Pliocene silts, sands, and gravels deposited by the ancestral Columbia River constitute the Ringold Formation, which overlies the basalt flows. A loess unit of Plio-Pleistocene age overlies the Ringold Formation in the western part of Hanford (Brown 1960). This silty fine sand to sandy silt consists of Ringold sediments reworked by the wind and redeposited. Relatively high caliche contents are found in much of this unit. The Hanford Formation overlies the Plio-Pleistocene unit. It consists of catastrophic flood sediments deposited when ice dams in western Montana and northern Idaho were breached and massive volumes of water flowed violently across eastern and central Washington (Bretz 1969). The floods scoured the land surface, locally eroding the Ringold Formation and the basalts and leaving a network of buried channels crossing the Pasco Basin (Tallman 1989). Thick sequences of sediments were deposited during several episodes of Pleistocene flooding; the last major flood occurred about 13,000 years ago (Myers et al. 1979). Occasional thin layers (0.3-10 cm) of air-fall tephra occur in the Pasco Basin.

The topography of the proposed NPR site is generally flat to gently undulating, with scattered low sand dunes. A relatively flat depositional terrace with scattered slight depressions occurs in the western half of the NPR site; this terrace is 15 m higher than the eastern half. Elevations vary from about 128 m above mean sea level (MSL) at the eastern end of the site to about 158 m MSL at the western end.

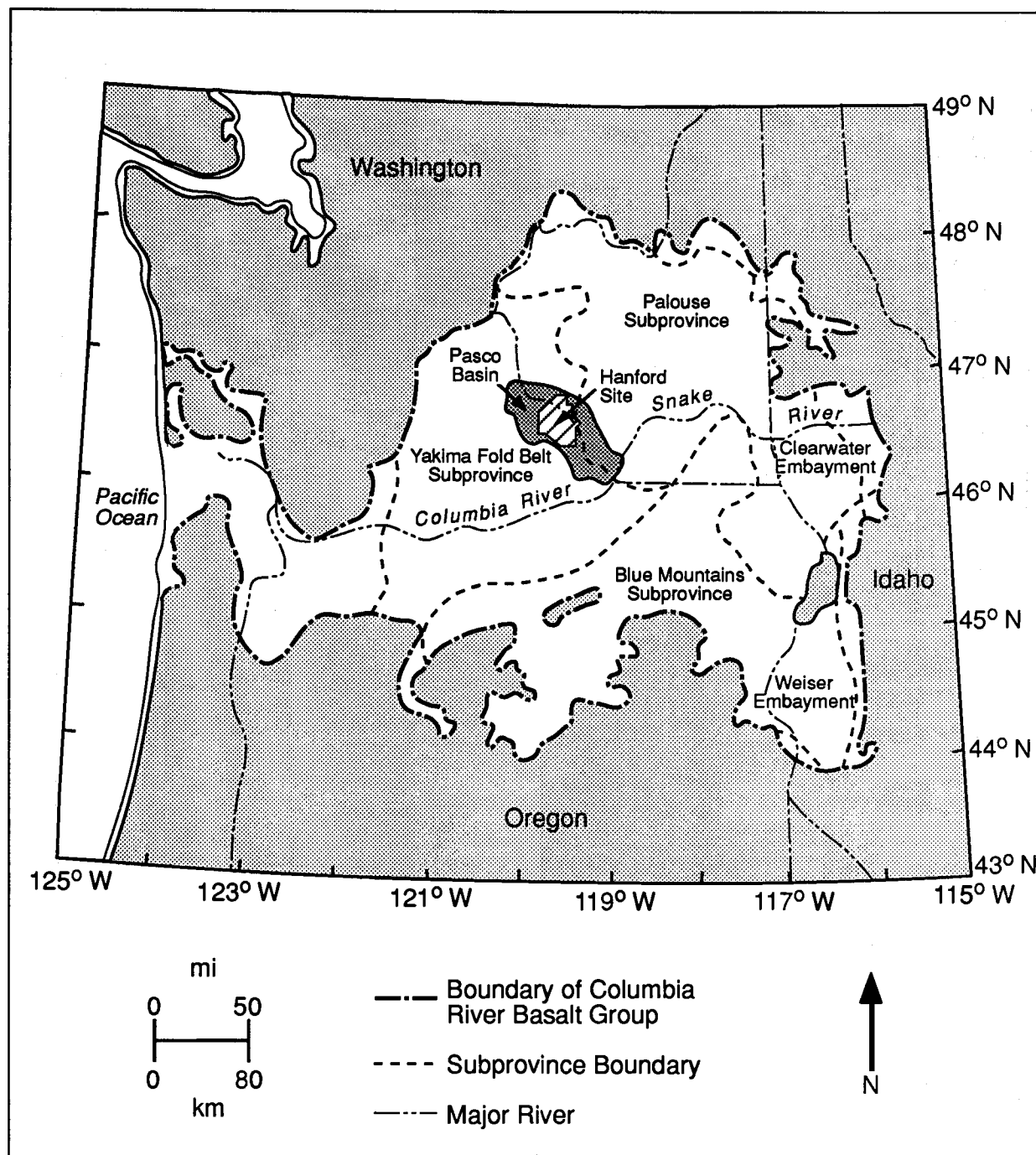


FIGURE D.2 Geologic Setting of the Pasco Basin and Neighboring Structural Subprovinces within the Region Defined by the Columbia River Basalt Group (Source: Modified from DOE 1988)

Quaternary		Period	Epoch	Group	Subgroup	Formation	Age (mil. yr)	Member or Sequence	Sediment Stratigraphy or Basalt Flows	
Pleistocene/Holocene	Pliocene							Surficial Units	Loess	
									Sand Dunes	
								Touchet Beds Pasco Gravels		
									Plio-Pleistocene Unit	
									Upper Ringold	
									Middle Ringold	
									Lower Ringold	
									Basal Ringold	
									Goose Island Flow	
									Martindale Flow	
									Basin City Flow	
									Levey Interbed	
									Upper Elephant Mountain Flow	
									Lower Elephant Mountain Flow	
									Rattlesnake Ridge Interbed	
									Upper Pomona Flow	
									Lower Pomona Flow	
									Selah Interbed	
									Upper Gable Mountain Flow	
									Gable Mountain Interbed	
									Lower Gable Mountain Flow	
									Cold Creek Interbed	
									Huntzinger Flow	
									Wahluke Flow	
									Sillusi Flow	
									Umatilla Flow	
									Mabton Interbed	
									Lolo Flow	
									Rosalia Flows	
									Quincy Interbed	
									Upper Roza Flow	
									Lower Roza Flow	
									Squaw Creek Interbed	
									Aphyric Flows	
									Phyric Flows	
									Vantage Interbed	
									Undifferentiated Flows	
									Rocky Coulee Flow	
									Unnamed Flow	
									Cohasset Flow	
									Undifferentiated Flows	
									McCoy Canyon Flow	
									Intermediate-Mg Flow	
									Low-Mg Flow Above Umtanum	
									Umtanum Flow	
									High-Mg Flows Below Umtanum	
									Very High-Mg Flow	
									At Least 30 Low-Mg Flows	

FIGURE D.3 Stratigraphic Units of the Pasco Basin (Source: Modified from DOE 1986)

D.2.2 Geologic Structures

All but the easternmost part of the Pasco Basin is within the Yakima Fold Belt structural subprovince (DOE 1988). The Yakima Fold Belt contains four major structural elements: (1) Yakima folds, (2) Cle Elum-Wallula disturbed zone, (3) Hog Ranch-Naneum Ridge anticline, and (4) northwest-trending wrench (predominantly strike-slip) faults.

The Yakima folds are a series of continuous, narrow, asymmetric anticlines separated by broad synclines. Thrust (high-angle reverse) faults are often found along both limbs of the anticlines, with the strike of the fault planes parallel or subparallel to the axes of the anticlines. Within the Hanford area, wrench faults with lengths of up to 3 km have been identified on Gable Mountain. The faults in Central Gable Mountain are considered "capable" by NRC criteria because they have displaced Hanford Formation gravels. (The criteria describing fault capability are given in 10 CFR 100, Appendix A; excerpts of the criteria are included in Section D.1.1).

The Cle Elum-Wallula disturbed zone is the central part of a larger topographic alignment called the Olympic-Wallowa lineament, which extends from the northwestern edge of the Olympic Mountains to the northern edge of the Wallowa Mountains in Oregon. The Cle Elum-Wallula disturbed zone is a narrow zone about 10 km wide that transects the Yakima Fold Belt and has been divided informally into three structural domains: (1) a broad zone of deflected or anomalous fold and fault trends, which extends south of Cle Elum, Washington, to Rattlesnake Mountain; (2) a narrow belt of aligned domes and doubly plunging anticlines ("The Rattles"), which extends from Rattlesnake Mountain to Wallula Gap; and (3) the Wallula Fault zone, which extends from Wallula Gap to the Blue Mountains. The Rattlesnake-Wallula alignment is a right-lateral strike-slip fault. This fault is interpreted as possibly being "capable," in part because of the lack of any distinct evidence to the contrary and because this structure continues along the northwest trend of faults that appear active at Wallula Gap (DOE 1986). Detailed mapping studies of the anticlinal ridges that bound the Pasco Basin identified some dextral movement on the Rattlesnake-Wallula alignment (DOE 1986).

The Hog Ranch-Naneum Ridge anticline is a broad structural arch that extends from southwest of Wenatchee, Washington, to at least Yakima Ridge. Northwest-trending wrench faults have been mapped west of 120°W longitude in the Columbia Plateau (DOE 1988). The mean strike of the dominant fault set (dextral wrench faults) is N40°W. A secondary fault set (sinistral wrench faults) strikes N13°E. These structures are not known to exist in the central Columbia Plateau.

D.2.3 Volcanism

Quaternary volcanism has been limited to the extreme western margin of the Columbia Basin subprovince and is associated with the Cascade Range. Air-fall tephra from at least three Cascadian volcanoes has blanketed the central Columbia Plateau since the Late Pleistocene; this tephra includes material from several eruptions of Mount St. Helens prior to the May 1980 eruption. Other volcanoes have erupted less frequently. Two closely spaced eruptions from Glacier Peak occurred about 11,200 years ago, and Mount Mazama erupted about 6,600 years ago. Generally, tephra layers do not exceed more than a few centimeters in thickness. However, up to 10 cm of tephra fell over eastern Washington when Mount Mazama erupted (DOE 1988).

Several major volcanoes are located west of Hanford in the Cascade Range. The nearest is Mount Adams, about 165 km west of Hanford, and the most active is Mount St. Helens, approximately 220 km west. A period of renewed volcanic activity at Mount St. Helens began in March 1980 and culminated in a major eruption on May 18, 1980. This eruption resulted in about 0.1 cm of ash fall over a 9-hour period at the Hanford Site, which was near the southern edge of the ash dispersal plume. Smaller eruptions of steam and ash occurred through October 1980, but none of these resulted in measurable amounts of ash being deposited at Hanford.

Ash fall from a large eruption is the dominant volcanic hazard at the Hanford Site (Tallman 1989). Volcanoes west of Hanford in the Cascade Range are the most likely source of eruptions in the region. A probabilistic risk assessment of volcanic hazards has not been done. However, the design-basis ash fall for the WNP-2 nuclear power plant is a 20-hour event that results in 3 compacted inches of accumulation (Tallman 1989). The major factors that determine the risk due to an ash fall are the (1) magnitude of the eruption, (2) distance from the eruption, and (3) prevailing wind speed and direction.

D.2.4 Seismicity

The historic record of earthquakes in the Pacific Northwest dates from about 1840. The early part of this record is based on felt reports of intensity and is probably incomplete because the region was sparsely populated. Seismic networks did not start providing locations and magnitudes of earthquakes in the Pacific Northwest until about 1960. A comprehensive network of seismic stations that provides accurate locations for most earthquakes greater than Richter magnitude 2.5 was installed in eastern Washington in 1969. The following discussion of seismicity at the Hanford Site is largely based on DOE (1988), a publication that includes a summary of the seismicity of the Pacific Northwest, a detailed review of the seismicity of the Columbia Plateau and the Hanford Site, and a description of the seismic networks used to collect the data.

Large earthquakes (greater than Richter magnitude 7) in the Pacific Northwest have occurred in the vicinity of Puget Sound in Washington and near the Rocky Mountains in eastern Idaho and western Montana. One of these events, which occurred near Vancouver Island in 1946, produced a maximum Modified Mercalli Intensity (MMI) of VIII and a magnitude of 7.3. Another large event, which occurred near Olympia, Washington, in 1949, had a maximum MMI of VIII and a magnitude of 7.1. The two largest events near the Rocky Mountains were the 1959 Hebgen Lake earthquake in western Montana, which had a magnitude of 7.5 and an MMI of X, and the 1983 Borah Peak earthquake in eastern Idaho, which had a magnitude of 7.3 and an MMI of IX.

A large earthquake occurred in north-central Washington in 1872. This event had an estimated maximum MMI that ranged from VIII to IX and an estimated magnitude of approximately 7. The distribution of intensities suggests a location within a broad region between Lake Chelan, Washington, and the British Columbia border (Malone and Bor 1979). This earthquake is estimated to have produced the largest ground-shaking effects (MMI V) ever observed at the Hanford Site.

Table D.2 lists all known earthquakes with an MMI of IV or larger and magnitudes of 4 or larger that occurred on the Columbia Plateau before 1969. Figure 4.1.1-2 shows the geographical distribution of these events and their relative magnitudes or intensities. Table D.3 lists all earthquakes with magnitudes of 3 or larger that occurred on the Columbia Plateau from March 23, 1969, to October 31, 1989. Figure 4.1.1-3 shows the geographical distribution of these events and their relative magnitudes or intensities. The largest earthquake in the Columbia Plateau was the Milton-Freewater, Oregon, earthquake, which occurred in 1936 and had a magnitude of 5.75 and a maximum MMI of VII. This earthquake was followed by a number of aftershocks that suggest a northeast-trending fault plane. The Milton-Freewater earthquake is estimated to have produced the second-largest ground-shaking effects (MMI IV) ever observed at the Hanford Site. Other earthquakes with magnitudes of 5 or larger and/or intensities of VI or larger occurred along the boundaries of the Columbia Plateau in a cluster near Lake Chelan that extends into the northern Cascade Range, in northern Idaho and Washington, and along the boundary between the western Columbia Plateau and the Cascade Range. Three MMI VI earthquakes occurred within the Columbia Plateau: one event occurred in the Milton-Freewater region in 1921; one near Yakima, Washington, in 1892; and one near Umatilla, Oregon, in 1893.

In the central portion of the Columbia Plateau, the largest historical earthquakes near the Hanford Site were two earthquakes that occurred north of the site in 1918 and 1973; both events had a magnitude of 4.4 and an MMI of V. Earthquakes in the central Columbia Plateau often occur in spatial and temporal clusters that are termed "earthquake swarms." The region north and east of the Hanford Site is a region of concentrated earthquake swarm activity, but earthquake swarms have also occurred in several locations within the Hanford Site. Earthquakes in a swarm tend to gradually increase and then subsequently decrease in frequency of events. In addition, no single large event within the sequence usually stands out. These earthquake swarms occur at shallow depths, with 75% of the events located at depths less than 4 km. Each earthquake swarm typically lasts several weeks to months, consists of several to 100 or more earthquakes, and is restricted to an area of 5-10 km in lateral extent. The longest dimension of the swarm area is often elongated in an east-west direction; however, detailed locations of swarm earthquakes indicate that the events occur on fault planes of variable orientation. The spatial pattern of seismicity in the central Columbia Plateau suggests an association of the shallow swarm activity with the east-west-oriented Saddle Mountains anticline.

Hypocenters in the central Columbia Plateau range downward to depths of about 30 km. Deeper earthquakes are less clustered and occur more often as isolated events. Seismic refraction surveys in the region show that the shallow earthquake swarms are occurring in the Columbia River basalts and that the deeper earthquakes are occurring in crustal layers below the basalts (Glover 1985; Catchings and Mooney 1988).

Earthquake focal mechanisms in the central Columbia Plateau generally indicate reverse faulting on east-west planes, which is consistent with a north-south-directed maximum compressive stress and with the formation of the east-west-oriented anticlinal folds of the Yakima Fold Belt (Rohay 1987). Earthquake focal mechanisms along the western margin of the Columbia Plateau also indicate north-south compression, but here the minimum compressive stress is oriented east-west, which results in strike-slip faulting (Rohay 1987). Geologic studies indicate an increased component of strike-slip faulting in the western portion of the Yakima Fold Belt. Earthquake focal mechanisms in the Milton-Freewater region to the southeast indicate a different stress field, one that has east-west- instead of north-south-oriented maximum compression.

TABLE D.2 Historical Earthquake Catalog for Eastern Washington and Areas near the Hanford Site (44.9-49.0N, 115.5-121.5W) through March 23, 1969^a (see Figure 4.1.1-2 for locations)

Date	Time (GMT)	Coordinates ^b			MMI ^c	Magnitude (M _L) ^c	Location
		Latitude (°N)	Longitude (°W)				
1866 Nov 24	18:10	45.595	121.173	IV	NA	The Dalles, Ore.	
1891 Sep 02	10:30 ^d	47.100	118.400	IV	NA	Ritzville, Wash.	
1891 Sep 17	04:30	44.940	121.033	IV	NA	Salem, Ore.	
1892 Feb 29	10:45	45.595	121.173	IV	NA	The Dalles, Ore.	
1892 Mar 05	d	46.600	120.500	VI	NA	North Yakima, Wash.	
1893 Mar 05	d	45.900	119.333	VI	NA	Umatilla, Ore.	
1897 Dec 15	d	47.800	120.000	V	NA	Lakeside, Wash.	
1905 Oct 18	23: ^d	47.800	120.000	V	NA	Chelan, Wash.	
1906 Jan 02	d	48.700	117.800	VI	NA	Stevens County, Wash.	
1906 Nov 02	01:49	48.500	117.900	V	NA	Colville, Wash.	
1907 Feb 18	12:20 ^d	47.842	120.023	V	NA	Chelan, Wash.	
1909 Jan 21	05: ^d	47.800	120.000	IV	NA	Chelan, Wash.	
1909 May 24	22: ^d	47.730	120.360	V	NA	Chelan-Leavenworth, Wash.	
1911 Jul 05	08:00	46.998	120.540	V	NA	Ellensburg, Wash.	
1913 Oct 14	23:00	45.700	117.100	V	NA	Seven Devils Region, Ore.	
1915 Mar 05	05:10	47.830	120.020	IV	NA	Lakeside, Wash.	
1915 Mar 05	05:30	47.830	120.020	IV	NA	Lakeside, Wash.	
1915 Jul 18	20:54	47.830	120.020	IV	NA	Lakeside, Wash.	
1915 Aug 18	14:05	48.500	121.400	V	NA	No town nearby	
1915 Dec 10	20:45	47.672	117.405	IV	NA	Spokane, Wash.	
1918 Feb 21	d	46.867	121.333	IV	NA	Bumping Lake, Wash.	
1918 Feb 28	23:15	46.500	120.500	V	NA	Yakima, Wash.	
1918 Mar 12	03:26	47.600	117.000	V	NA	Rathdrum, Idaho	
1918 Apr 18	21:13	47.600	117.400	IV	NA	White Bluffs Prairie, Wash.	
1918 Nov 01	17:20	46.858	119.375	V	NA	Corfu, Wash.	
1920 Oct 07	02: ^d	47.633	120.067	V	NA	Waterville, Wash.	
1920 Nov 28	11:30	45.700	121.500	IV	NA	Hood River, Ore.	
1921 Sep 14	11:00	46.100	118.250	VI	NA	Dixie-Walla Walla, Wash.	

TABLE D.2 (Cont'd)

Date	Time (GMT)	Coordinates ^b			MMI ^c	Magnitude (M _L) ^c	Location
		Latitude (°N)	Longitude (°W)				
1922 Jun 01	23:30	47.672	117.405		IV	NA	Spokane, Wash.
1924 Jan 06	13:09	46.070	118.328		IV	NA	Walla Walla, Wash.
1924 Jan 06	23:10	45.800	118.300		V	NA	Milton, Weston, Ore.
1924 May 27	00:19	46.070	118.328		IV	NA	Walla Walla, Wash.
1925 Nov 28	01:25	47.500	116.000		NR	4.30	No town nearby
1926 Apr 23	13:56	46.070	118.328		IV	NA	Walla Walla, Wash.
1926 Oct 17	02:45	45.730	121.483		V	NA	White Salmon, Wash.
1926 Nov 27	18:25	47.500	116.000		V	NA	Rathdrum, Idaho
1926 Dec 30	17:57	47.700	120.200		VI	NA	Chelan-E. Central, Wash.
1927 Jan 03	04:58	47.593	120.658		VI	NA	Leavenworth, Wash.
1930 Sep 03	13:00	47.300	117.800		V	NA	Lamont, Wash.
1931 Dec 08	14:25	47.830	120.020		IV	NA	Lakeside-Chelan Falls, Wash.
1933 May 31	20:20	47.842	120.013		IV	NA	Chelan, Wash.
1933 May 31	20:30	47.842	120.013		IV	NA	Chelan, Wash.
1934 Mar 09	16:00	47.830	120.020		IV	NA	Lakeside, Wash.
1934 Sep 18	24: ^d	46.998	120.540		V	NA	Ellensburg, Wash.
1934 Sep 22	11:30 ^d	46.998	120.540		IV	NA	Ellensburg, Wash.
1934 Sep 22	17:37 ^d	46.998	120.540		IV	NA	Ellensburg, Wash.
1934 Sep 26	16:15 ^d	46.998	120.540		V	NA	Ellensburg, Wash.
1934 Sep 26	16:45 ^d	46.998	120.540		V	NA	Ellensburg, Wash.
1934 Sep 26	21:15 ^d	46.998	120.540		V	NA	Ellensburg, Wash.
1934 Oct 04	02:26 ^d	46.998	120.540		IV	NA	Ellensburg, Wash.
1934 Oct 11	21:19 ^d	46.998	120.540		IV	NA	Ellensburg, Wash.
1934 Oct 19	23:31 ^d	46.998	120.540		V	NA	Ellensburg, Wash.
1934 Oct 29	18:36 ^d	46.998	120.540		V	NA	Ellensburg, Wash.
1934 Nov 01	07:28 ^d	46.998	120.540		IV	NA	Ellensburg, Wash.
1934 Nov 02	15:17 ^d	46.998	120.540		V	NA	Ellensburg, Wash.
1935 Jul 09	22:45	47.700	120.000		V	NA	Chelan Falls, Wash.
1935 Oct 12	01:03	47.662	120.223		V	NA	Entiat, Wash.

TABLE D.2 (Cont'd)

Date	Time (GMT)	Coordinates ^b		Magnitude M _L ^c	Location
		Latitude (°N)	Longitude (°W)		
1935 Nov 01	03:35	47.472	115.925	IV	Wallace, Idaho
1936 Jul 16	07:07	46.208	118.233	VII	Milton-Freewater, Ore.
1936 Jul 18	16:30	45.933	118.383	V	Milton-Freewater, Ore.
1936 Jul 30	11:20	45.935	118.338	IV	Freewater, Ore.
1936 Jul 30	12:00	45.935	118.338	IV	Freewater, Ore.
1936 Jul 30	12:20	46.070	118.328	IV	Walla Walla, Wash.
1936 Aug 04	09:19	45.800	118.600	V	Helix, Ore.
1936 Aug 28	04:39	45.933	118.383	V	Milton-Freewater, Ore.
1937 Feb 09	22:20	46.070	118.328	IV	Walla Walla, Wash.
1937 Jun 04	14:43	46.070	118.328	IV	Walla Walla, Wash.
1938 Aug 11	18:52	45.935	118.388	IV	Milton, Ore.
1938 Oct 27	23:10	45.935	118.388	IV	Milton, Ore.
1939 Jan 25	07:59	45.700	118.700	IV	Mission, Ore.
1940 Mar 24	03:04	46.000	121.200	IV	Mt. Rainier, Wash.
1941 Apr 07	09:25	48.300	119.600	VI	No town nearby
1941 Apr 12	17:40	47.648	120.069	IV	Waterville, Wash.
1942 Feb 23	14:03	47.600	120.200	V	Wenatchee-Chelan Falls, Wash.
1942 Jun 12	09:30	44.900	117.100	V	Halfway and Pine, Ore.
1942 Oct 14	11:30	48.310	120.652	V	Stehekin, Wash.
1942 Nov 01	18:50	48.000	116.700	VI	Sandpoint, Idaho
1943 Apr 24	00:10	47.300	120.600	VI	Leavenworth, Wash.
1943 Sep 22	21:50 ^d	47.967	119.000	IV	Grand Coulee, Wash.
1944 Sep 02	01:25	46.070	118.328	IV	Walla Walla, Wash.
1944 Sep 20	03:00	44.900	116.900	IV	Rockville, Ore.
1944 Oct 31	11:34	47.800	120.600	V	No town nearby
1944 Dec 25	13:12	47.662	120.223	IV	Entiat, Wash.
1945 Jan 04	02:34	47.662	120.223	V	Entiat, Wash.
1945 Feb 27	11:00	48.480	121.190	IV	Winthrop, Wash.

TABLE D.2 (Cont'd)

Date	Time (GMT)	Coordinates ^b		Magnitude M _L ^c	Location
		Latitude (°N)	Longitude (°W)		
1945 Mar 02	07:54	47.662	120.223	IV	Entiat, Wash.
1945 Sep 23	02:40	46.070	118.328	IV	Walla Walla, Wash.
1946 Feb 05	16:12	47.800	120.200	IV	Chelan-Ardenvoir, Wash.
1946 Feb 06	03:20	48.527	121.430	IV	Marblemount, Wash.
1947 Dec 22	10:30	47.662	120.223	IV	Entiat, Wash.
1948 Jan 13	06:55	47.900	120.300	V	Lucerne-Waterville, Wash.
1948 Aug 28	22:25	47.957	117.485	IV	Deer Park, Wash.
1948 Oct 25	19:50	47.842	120.013	IV	Chelan, Wash.
1948 Dec 20	16:18	44.996	120.215	IV	Fossil, Ore.
1949 Mar 15	20:53	45.500	117.000	NR	No town nearby
1949 Oct 20	16:00	48.500	120.500	IV	Lost River, Wash.
1950 Mar 08	06:25	47.662	120.223	IV	Entiat, Wash.
1950 Jun 25	23:45	47.491	117.575	IV	Cheney, Wash.
1951 Jan 04	13:45	47.700	120.000	V	Chelan-Waterville, Wash.
1951 Jan 07	22:45	45.900	119.200	V	McNary, Ore.
1952 Mar 04	19:42	47.672	117.405	V	Spokane, Wash.
1952 Sep 09	09:30	48.698	116.315	IV	Bonniers Ferry, Idaho
1952 Sep 09	09:45	48.698	116.315	IV	Bonniers Ferry, Idaho
1953 Sep 09	09:30	48.698	116.315	IV	Bonniers Ferry, Idaho
1954 May 23	13:41	48.342	120.137	V	Twisp, Wash.
1954 Jun 08	00:16	47.500	116.000	V	Mortain, C. D'Alene, Idaho
1955 Feb 06	d	47.967	119.000	IV	Grand Coulee, Wash.
1955 May 31	23:35	47.680	116.773	IV	Coeur D'Alene, Idaho
1956 Feb 24	22:00	47.900	119.100	V	Electric City, Wash.
1957 Nov 01	10:12	46.700	121.500	V	No town nearby
1957 Dec 18	23:25	47.500	116.000	VI	No town nearby
1958 Apr 12	00:00	47.900	119.100	IV	Electric City, Wash.
1958 Apr 12	22:37	48.000	120.000	VI	No town nearby
1959 Jan 21	07:15	46.070	118.328	IV	Walla Walla, Wash.

TABLE D.2 (Cont'd)

Date	Time (GMT)	Coordinates ^b		Magnitude M _L ^c	MMI ^c	Location
		Latitude (°N)	Longitude (°W)			
1959 Jul 11	15: ^d	47.600	119.300	NA	IV	Deep Lake, Wash.
1959 Aug 06	03:44	47.800	119.900	4.40	VI	Chelan, Wash.
1959 Nov 09	21:10	45.353	119.550	NA	IV	Heppner, Wash.
1961 May 22	01:57	47.600	120.200	NA	IV	Entiat, Wash.
1961 Jun 28	10:22	47.537	120.293	NA	IV	Rocky Reach Dam, Wash.
1961 Oct 31	02:35	48.400	120.000	4.30	V	No town nearby
1961 Oct 31	03:34	48.400	120.000	NA	V	Okanogan, Wash.
1962 Jan 15	05:29	47.800	120.200	4.30	V	No town nearby
1963 Dec 22	02:54	48.590	119.760	4.40	V	No town nearby
1965 Apr 28	19:00 ^d	48.600	116.900	4.30	V	No town nearby
1965 Nov 07	16:41	44.900	117.000	4.30 ^e	NR	No town nearby
1966 Jul 23	01:57	47.200	119.500	4.30	NR	No town nearby
1966 Dec 30	03:51	44.900	117.000	4.20 ^e	NR	No town nearby

^aIncludes all earthquakes with a Modified Mercalli Intensity (MMI) of IV or larger and with local Richter magnitudes larger than 4.

^bThe coordinate locations of early events are often the location of maximum reported intensity; their precision is probably no better than 0.1°.

^cNR = not reported; M_L = local Richter magnitude; NA = data not available.

^dLocal time (i.e., not converted to equivalent Greenwich mean time [GMT]); generally indicates a greater uncertainty in time. Where no time is provided, the date indicates the best approximation of the time of occurrence from available information sources.

^eBody wave magnitude (mb [approximately equal to M_L]); local Richter magnitude (M_L) not available.

Source: Rohay 1989.

TABLE D.3 Instrumental Earthquake Catalog for the Columbia Plateau and Surrounding Area, 1969-1989^a (see Figure 4.1.1-3 for locations)

Date	Time (GMT)	Coordinates		Depth (km)	Magnitude (M _c) ^b
		Latitude (°N)	Longitude (°W)		
1969 Aug 02	03:04	47.346	117.396	39.55	3.1
1970 Jul 11	10:24	48.204	121.355	0.05 ^c	3.0
1970 Aug 23	11:11	46.746	119.348	2.96	3.4
1970 Sep 11	02:20	46.654	120.404	18.15 ^d	3.5
1970 Oct 01	11:46	46.756	119.378	4.52 ^c	3.0
1970 Oct 02	15:56	46.757	119.373	4.01	3.4
1970 Oct 19	07:15	46.886	117.569	15.53	3.2
1970 Nov 06	18:15	46.687	118.872	2.42	3.1
1970 Nov 29	02:15	47.412	121.421	14.94 ^e	3.3
1971 Jan 26	10:17	46.901	119.569	4.36	3.2
1971 May 22	15:57	47.790	118.953	35.08	3.3
1971 Jul 13	23:29	45.024	117.962	17.62	3.5
1971 Aug 18	23:44	47.649	120.146	13.23	3.2
1971 Oct 25	18:52	46.708	119.550	3.99	3.8
1971 Nov 23	02:12	48.166	121.427	24.81	4.1
1972 Jun 19	23:57	48.063	121.399	2.34	3.2
1973 Sep 04	17:56	48.233	121.227	4.87	3.7
1973 Dec 20	01:08	46.867	119.357	3.31	4.4
1974 Apr 20	03:00	46.716	121.476	5.55	4.7
1974 Apr 21	14:08	46.691	121.430	4.82	3.5
1975 Jun 15	17:51	46.234	119.113	1.74	3.1
1975 Jun 28	16:33	46.105	119.704	9.34	3.3
1975 Jun 28	22:17	46.099	119.706	10.97	3.8
1975 Jul 01	05:28	45.605	120.016	15.32	3.6
1975 Jul 07	20:41	45.951	118.234	8.04 ^d	3.2
1975 Sep 18	12:19	47.811	118.258	6.52	3.5
1976 Mar 29	12:36	45.122	120.890	15.00 ^c	3.0
1976 Apr 02	20:10	45.136	120.876	15.00 ^c	3.2 ^f
1976 Apr 06	17:56	45.155	120.802	15.00 ^c	3.2 ^f
1976 Apr 06	23:16	45.097	120.721	15.00 ^c	3.4 ^f
1976 Apr 08	10:15	45.155	120.802	15.00 ^c	3.8 ^f
1976 Apr 09	09:11	45.208	120.887	15.00 ^c	3.5 ^f
1976 Apr 13	00:02	45.180	121.007	15.00 ^c	3.3 ^f
1976 Apr 13	00:47	45.076	120.859	15.00 ^c	4.6
1976 Apr 13	01:20	45.121	120.894	15.00 ^c	3.4 ^f
1976 Apr 13	13:29	45.147	120.860	15.00 ^c	3.1 ^f
1976 Apr 17	02:11	45.159	120.847	15.00 ^c	4.0
1976 May 15	13:04	47.711	120.062	8.04	3.1
1976 Jun 15	01:01	46.465	117.766	0.02 ^c	3.0
1976 Jun 15	09:08	47.625	120.327	0.75	3.0
1976 Jul 23	17:59	46.085	118.750	0.02 ^c	3.1
1976 Aug 30	16:34	47.654	120.200	5.50	3.1
1976 Oct 10	05:41	45.270	120.500	15.00 ^c	3.6
1976 Nov 19	07:04	45.087	120.925	91.39	3.1

TABLE D.3 (Cont'd)

Date	Time (GMT)	Coordinates		Depth (km)	Magnitude (M _c) ^b
		Latitude (°N)	Longitude (°W)		
1976 Dec 13	08:47	47.651	120.130	5.61	3.1
1977 Jan 27	07:47	46.939	119.591	1.47	3.2
1977 Mar 11	22:50	45.899	119.666	0.02 ^c	3.1
1977 Jul 13	07:15	47.058	120.986	4.07	3.9
1978 Jan 25	01:09	47.897	120.112	0.56	3.3
1978 Mar 22	03:08	48.111	119.501	13.06	3.3
1978 Apr 16	19:45	47.737	120.239	4.75	3.3
1978 Jun 27	02:18	46.877	120.972	12.38	3.6
1978 Oct 10	12:04	47.899	119.691	4.48	3.1
1979 Jan 19	14:55	47.901	119.688	6.60	3.9
1979 Jan 21	20:40	47.899	119.687	4.86	3.0
1979 Jan 30	16:06	47.666	120.126	7.19	3.0
1979 Feb 17	08:36	46.164	119.933	18.11	3.6
1979 Apr 07	03:47	46.979	120.451	16.89	3.0
1979 Apr 08	07:29	45.991	118.399	7.88	4.3
1979 Jul 28	02:19	46.675	120.614	0.04 ^c	3.7
1979 Nov 10	04:53	47.724	120.069	4.22	3.1
1979 Nov 24	11:51	46.930	119.577	2.80	3.4
1979 Dec 10	05:40	46.665	120.601	7.54	3.1
1980 Nov 19	21:35	46.950	119.470	0.86	3.3
1981 Feb 02	01:23	46.263	120.989	1.98 ^c	4.0
1981 Feb 18	06:09	47.197	120.893	3.37	4.2
1981 Mar 15	07:23	47.987	121.493	5.35	3.6
1981 May 28	08:55	46.530	121.399	2.98	4.6
1981 May 28	09:10	46.525	121.394	3.22	5.0
1981 Jun 14	13:12	45.962	120.507	13.58	3.2
1981 Jul 22	06:05	47.778	120.288	9.53	3.0
1981 Oct 25	03:20	47.759	120.196	7.58	3.2
1982 Jan 23	15:31	46.546	121.378	3.33	3.2
1982 Sep 26	10:09	46.867	121.048	3.25	3.4
1983 Mar 22	12:47	45.992	118.403	7.53	3.8
1983 Apr 25	15:48	48.630	119.567	8.03 ^c	3.0
1983 Oct 20	09:44	46.717	119.584	1.86	3.4
1983 Nov 14	11:18	46.655	120.600	7.87	3.8
1983 Dec 05	07:24	46.915	120.713	7.76 ^d	3.8
1984 Jan 31	05:29	45.496	116.673	5.65 ^d	4.0
1984 Apr 11	03:07	47.535	120.186	8.02	4.3
1984 Jun 18	19:34	45.231	118.688	10.08	3.1
1984 Aug 24	04:42	47.650	120.955	0.75 ^d	3.0
1984 Oct 10	03:24	47.904	119.079	15.39	3.0
1985 Jan 09	05:46	47.064	120.094	0.34	3.3
1985 Jan 25	07:28	46.500	120.632	16.63	3.1
1985 Jan 31	03:02	47.060	120.084	0.29	3.3
1985 Feb 10	20:29	45.705	119.635	18.41	3.9
1985 Mar 09	01:31	46.985	118.590	3.41	3.3

TABLE D.3 (Cont'd)

Date	Time (GMT)	Coordinates		Depth (km)	Magnitude (M _c) ^b
		Latitude (°N)	Longitude (°W)		
1985 Apr 19	10:52	46.897	120.284	5.35	3.2
1985 Jun 09	01:24	46.675	118.977	3.21	3.2
1985 Jun 17	07:00	47.058	120.077	0.28	3.0
1985 Jul 16	21:13	46.189	121.099	0.02 ^c	3.2
1985 Oct 01	05:25	46.796	120.048	1.09	3.0
1985 Oct 01	06:53	46.789	120.047	1.71	3.0
1985 Oct 10	10:06	47.749	120.266	7.04	3.2
1985 Nov 22	18:09	47.263	119.351	20.82	3.2
1986 Feb 04	01:58	46.044	118.810	7.80	3.2
1986 Apr 08	10:57	47.770	120.230	13.76	3.3
1986 Sep 01	21:32	46.719	119.285	14.09	3.4
1987 May 26	16:11	45.205	116.246	38.11	4.0
1987 Jun 11	19:50	56.778	120.694	17.23	3.0
1987 Sep 08	05:02	45.191	120.072	13.01	3.1
1987 Dec 02	07:12	46.675	120.684	18.20	4.1
1987 Dec 02	09:02	46.679	120.673	17.80	4.3
1988 Feb 06	12:51	47.666	120.024	7.15	3.0
1988 May 05	00:18	47.651	120.321	6.72	3.3
1988 May 28	09:02	46.811	119.428	0.02 ^c	3.5
1988 Jun 18	14:44	48.407	116.177	0.05 ^c	3.4
1988 Jul 09	01:17	46.842	119.710	3.81	3.7
1988 Jul 14	12:45	46.888	119.413	1.79	3.3
1988 Jul 30	16:13	47.650	120.074	0.02 ^c	3.2
1988 Sep 29	08:09	45.850	120.260	13.89	3.5
1989 Mar 27	20:17	45.816	120.262	12.25 ^c	3.1
1989 May 09	18:28	48.231	119.854	15.59	4.5

^aIncludes all earthquakes with coda-length magnitude 3 or larger from March 23, 1969, to October 31, 1989.

^bM_c = coda-length magnitude; approximately equals local Richter magnitude, M_L.

^cDepth fixed (depth less than 0.05 km or unstable within 0.2 km).

^dMaximum iterations (24) exceeded (last location used).

^eMaximum iterations (24) exceeded while depth fixed (last location used).

^fLocation and magnitude determined by Couch, Thrasher, and Keeling 1976.

Conversion factor: 1 km = 0.6214 mi.

Source: University of Washington On-Line Earthquake Catalog Data Base.

Estimates of the earthquake potential of structures and zones in the central Columbia Plateau were developed during the licensing of nuclear power plants at the Hanford Site. In its review of the operating license application for WNP-2, the U.S. Nuclear Regulatory Commission (NRC 1982) concluded that four earthquake sources should be considered for the purpose of seismic design: (1) the Rattlesnake-Wallula alignment, (2) Gable Mountain, (3) a floating earthquake in the tectonic province, and (4) earthquakes occurring in a swarm area.

For the Rattlesnake-Wallula alignment, which runs along the southwestern boundary of the Hanford Site, the NRC estimates a maximum magnitude of 6.5. For Gable Mountain, an east-west structure that passes through the northern portion of the Hanford Site, a maximum magnitude of 5.0 was estimated. These estimates were based on the inferred sense of slip, the fault length, and/or the fault area. The floating earthquake for the tectonic province was developed from the largest event located in the Columbia Plateau, which is the Milton-Freewater earthquake of magnitude 5.75. The maximum swarm earthquake for the purpose of WNP-2 seismic design was of magnitude 4.0, a value based on the 1973 maximum swarm earthquake, which NRC had concluded was smaller than previously estimated.

Among the potential earthquake sources considered for the WNP-2 site, earthquakes associated with the Gable Mountain structure dominated the exceedance probability calculations. Rohay (1989) analyzed earthquake recurrence rates based on historical and instrumental data for the Yakima Fold Belt and the eastern Washington region. He concluded that the appropriate recurrence rate for the Hanford Site was the same as the recurrence rate based on historical data for the eastern Washington region. The seismic hazard curves calculated for a number of areas at Hanford are based on ground accelerations and recurrence intervals of earthquakes larger than magnitude 5. The recurrence curves are given in Figure D.4. The proposed NPR site is about 5 km northwest of the 400 Area. Note that the highest acceleration-probability conditions occur closest to Gable Mountain.

D.2.5 Soils

Hajek (1966) lists and describes 15 different soil types on the Hanford Site. The soil types, which vary from sand to silty and sandy loam, are listed and briefly described in Table D.4 and shown in Figure D.5. Various classifications, including a land use classification, are also given in Hajek (1966). Hajek (1966) defines soil thickness as the depth to which the roots of native plants extend. For Hanford vegetation, the soil thickness is several meters. Estimates of unsaturated hydraulic conductivity for sand/sandy soils generally fall within the range 1×10^{-6} to 1×10^{-4} cm/s, depending on the soil type, vegetation, and depth (Gee, Rockhold, and Downs 1989). The erosion potential of the Hanford surficial material is low because of the relatively flat slopes and low annual precipitation. The soils at the proposed NPR site have not been classified with respect to Prime Farmland and Unique Farmland criteria by the U.S. Department of Agriculture's Soil Conservation Service. However, none of the soils on the NPR site would be classified as Unique Farmland.

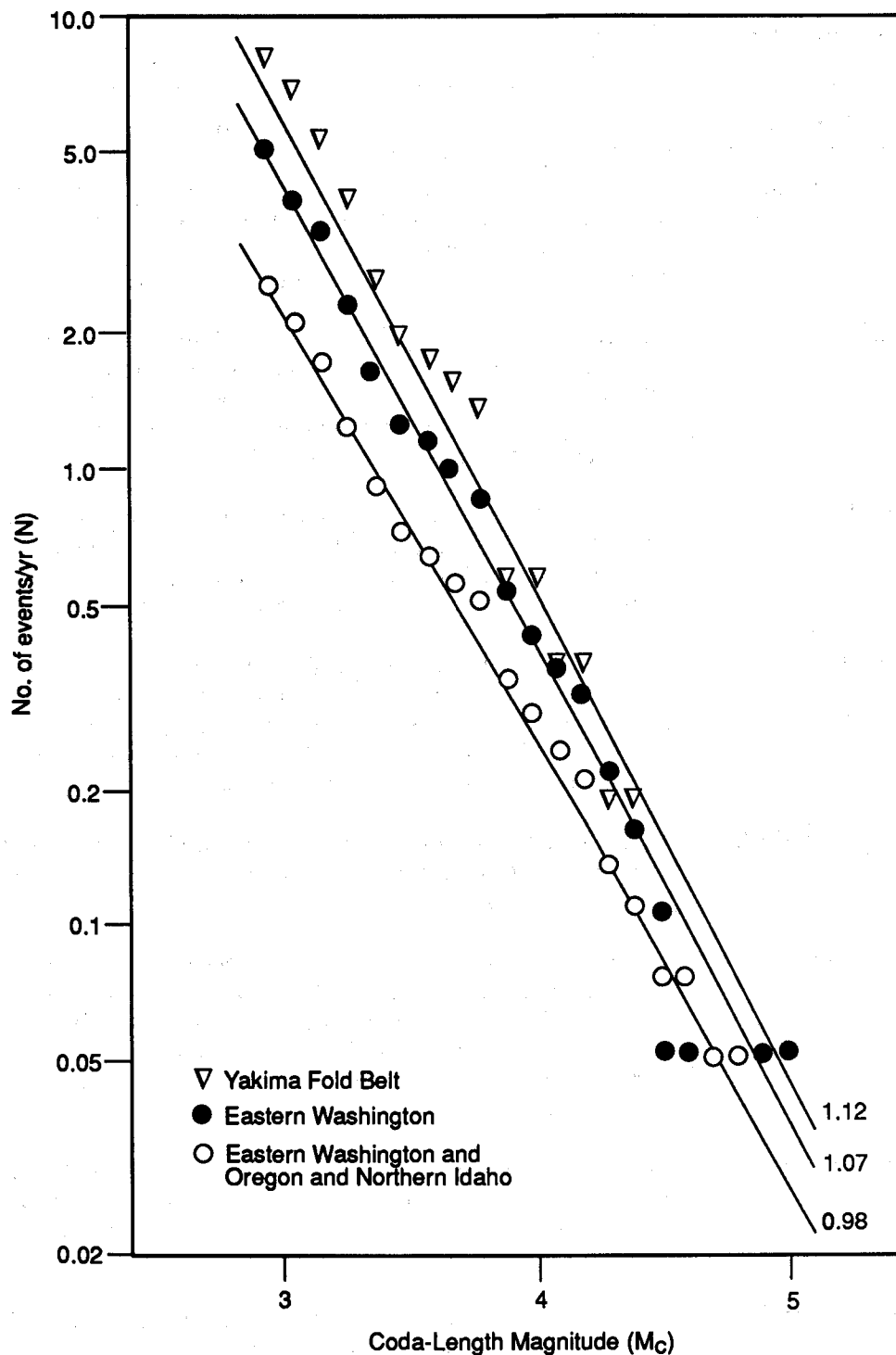


FIGURE D.4 Earthquake Recurrence Curves, 1969-1986 (cumulative annual recurrence rates based on the coda-length magnitudes [M_c] for three regions, normalized to a 100,000-km² area; slopes [b -values], determined by the maximum likelihood method, are shown in the lower right-hand corner) (Source: Modified from Rohay 1989)

TABLE D.4 Soil Types at the Hanford Site

Name (Symbol)	Description
Ritzville Silt Loam (Rl)	Dark-colored silt loam soils midway up the slopes of the Rattlesnake Hills. Developed under bunch grass from silty wind-laid deposits mixed with small amounts of volcanic ash. Characteristically >150 cm deep, but bedrock may occur at <150 cm but >75 cm.
Rupert Sand (Rp)	One of the most extensive soils on the Hanford Site. Brown-to-grayish-brown coarse sand grading to dark grayish brown at about 90 cm. Developed under grass, sagebrush, and hopsage in coarse sandy alluvial deposits that were mantled by windblown sand. Hummocky terraces and dunelike ridges.
Hezel Sand (He)	Similar to Rupert sand; however, a laminated grayish brown, strongly calcareous silt loam subsoil usually encountered within 100 cm of the surface. Surface soil is very dark brown and formed from windblown sands that mantled lake-laid sediments.
Koehler Sand (Kf)	Similar to other sandy soils on the Hanford Site. Developed in a windblown sand mantle. Differs from other sands in that the sand mantles a lime-silica cemented layer "hardpan." Very dark, grayish brown surface layer is somewhat darker than Rupert sand. Calcareous subsoil is usually dark grayish brown at about 45 cm.
Burbank Loamy Sand (Ba)	Dark-colored, coarse-textured soil underlain by gravel. Surface soil is usually about 40 cm thick but can be 75 cm thick. Gravel content of subsoil ranges from 20% to 80%.
Kiona Silt Loam (Si)	Occupies steep slopes and ridges. Surface soil is very dark grayish brown and about 10 cm thick. Dark brown subsoil contains basalt fragments 30 cm and larger in diameter. Many basalt fragments found in surface layer. Basalt rock outcrops present. A shallow stony soil normally occurring in association with Ritzville and Warden soils.
Warden Silt Loam (Wa)	Dark grayish brown soil with a surface layer usually 23 cm thick. Silt loam subsoil becomes strongly calcareous at about 50 cm and becomes lighter colored. Granitic boulders are found in many areas. Usually >150 cm deep.
Ephrata Sandy Loam (El)	Surface is dark colored; subsoil is dark grayish brown, medium-textured soil underlain by gravelly material, which may continue for many feet. Level topography.
Ephrata Stony Loam (Eb)	Similar to Ephrata sandy loam. Differs in that many large hummocky ridges are made up of debris released from melting glaciers. Areas between hummocks contain many boulders several feet in diameter.

TABLE D.4 (Cont'd)

Name (Symbol)	Description
Scootney Stony Silt Loam (Sc)	Developed along the north slope of Rattlesnake Hills; usually confined to floors of narrow draws or small fan-shaped areas where draws open onto plains. Severely eroded with numerous basaltic boulders and fragments exposed. Surface soil is usually dark grayish brown, grading to grayish brown in the subsoil.
Pasco Silt Loam (P)	Poorly drained, very dark grayish brown soil formed in recent alluvial material. Subsoil is variable, consisting of stratified layers. Only small areas found on Hanford Site; located in low areas adjacent to the Columbia River.
Esquatzel Silt Loess (Qu)	Deep dark brown soil formed in recent alluvium derived from loess and lake sediments. Subsoil grades to dark grayish brown in many areas, but color and texture of the subsoil vary because of the stratified nature of the alluvial deposits.
Riverwash (Rv)	Wet, periodically flooded sand, gravel, and boulder deposits; occurs on islands in the Columbia River subject to overflow and on adjacent land.
Dune Sand (D)	Miscellaneous land type consisting of hills or ridges of sand-sized particles that have been drifting and piled up by wind and are either actively shifting or so recently fixed or stabilized that no soil horizons have developed.
Lickskiller Silt	Occupies ridge slopes of Rattlesnake Hills at elevations higher than 765 m. Similar to Kiona series except surface soils are darker. Shallow over basal bedrock, with numerous basalt fragments throughout the profile.

Source: Hajek 1966.

D.3 IDAHO NATIONAL ENGINEERING LABORATORY

D.3.1 Geologic Setting

The INEL is in southeastern Idaho on the Eastern Snake River Plain, a physiographic province extending from the Idaho-Oregon border to the Island Park-Yellowstone Volcanic Plateau (Figure D.6). The Eastern Snake River Plain is bordered on the northwest and southeast by the Basin and Range province. The Snake River Plain apparently began developing in southwestern Idaho approximately 15 million years ago as the North American plate moved westward over a stationary plume or "hot spot" in the earth's mantle (Armstrong, Leeman, and Malde 1975; Smith and Christiansen 1980; Mabey 1982).

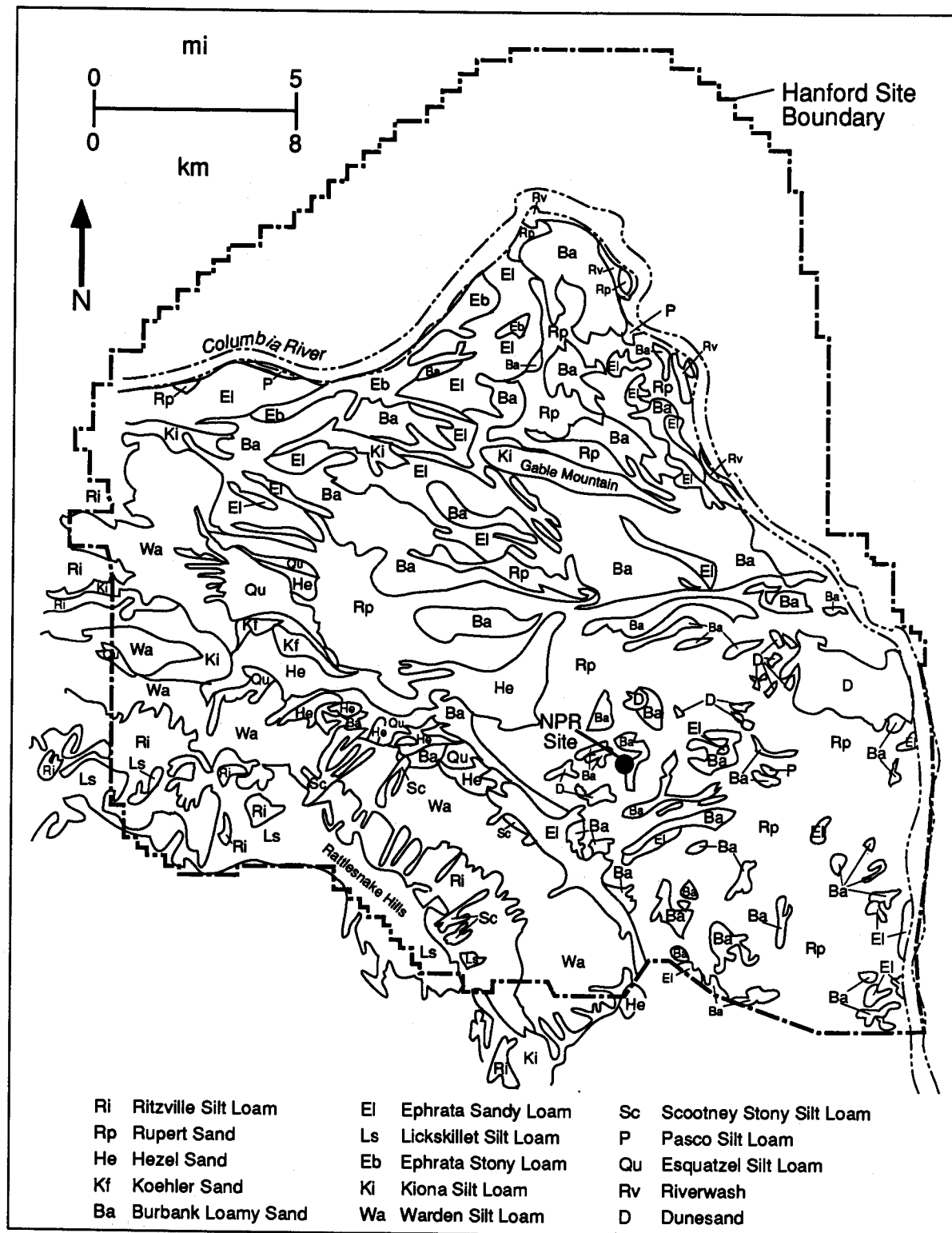


FIGURE D.5 Soils Map of the Hanford Site (Source: Modified from Hajek 1966)

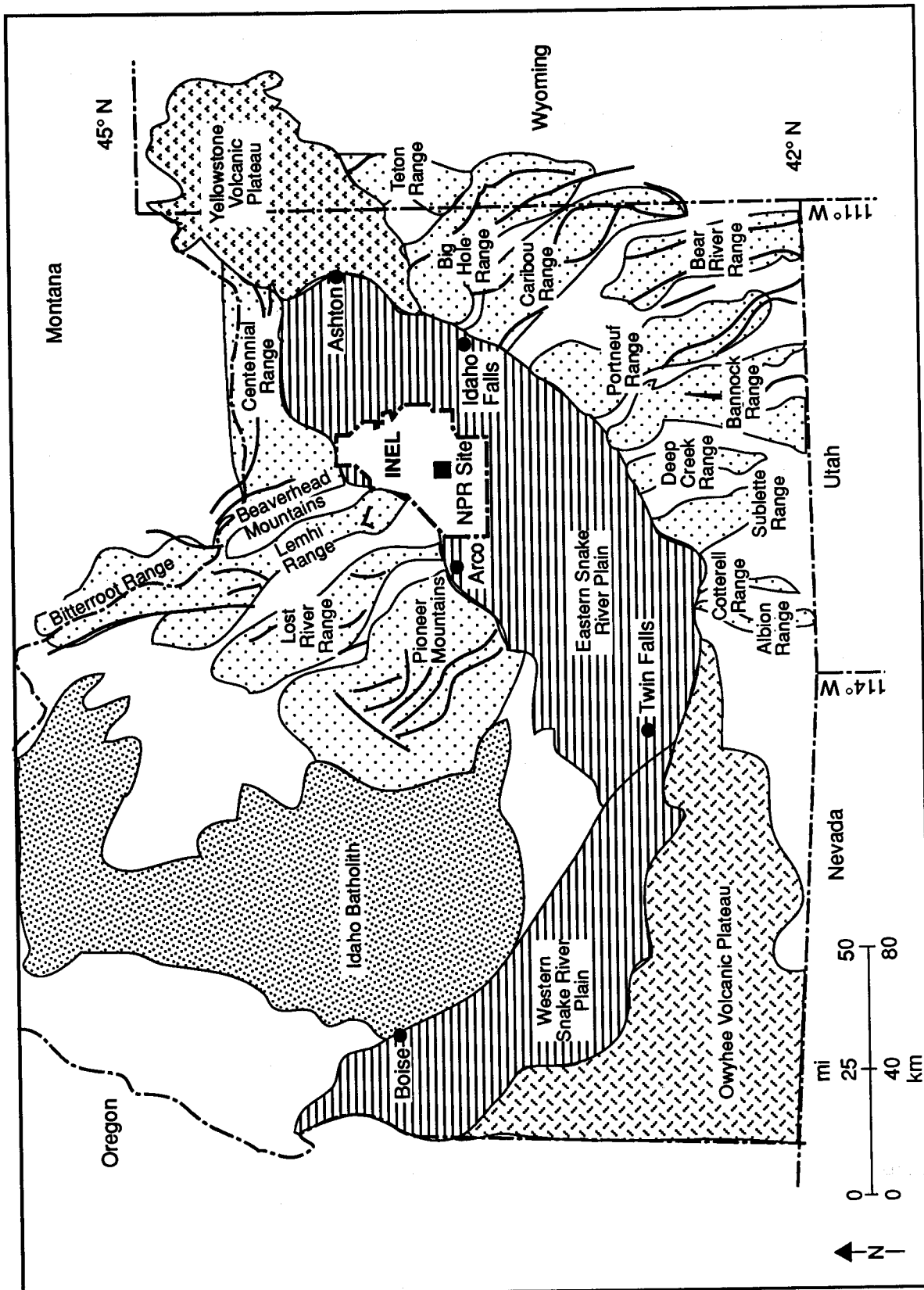


FIGURE D.6 Geologic Setting of INEL (Source: Modified from EG&G 1984)

The general stratigraphy of the Snake River Plain consists of Miocene and younger volcanic rocks resting upon deformed or tilted sedimentary and plutonic rocks ranging in age from Precambrian to Mesozoic and faulted remains of Middle to Late Eocene volcanics. Rhyolite, basalt, and sedimentary rocks dominate the lithologic assemblages. Rhyolitic rocks, by far the most voluminous, are represented by ash-flow tuffs and other pyroclastics, domes, and lava flows. These silicic rocks are oldest to the west, becoming progressively younger toward Yellowstone to the east. The silicic rocks of the Eastern Snake River Plain are covered by basalt flows. Interlayered sediments consist of loess, eolian sand, alluvium, and lacustrine and alluvial fan deposits.

Basalts of the Snake River Group are the predominant rock type at INEL. The estimated thickness of the basalt in the Eastern Snake River Plain is 1-2 km (Doherty, McBroome, and Kuntz 1979; Kuntz et al. 1983). This statement is supported by data from the INEL-1 deep well, which penetrated 690 m of basalt.

The proposed NPR site is in the south-central portion of INEL. The rocks at the surface are basalts formed from lavas that erupted 233,000 years ago from a vent about 10 km southwest of the NPR site (Champion, Lanphere, and Kuntz 1988). Locally, the basalts are covered by a thin veneer (0-1.8 m) of loess. The local stratigraphy to a depth of 186 m is shown in Figure D.7. The letter designations of the lava flows correspond to those in Figure D.8 so that the potassium/argon ages (Champion, Lanphere, and Kuntz 1988) can be associated with the lava flows.

The adjacent Basin and Range province comprises displaced Precambrian and Paleozoic sedimentary rocks that were folded and faulted during the early Cretaceous as they were transported to the east on gently dipping thrust faults (Crone et al. 1987). Subsequent Cenozoic tectonism produced the modern basins and ranges by northeast-southwest extension on normal faults bounding one or both flanks of the ranges (Scott, Pierce, and Hait 1985). These faults cut or merge at depth with the thrust faults formed earlier (Armstrong and Oriel 1965). The three mountain ranges that border INEL on the north are the Lost River, Lemhi, and Beaverhead.

D.3.2 Geologic Structures

In southern Idaho, the major tectonic features are the Basin and Range province and the Yellowstone-Snake River Plain Volcano-Tectonic province (Figure D.6). Tectonic activity in late Tertiary and Quaternary time in each of these provinces has profoundly affected the earth's crust and surface. In the Basin and Range province, east-northeast-directed crustal extension has broken the crust into long, narrow fault blocks bounded by north-northwest-trending normal faults. Tilting and differential uplift of the blocks have produced the long, north-northwest-trending mountain ranges characteristic of southern Idaho and of the main part of the Basin and Range province in Nevada and western Utah (Lipman, Prostka, and Christiansen 1972).

Faults in the region can be grouped into three categories: (1) Paleozoic and Mesozoic thrust faults related to development of the overthrust belt, (2) range-front normal faults associated with the Basin and Range province to the south and north of the Snake River Plain, and (3) faults related to caldera collapse associated with the rhyolitic stage of volcanic activity

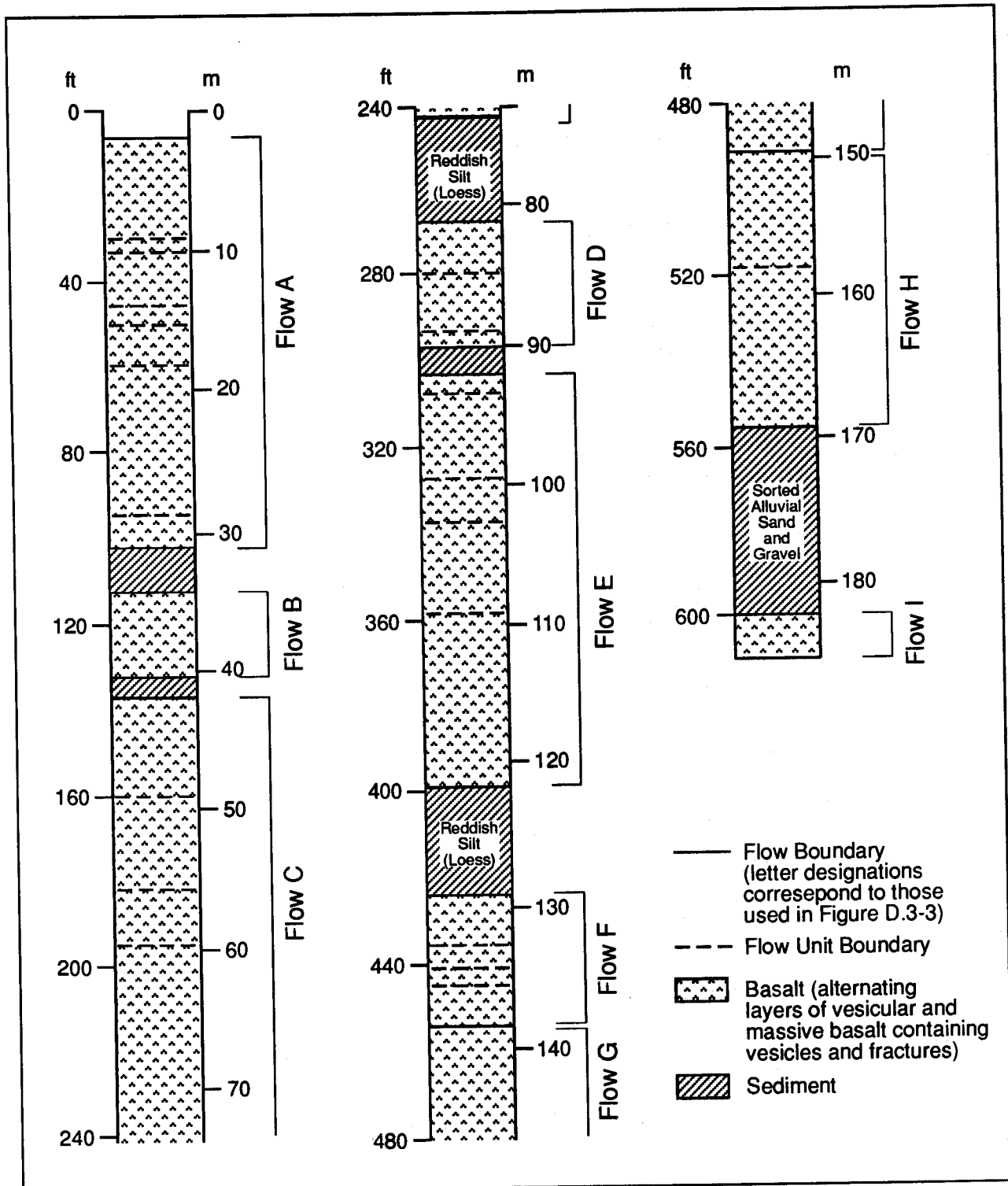


FIGURE D.7 Lithostratigraphy near the Proposed NPR Site at INEL (basalt flows correspond to those shown in Figure D.8) (Source: Modified from Morgan 1985)

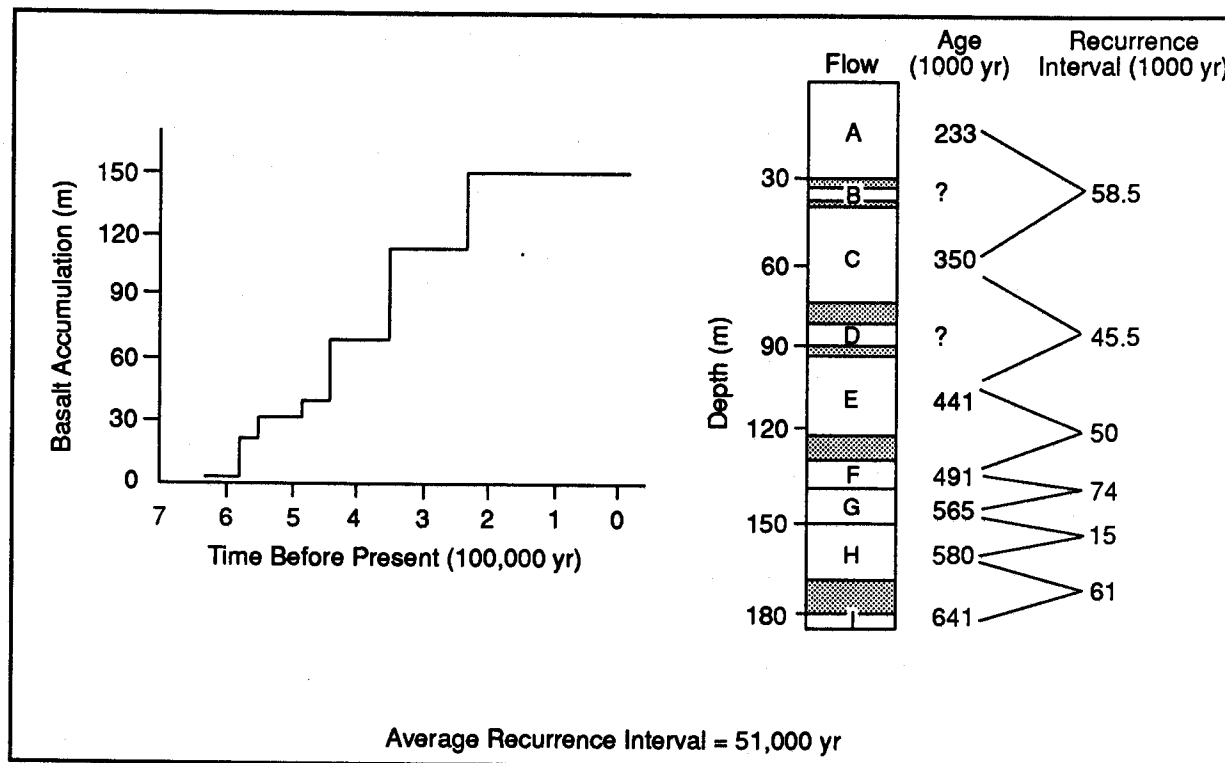


FIGURE D.8 Accumulation Rates and Recurrence Intervals for Basalt Flows (based on data from the Site E borehole) (Source: Modified from Champion, Lanphere, and Kuntz 1988)

during development of the Snake River Plain. Table D.5 summarizes known information about the Tertiary to Holocene faults and their constitutive segments. Figure 4.2.1-1 shows the distribution of these structures.

The thrust faults associated with the overthrust belt occur both north and south of the Eastern Snake River Plain in southeastern Idaho. They were formed when the Pacific tectonic plate was impinging directly (in a trench relationship) against the western edge of the North American plate. This activity died out near the end of Mesozoic time and in early Tertiary time (65 million to 40 million years ago) (Lipman, Prostka, and Christiansen 1972). The faults may have been reactivated during Basin and Range rifting so that normal faults developed on old thrust fault ramps.

Basin and Range faults in the INEL region are normal faults that occur along the flanks of north-northwest-trending mountain ranges. They are half-graben faults that allowed the intervening blocks to tilt to the east and produce the Lost River, Lemhi, and Beaverhead ranges. The location of the focus of the 1983 Borah Peak earthquake and an analogy with other Basin and Range faults indicate that these faults dip about 45° to the west-southwest (Doser 1985). The Lost River, Lemhi, and Beaverhead faults are late Tertiary to Holocene in age (Scott, Pierce, and Hait 1985). They extend from the north margin of the Eastern Snake River Plain for a distance of 100 km or more to the Salmon River drainage, where they die out against the Idaho Batholith. These faults, or at least segments of them, are "capable" faults. They control the seismic design of facilities at INEL.

TABLE D.5 Major Fault Systems Proximal to the Eastern Snake River Plain

Fault/Segment	Important Points and Characteristics ^a
Sawtooth, White Cloud Peaks area ^b	Contemporary earthquake swarms; maximum magnitude = 6.2; several mapped normal faults
Lost River Fault ^c	
Arco segment	(1) ~30; (2) trenching; (3) 2-3; (4) 10; (5) 0.12; (6) 30-40
Pass Creek segment	(1) ~30-50; (2) scarp morphology; (4) 30; (6) 30-50 (?)
Mackay segment	(1) 4.3-6.8; (2) trenching; (4) 22; (5) 0.3; (6) 4-7
Thousand Springs segment	(1) 1983; (2) trenching/earthquake; (3) 1.5; (4) 36; (5) 0.3
Warm Springs segment	(1) 1983; (2) trenching; (4) 15; (5) 0.3; (6) <15
North segment	(1) >Late Quaternary; (2) scarp morphology; (3) low structural relief; (4) 20
Lemhi Fault ^d	
Howe segment	(1) ~30; (2) trenching; (3) 2-3; (4) 15-20; (6) 10
Fallert Springs segment	(1) 15-30; (2) scarp morphology; (4) ~25-30
Sawmill Gulch segment	(1) <10; (2) trenching; (3) 1.7; (4) 43
Goldburg segment	(1) 10-15; (2) scarp morphology; (4) 12
Patterson segment	(1) 10-15; (2) scarp morphology; (4) 23
May segment	(1) 15-30 (?); (2) scarp morphology; (4) 23
Beaverhead Fault ^e	
Blue Dome segment	(1) >30; (2) scarp morphology; (4) 25
Nicholia segment	(1) 10-15; (2) scarp morphology; (4) 42
Baldy Mountain segment	(1) >30; (2) scarp morphology; (4) 21
Leadore segment	(1) <10; (2) scarp morphology; (4) 23
Mollie Gulch segment	(1) 10-15 (?); (2) scarp morphology; (4) 20
Lemhi segment	(1) >30; (2) scarp morphology; (4) 20
Red Rock Fault ^e	
Sheep Creek segment	(1) <10; (2) scarp morphology; (4) 16
Timber Butte segment	(1) 10-15; (2) scarp morphology; (4) 11
Centennial Fault ^f	
Western Centennial Valley segment	(1) <10; (2) scarp morphology; (4) 23
Red Rocks Lake segment	(1) >20 (?); (2) scarp morphology; (4) 1
Henrys Lake segment	(1) <10; (2) scarp morphology; (4) 1
Madison Fault ^g	
Madison Canyon segment	(1) Late Holocene (1947? 1959?); (2) scarp morphology; (4) 34 (total fault length = 117 km)
Hebgen Fault ^h	(1) 1959; (2) scarp morphology; (3) 4.4; (4) 13 (+14 km on Red Canyon Fault)
Red Canyon Fault ^h	(1) 1959; (2) scarp morphology; (3) 4.4; (4) 45; (5) 1.2-1.5 (pre-1959); (6) 4.3

TABLE D.5 (Cont'd)

Fault/Segment	Important Points and Characteristics ^a
Yellowstone area ⁱ	Numerous north-trending normal faults around Yellowstone Caldera, with Quaternary movement; contemporary seismicity, 1975, maximum magnitude = 6.1; recurrence rate = 700-750 years for magnitude 7 earthquake based on seismic moment rates
Teton Fault ^j	(1) <11-14; (2) scarp morphology; (3) 3; (4) 80-100; (5) 1.7-2.2; (6) 1.4-2.3
Grand Valley Fault ^k	(1) >15-30; (2) scarp morphology; (4) 72; (5) 0.02-0.04
SNAKE RIVER FAULT ^k	(1) >15-30; (2) scarp morphology; (4) 50; (5) 0.001
Star Valley Fault ^k	(1) <10-15; (2) scarp morphology; (3) 3.3-6.3; (4) 58; (5) 0.6-1.2; (6) 5-7
Bear Lake Fault	No data available
Gem Valley Fault	No data available
Northern Wasatch Fault ^l	
Collinston segment	(1) >13; (2) scarp morphology; (4) 30
Brigham City segment	(1) 4.7 ± 0.5; (2) trenching; (4) 40; (6) 1.1 ± 1.0
Weber segment	(1) 0.5 ± 0.3; (2) trenching; (3) 1.7-3.7; (4) 61; (5) 1.2-2.8; (6) 3.25
East Cache Valley Fault	No data available

^aCharacteristics are coded in the following order: (1) = age of last movement (thousands of years ago, except when specific year is given); (2) = type of study; (3) = displacement per last event (m); (4) = length (km); (5) = slip rate (mm/yr); and (6) = recurrence interval (thousands of years); (?) indicates uncertainty. Absence of a particular code in an entry means that data are not available for that characteristic.

^bFrom Dewey 1987; Jackson and Zollweg 1988; Smith, Richins, and Doser 1985.

^cFrom Pierce 1985, 1988; Scott, Pierce, and Hait 1985; Crone et al. 1987; Schwartz 1988; Malde 1987; Piety et al. 1986.

^dFrom Knuepfer 1989; Turko 1988; Baltzer, Knuepfer, and Turko 1989; Malde 1987; Haller 1988; Scott, Pierce, and Hait 1985.

^eFrom Haller 1988; Stickney and Bartholomew 1987.

^fFrom Stickney and Bartholomew 1987; Johnson 1981.

^gFrom Stickney and Bartholomew 1987; Schneider 1985.

^hFrom Stickney and Bartholomew 1987; Doser 1985.

TABLE D.5 (Cont'd)

ⁱFrom Pitt, Weaver, and Spence 1979; USGS 1988; Doser 1984.

^jFrom Pierce and Good 1990; Pierce and Morgan 1990; Smith, Byrd, and Susong 1990; Gilbert, Ostenaar, and Wood 1983.

^kFrom Anders et al. 1989; Piety et al. 1986.

^lFrom Schwartz 1988; Machette, Personius, and Nelson 1991; Machette, Personius, and Nelson 1987; Swan, Schwartz, and Cluff 1980; Personius 1988; Nelson and Personius 1990; Piety et al. 1986.

Caldera collapse faults occur beneath the basalt and sedimentary cover of the Eastern Snake River Plain. At a few localities along the edge of the plain, there is evidence of these collapse faults having propagated through the overlying basalts. The ones beneath INEL were active 6.5 million to 4.3 million years ago. There is no evidence of more recent movement.

The Howe segment of the Lemhi Fault, which is located at the northwestern edge of INEL, is the fault closest to the proposed NPR site. The possibility of a large-magnitude earthquake on this segment is the most serious seismic hazard at the NPR site. Studies to quantify the magnitude of events that have occurred on this segment are underway and should be completed in time for Phase I designs.

D.3.3 Volcanism

The Snake River Plain has been produced over the past 14 million years by a bimodal progression from rhyolitic to basaltic magmatism from southwestern Idaho to the Yellowstone area. The first stage in this progression is large eruptions of rhyolitic ash from huge calderas perched over immense silicic magma chambers. This stage is currently operative at Yellowstone. The second stage is characterized by basalt extrusions. Basalt flows cover much of the Eastern Snake River Plain. The areal extent and timing of the rhyolitic and basaltic events on the Eastern Snake River Plain are summarized in Table D.6.

Directly beneath and to the east of INEL are three major calderas inferred from mapping, well logging, and geophysical exploration (Figure 4.2.1-1). Associated with these calderas are three voluminous and areally extensive ignimbrite sheets (ash-flow tuffs) (Hackett and Morgan 1988). The recurrence interval of the major ignimbrite eruptions on the Eastern Snake River Plain is on the order of 0.5-1 million years (Izett and Wilcox 1982). The volcanic centers have gradually migrated northeastward at a rate of 3-4 cm/yr and are now located in the Yellowstone Plateau. Northeast of INEL, at Island Park, the calderas are 2.0 million to 1.3 million years old. The Yellowstone Caldera is 0.6 million years old.

Four rhyolite lava domes, ranging in age from 1.4 million to 0.3 million years (Kuntz and Dalrymple 1979), were emplaced along a northeast-trending zone that parallels the length of the Eastern Snake River Plain and crosses INEL's southern boundary. Middle Butte is an uplifted block of basalt with a rhyolite core and is located about 8 km inside the southern boundary of INEL. East Butte lies 6.4 km east of Middle Butte at the eastern boundary of INEL.

TABLE D.6 Summary of Volcanic Events on the Eastern Snake River Plain

Age (years before present)	Volcanic Event	Rock Type	Areal Extent (km ²)
15,000 to 2,076	Craters of the Moon lava field	Basalt	1,600
2,130	Kings Bowl lava field	Basalt	4.5
2,360	Wapi lava field	Basalt	300
4,100	Hell's Half Acre lava field	Basalt	425
10,780	Cerro Grande lava field	Basalt	180
11,940	North and South Robbers lava fields	Basalt	8.5
300,000	Big Southern Butte	Rhyolite dome	12.5
600,000	East Butte	Rhyolite dome	2
4 million to 10,000	Snake River Group	Basal	23,600
4.3 million	Kilgore tuff	Ignimbrite ^a	>35,000
6.0 million	Blue Creek tuff	Ignimbrite ^a	
6.5 million	Blacktail tuff	Ignimbrite ^a	

^aIgnimbrite is the same as rhyolite ash-flow tuff.

Conversion factor: 1 km² = 0.3861 mi².

Sources: Data from Kuntz 1978, 1989; Kuntz et al. 1986b; Morgan 1988.

The largest rhyolite dome is Big Southern Butte, which is located 6.4-8 km south of the INEL boundary (Spear 1979; Spear and King 1982). These domes are anomalous, in that they formed during a time of basaltic activity. The shortest known interval between rhyolite dome formation is 300,000 years.

The basaltic volcanism around INEL began about 4 million years ago, occurring most recently about 2,000 years ago at Craters of the Moon. Nine distinct basalt lavas were extruded during an interval from 600,000 to 230,000 years ago (Champion, Lanphere, and Kuntz 1988). The vent for the uppermost basalt lava flow is located a few kilometers southeast of the proposed NPR site (Champion, Lanphere, and Kuntz 1988). These episodes of basaltic activity lasted about 100 years and were separated by long intervals of quiescence (Kuntz 1978; Kuntz and Dalrymple 1979). Investigations of flows from basaltic buttes indicate that eruptions from individual vents occurred intermittently over a period of 5-10 years (EG&G 1984; Kuntz et al. 1986b).

Most of the basalts at INEL are associated with shield volcanoes, which are commonly aligned along volcanic rift zones. Fissure flows are also associated with rift zones. Four major northwest-southeast-trending rift zones are postulated to cut across the Eastern Snake River Plain (Figure 4.2.1-2). The Great Rift zone (approximately 72 km southwest of the NPR site) is the source of three lava fields: Craters of the Moon, Wapi, and Kings Bowl (Kuntz et al. 1986a). Other lava fields associated with rift zones include the North and South Robbers, the Cerro Grande, and the Hell's Half Acre (EG&G 1984). Volcanism in the Arco Rift zone was in progress from 600,000 to 500,000 years ago (Smith, Hackett, and Rodgers 1989).

Forty kilometers southwest of INEL, the Great Rift in the Craters of the Moon area has experienced eight episodes of basalt eruption in the past 15,000 years. Each episode lasted about 100 years, and the time between episodes ranged from about 500 to 3,000 years, with an average recurrence interval of 2,000 years (Kuntz et al. 1986a, 1986b). Kuntz et al. (1986a) suggest that the next eruptive period along the Great Rift will most likely occur within the next 1,000 years and produce 5-6 km³ of lava. The eruptive history of other volcanic rift zones on the Eastern Snake River Plain is not well chronicled because of the lack of radiometric age determinations.

The Radioactive Waste Management Complex (RWMC) and ANL-West sites are located within the same topographic/volcanic basin as the proposed NPR site. Recurrence intervals have been determined for those specific areas through studies of rock cores (Kuntz and Dalrymple 1979; Kuntz et al. 1980). Within the RWMC area, flows from seven different sources have been deposited in the last 500,000 years. These flows can be grouped into three major episodes that occurred 450,000, 225,000, and 95,000 years ago. These episodes were separated by intervals of 150,000 to 225,000 years during which no flows reached the area. Approximately one out of five eruptions within the basin has reached the RWMC site (Kuntz et al. 1980). The ANL-West site has experienced five flows during the past 500,000 years. These flows were separated by irregular intervals ranging from 1,000 to 10,000 and 300,000 to 400,000 years. The average recurrence interval for the ANL-West site is 1×10^{-5} to 1.25×10^{-5} per year (Kuntz and Dalrymple 1979).

The Site E drill hole, which is close to the western boundary of the NPR site, was drilled in the spring of 1984 to a depth of 185.6 m. Reliable ages were obtained for seven of the nine lava flows encountered (Champion, Lanphere, and Kuntz 1988). The ages show that the eruptions took place between 641,000 and 233,000 years ago (Figure D.8). The average recurrence interval during that period of activity is 51,000 years. The time interval since the last activity (233,000 years) is about 4.6 times the average recurrence interval and 3 times the largest recurrence interval. These data suggest that the source or sources of basaltic lava at the NPR site are no longer active and therefore not expected to erupt again. If basaltic lava were to erupt from nearby volcanic rift zones, the flows would have to travel 10-20 km to reach the NPR site. Further, such lava flows would be unlikely to directly affect the site because the median thickness of lava flows on the Eastern Snake River Plain is about 4 m and the site is 12-15 m higher than the surrounding area.

Lawrence Livermore National Laboratory is assessing the volcanic hazard for the proposed NPR site. Study results will be incorporated into whatever guidance is forthcoming from DOE's Office of New Production Reactors with respect to NPR facilities at INEL.

D.3.4 Seismicity

The INEL is on the Eastern Snake River Plain, which is bordered on the north, east, and southeast by the Intermontaine Seismic Belt (Richins et al. 1987). This region of seismic activity extends from southern Arizona through eastern Idaho into western Montana (Figure D.9) and is more than 1,300 km long and 96 km wide. Two seismically active zones are associated with it: the Centennial Tectonic Belt (Stickney and Bartholomew 1987) and a zone that extends from southwestern Utah through southern Nevada, where it joins with the Nevada seismic zone.

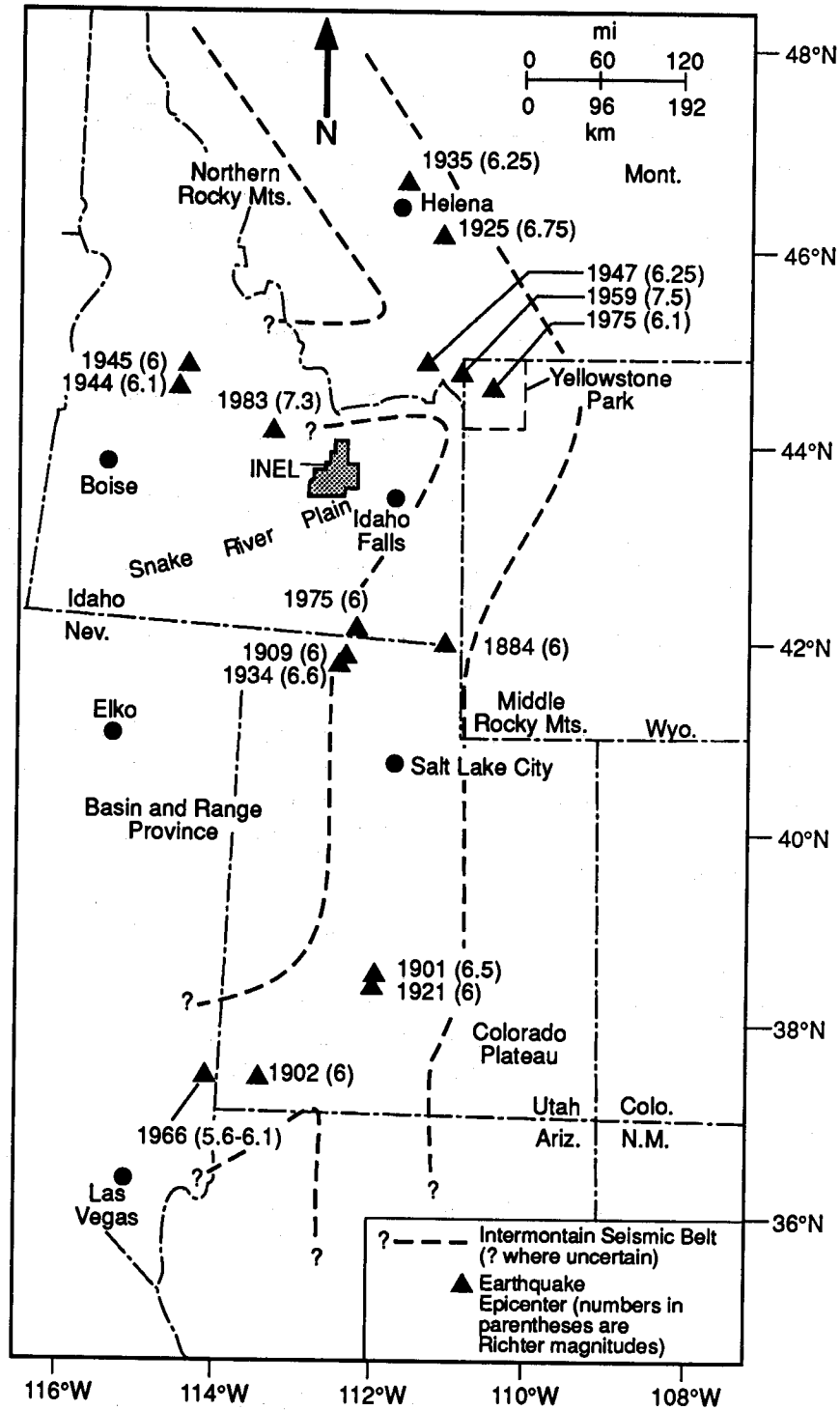


FIGURE D.9 Seismicity Associated with the Intermontaine Seismic Belt (Source: Modified from Richins et al. 1987)

Historical and recent seismic data cataloged by the National Geophysical and Solar Terrestrial Data Center of the National Oceanic and Atmospheric Administration, the National Earthquake Information Center, the University of Utah, and the INEL Seismic Network indicate that earthquakes within the region primarily occur in the Intermontaine Seismic Belt. Table D.7 lists the date, location, and size of 28 historic earthquakes of Richter magnitude 5.5 and larger that have occurred within a radius of 322 km of INEL. Figure D.10 shows the locations of the Table D.7 earthquakes, but not the aftershocks. Table D.8 lists eight of the historic earthquakes of magnitude 5.5 and larger that have been associated with known tectonic structures. Figure 4.2.1-3 shows the locations of earthquakes occurring between 1972 and 1989 as given in the INEL catalog.

Earthquakes occurring before the late 1920s and early 1930s could have large epicentral errors because the locations were based on felt reports. The locations of earthquakes occurring after the 1930s have predominantly been based on instrumentally recorded data. With the exception of the 1905 earthquake of magnitude 5.7 (epicentral error of approximately 0.5°), no earthquakes exceeding magnitude 5.5 have been recorded on the Eastern Snake River Plain.

The INEL Seismic Network began operating with one seismic station in 1971. This network now consists of 20 strong-motion accelerographs located at various critical facilities at INEL and 7 permanent seismic stations located on and around the Eastern Snake River Plain. This network, which was to have been expanded to 11 seismic stations by the end of 1990, has accumulated 19 years of earthquake data. During this period of record, 12 microearthquakes with magnitudes less than 1.3 have occurred on the Eastern Snake River Plain. Between 1983 and 1988, eight microearthquakes with magnitudes lower than 1.0 occurred within INEL boundaries (Pelton, Vincent, and Anderson 1990; Jackson et al. 1989). Several microearthquakes occurred on the Eastern Snake River Plain along the western edge of the INEL boundary and along the eastern edge of the plain near Idaho Falls, Idaho. Although the period of record is short, the frequency of microseismic and macroseismic events demonstrates that the Eastern Snake River Plain is seismically inactive compared with the neighboring Intermontaine Seismic Belt.

The largest historical earthquake that occurred in the Centennial Tectonic Belt was the October 28, 1983, earthquake at Borah Peak, Idaho. This event, which had a magnitude of 7.3, resulted from slippage on a normal range-front fault. Its epicenter was located along the western edge of the Thousand Springs Valley near Borah Peak in the Lost River Range, a location approximately 90-110 km from INEL facilities. The earthquake ruptured 34 km of the Thousand Springs segment of the Lost River Fault. The maximum scarp height was 2.7 m (Crone et al. 1985). Although the shock was felt at INEL, no structural or safety-related damage occurred at INEL facilities. All reactors in operation at the time of the earthquake scrambled automatically or were shut down manually without any problems (Gorman and Guenzler 1983). The INEL was within MMI zone VI (Stover 1985). Thirteen of 16 accelerographs in operation at the time of the Borah Peak earthquake recorded peak horizontal accelerations of 0.022-0.078 g at bedrock and free-field sites (Jackson 1985).

Another major earthquake occurred on August 17, 1959, at Hebgen Lake, Montana, approximately 161 km from INEL. The shock (magnitude 7.5) was felt at INEL, but there was no damage. The INEL was within MMI zone VI for this earthquake (Stover 1985). During 1975, two moderate earthquakes, with magnitudes of 6.0 and 6.1, occurred at locations 183 km and 209 km from INEL. The INEL facilities sustained no damage and were within MMI zones II-III for these events (Cook and Nye 1979; Pitt, Weaver, and Spence 1979).

TABLE D.7 Earthquakes with Magnitudes Greater than 5.5 within 322 km of INEL

Date	Magnitude/ Reference ^a	Modified Mercalli Intensity	Coordinates		Location	Distance from INEL (km)
			Lat. (°N)	Long. (°W)		
1884 Nov 10	6.3/I	VII	42.00	111.26	Bear Lake, Utah	225
1905 Nov 11	5.7/I	VII	42.90	114.50	Shoshone, Idaho	164
1909 Oct 06	6.3/I	VII-IX	41.76	112.66	Hansel Valley, Utah	216
1914 May 13	5.7/I	VI-VII	41.22	111.96	Ogden, Utah	283
1925 Jul 10	6.7/d	NA ^b	46.08	112.13	Clarkston, Mont.	201
1930 Jun 12	5.8/N	NA	42.60	111.00	E of Soda Springs	190
1934 Mar 12	6.6/P	NA	41.70	112.80	Hansel Valley, Utah	222
1934 Mar 12	6.1/N	NA	41.70	112.80	Hansel Valley, Utah ^c	222
1934 Apr 14	5.6/N	NA	41.50	112.50	Hansel Valley, Utah ^c	245
1934 May 06	5.6/N	NA	41.70	112.80	Hansel Valley, Utah ^c	222
1944 Jul 12	6.1/P	VII	44.41	115.06	N of Stanley, Idaho	235
1945 Feb 14	6.0/P	VI	44.70	115.40	N of Stanley, Idaho	235
1947 Dec 17	6.3/P	NA	44.92	111.53	Virginia City, Mont.	225
1959 Aug 18	7.5/D	VII-X	44.83	111.08	Hebgen Lake, Mont.	187
1959 Aug 18	6.5/B	NA	45.00	110.50	Hebgen Lake, Mont. ^d	208
1959 Aug 18	6.0/B	NA	44.80	110.70	Hebgen Lake, Mont. ^d	208
1959 Aug 18	6.5/P	NA	44.88	110.73	Hebgen Lake, Mont. ^d	209
1959 Aug 18	5.6/B	NA	44.80	111.10	Hebgen Lake, Mont. ^d	182
1959 Aug 19	6.0/B	NA	44.76	111.62	Hebgen Lake, Mont. ^d	209
1962 Aug 30	5.7/U	NA	42.03	111.74	Cache Valley, Utah	208
1964 Oct 31	5.8/d	NA	44.86	111.06	Hebgen Lake, Mont.	154
1975 Mar 28	6.0/U	VIII	42.06	112.54	Pocatello Valley, Utah	183
1975 Jun 30	6.1/B	VII	44.74	110.60	Yellowstone National Park, Wyo.	209
1976 Dec 08	5.5/G	NA	44.76	110.79	Yellowstone National Park, Wyo.	198
1983 Oct 28	7.3/S	VII	43.97	113.90	NW of Mackay, Idaho (Borah Peak)	93
1983 Oct 28	5.8/S	NA	44.07	113.91	NW of Mackay, Idaho ^e	98
1983 Oct 29	5.8/S	NA	44.23	114.10	NW of Mackay, Idaho ^e	121
1984 Aug 08	5.8/S	NA	44.47	114.01	Challis, Idaho ^e	127

TABLE D.7 (Cont'd)

^aMagnitude references: B = University of California, Berkeley, unpublished data; d = Doser 1989; D = Doser 1985; G = National Earthquake Information Center, unpublished data; S = Richins et al. 1987; I = conversion from intensity; N = University of Nevada, unpublished data; P = Seismological Laboratory, Pasadena, unpublished data; and U = University of Utah from Richins 1979. Intensity obtained from Stover, Reagor, and Algermissen 1986 or Stover 1985.

^bNA = data not available.

^cAftershock following the M = 6.6, 1934 Hansel Valley, Utah, earthquake.

^dAftershock following the $M_s = 7.5$, 1959 Hebgen Lake, Montana, earthquake.

^eAftershock following the $M_s = 7.3$, 1983 Borah Peak, Idaho, earthquake.

Conversion factor: 1 km = 0.6214 mi.

Source: Data compiled by National Earthquake Information Center.

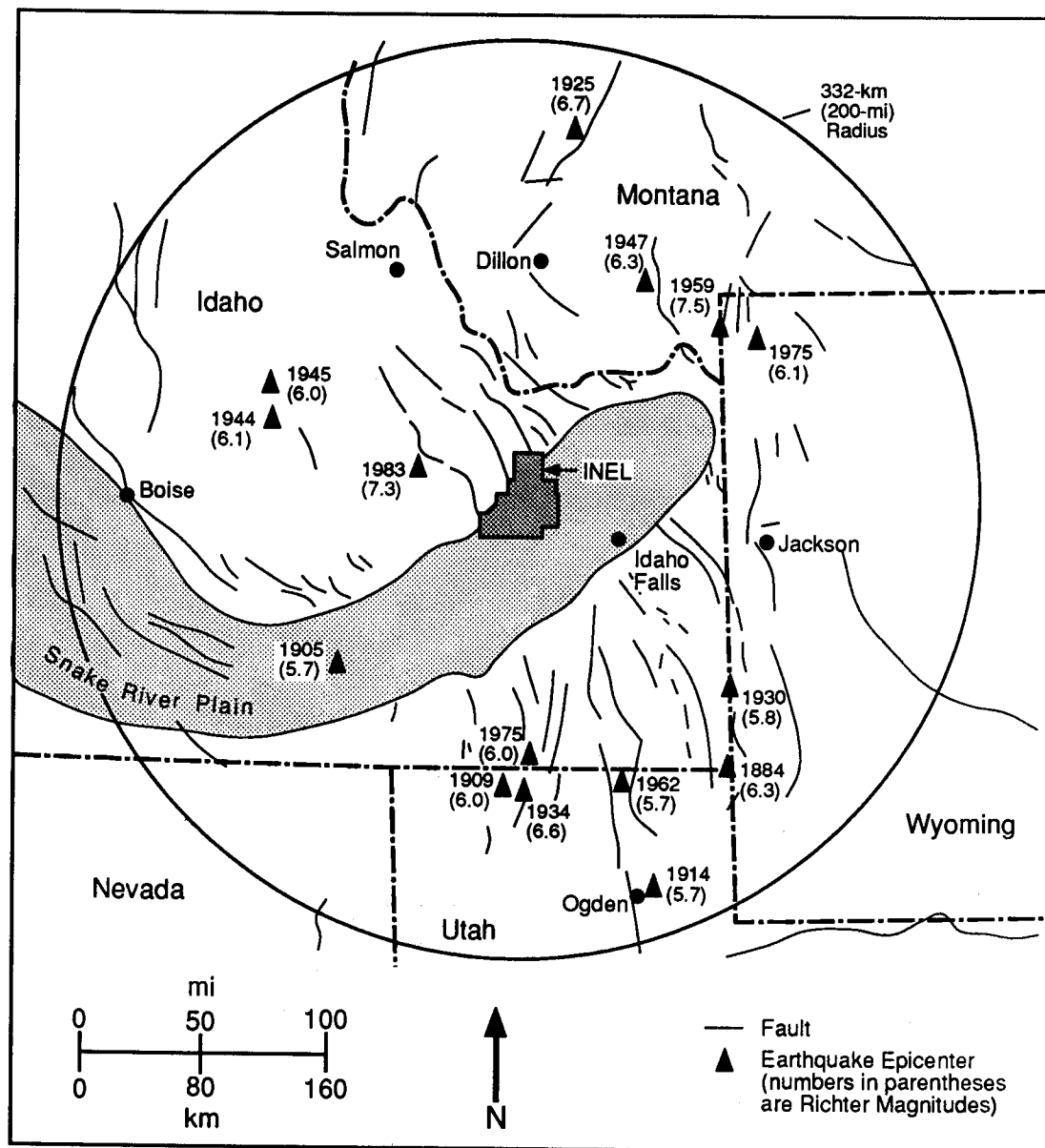


FIGURE D.10 Historic Earthquakes within 322 km of INEL with Calculated Magnitudes of 5.5 or Larger (numbers in parentheses are Richter magnitude values) (Source: Data compiled by National Earthquake Information Center)

Seismic hazard sources at INEL can be divided into two groups: those on the Eastern Snake River Plain and those in the surrounding Basin and Range region. With the possible exception of the 1905 earthquake (epicentral error of about 0.5°), seismic events on the Eastern Snake River Plain have been restricted to small-magnitude earthquakes. In the Basin and Range region, large-magnitude earthquakes are relatively common. Because on-site acceleration depends in part on the magnitude of the source, the magnitude of events that have occurred in association with "capable" faults near the proposed NPR site should be quantified. Woodward-Clyde Consultants conducted paleoseismological studies of the southern Lemhi Fault to

TABLE D.8 Descriptions of Earthquakes within 322 km of INEL That Have Occurred on Normal Faults

Earthquake	Magnitude	Description
1925 Clarkston Valley, Mont.	6.7	Occurred along an oblique normal fault with a trend similar to that of the southern end of the Clarkston Valley Fault (Doser 1989)
1934 Hansel Valley, Utah	6.6	Caused a fault scarp along a north-trending normal fault in Hansel Valley; rupture length was 8 km, with a maximum scarp height of 0.5 m (Richins 1979)
1947 Virginia City, Mont.	6.3	Occurred along a northwest-trending segment of the Madison Fault (Doser 1989)
1959 Hebgen Lake, Mont. (and associated aftershocks)	7.5	Caused a fault scarp along the Hebgen, Red Canon, and part of the Madison faults; combined rupture length was 45 km, with a maximum scarp height of 6.7 m (Doser 1985; Stickney and Bartholomew 1987)
1962 Cache Valley, Utah	5.7	Possibly associated with the Temple Ridge Fault located 15 km east of the East Cache Fault (Westaway and Smith 1989)
1975 Pocatello Valley, Utah	6.0	Occurred along a north-trending normal fault in Pocatello Valley (Arabasz, Richins, and Langer 1979)
1975 Yellowstone National Park, Wyo.	6.1	Associated with normal faulting near the north-central boundary of the Yellowstone Caldera (Pitt, Weaver, and Spence 1979)
1983 Borah Park, Idaho (and associated aftershocks)	7.3	Caused a fault scarp along the Thousand Springs and Warm Spring segments of the Lost River Fault; the combined rupture length was 36 km, with a maximum scarp height of 2.7 m (Crone et al. 1987)

Conversion factors: 1 m = 3.281 ft; 1 km = 0.6214 mi.

determine magnitudes and recurrence intervals. These investigations involved two trenches through the Howe segment of the fault and one trench through the Fallert Springs segment. The resultant data are being analyzed.

TERA Corp. (1984) generated a recurrence curve for seismic events at the ANL-West site, which is part of INEL (Figure 4.2.1-5). Given that acceleration due to gravity is 981 cm/s^2 , the best estimate is a return period of approximately 500 years for a 0.1-g event. For a return period of 10,000 years, the best estimate of peak ground acceleration is approximately 235 cm/s^2 , or 0.24 g.

Deterministic and probabilistic seismic hazard assessments for the NPR site are being conducted by Woodward-Clyde Consultants and Lawrence Livermore National Laboratory, respectively. NPR facilities built at INEL will be designed based on consideration of these studies.

D.3.5 Soils

Underlain by basalt, the soils at INEL are derived from silicic volcanic and Paleozoic rocks from nearby mountains and buttes (McBride et al. 1978). In the southern part of INEL, soils are gravelly to rocky and generally shallow. The northern part of INEL is covered by lacustrine and eolian deposits composed mostly of unconsolidated clay, silt, and sand (Atwood 1970). A survey of the proposed NPR site identified the following soil series: Matherson, Malm, Bockston, Grassy Butte, Whiteknob, Wolverine, and Bondfarm (Breckenridge and McGrath 1989). In general, most areas consist of combinations of these soils. After the soils were classified into series, the site was mapped. The numerical system used corresponds to complexes of the soil series (Figure D.11). Tables D.9 and D.10 describe the soil series found at the NPR site. The subsoils identified are included in the mapping unit descriptions. No weathered rock is associated with the soils at the site because the soils are derived from alluvium or from loess deposited on the surface of basalt flows. The basalts weather very slowly and therefore contribute little, if any, parent material. Soils found at the NPR site do not qualify as either Prime Farmland or Unique Farmland.

D.4 SAVANNAH RIVER SITE

D.4.1 Geologic Setting

The SRS is located in the Aiken Plateau division of the Upper Atlantic Coastal Plain physiographic province of South Carolina (Cooke 1936; du Pont 1980). The Fall Line, which marks the border between the Coastal Plain and Piedmont provinces, lies about 40 km northwest of the center of SRS (Davis 1902). The SRS is bordered to the southwest by the Savannah River (Figure D.12).

Pre-Mesozoic crystalline rocks similar to those of the Appalachian Piedmont underlie SRS at depth. The Dunbarton Basin, an extensional basin within these rocks, is filled with sedimentary rocks of probable Triassic age (Marine and Siple 1974; du Pont 1980; Stephenson, Talwani, and Rawlins 1985). Coastal plain sediments of Cretaceous to Holocene age overlie the basin and basement units and dip gently seaward. These mostly unconsolidated sediments form a wedge that thickens toward the east. These sediments are approximately 280 m thick near the H Area (Siple 1967). In some areas at SRS, the sedimentary wedge rests on crystalline basement and in others on the Triassic rocks (sandstones, silt and claystones, and conglomerates) of the Dunbarton Basin.

The proposed NPR site is located near the geographic center of SRS. This location coincides with the approximate mid-point of the northwestern border of the Triassic Dunbarton Basin, which is associated with the ancient Dunbarton Fault. The Pen Branch Fault -- also

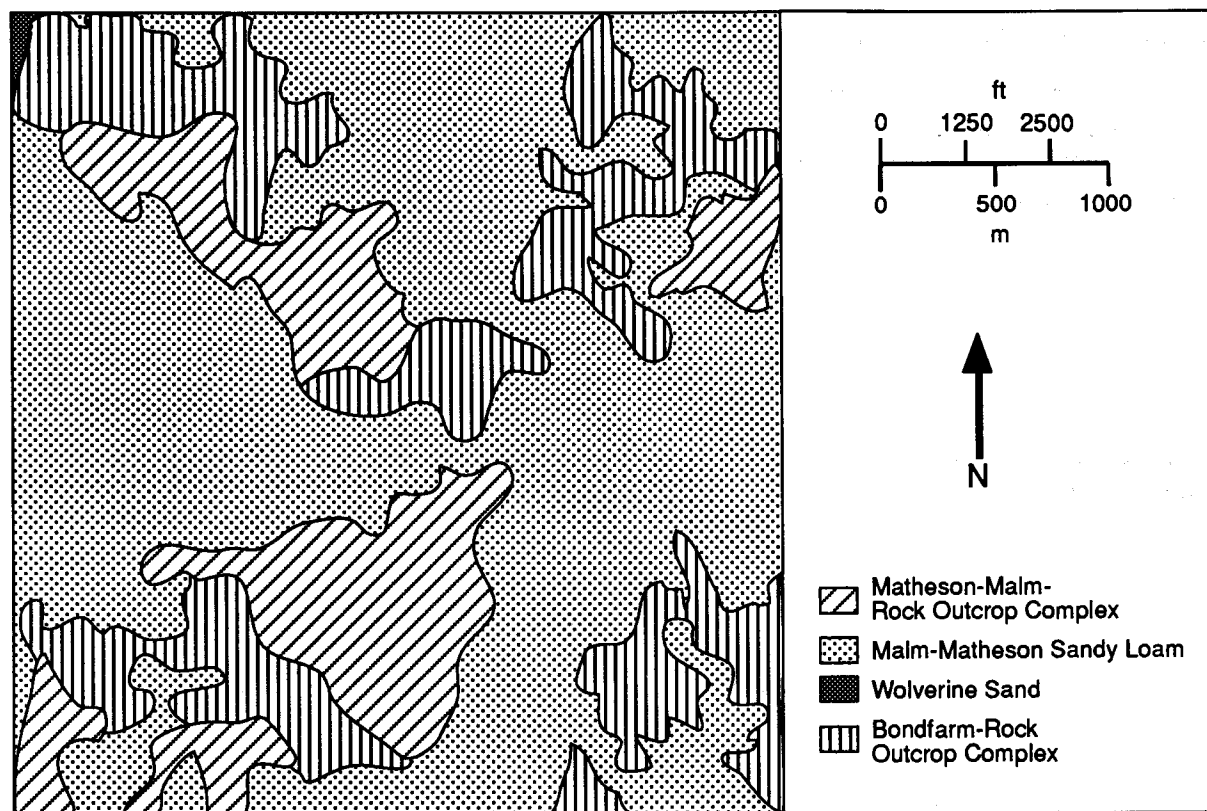


FIGURE D.11 Soils Map of the Proposed NPR Site at INEL (Source: Modified from Breckenridge and McGrath 1989)

identified in this area -- closely parallels the Dunbarton Fault and probably coincides with it in places.

The stratigraphy near the proposed NPR site is shown in Figure D.13. Since 1984, the stratigraphic nomenclature for Cretaceous and younger sediments at SRS has been reevaluated. Under the previously used nomenclature, there are facies at SRS that differ lithologically from time-equivalent facies at their type localities. To improve off-site correlation, an alternate nomenclature for strata at SRS has therefore been proposed (WSRC 1989a). Figure D.14 compares the proposed nomenclature with the previously used nomenclature. Unless otherwise specified, the following lithologic descriptions are characteristic of the geologic units found within SRS boundaries.

Sediments of Cretaceous age begin with the Cape Fear Formation, which consists of poorly sorted, silty-to-clayey quartz sands and interbedded clays. The formation is approximately 9 m thick at the northwestern SRS boundary and thickens to 55 m near the southeastern boundary. Although the Cape Fear Formation has not been traced directly to SRS, the term *Cape Fear* is proposed pending regional correlation (WSRC 1989a).

The Cape Fear Formation is unconformably overlain by the Lumbee Group. This group is composed of (from oldest to youngest) the Middendorf, Black Creek, and Pee Dee formations

TABLE D.9 Soil Types at INEL

Name (Symbol)	Description
Bockston Loam (19)	Surface layer is a brown loam about 23 cm thick. The subsoil is pale brown silt loam about 30 cm thick. The upper part of the substratum is light gray to light brownish gray silt loam and fine sandy loam about 43 cm thick. The lower part to a depth of 152 cm or more is fine sandy loam, to very gravelly loamy sand. The soil is calcareous throughout and has a layer of lime at about 53 cm.
Bondfarm/Rock Outcrop Complex (20)	Occurs on basalt plains. The Bondfarm sandy loam makes up 60% of the complex and rock outcrop 25%. The remaining 15% consists of Malm/Matheson sandy loams. Also included in this mapping unit are areas of soil that have as much as 5% stones on the surface. The Bondfarm soil is formed in sandy eolian deposits. The surface layer is a light brownish gray sandy loam about 10 cm thick. The subsoil and substratum are pale brown sandy loam about 36 cm thick. Basalt occurs at about 46 cm. The soil is calcareous throughout and has a layer of lime at about 10 cm.
Malm/Matheson Sandy Loams (2)	Occurs on basalt plains. The Malm sandy loam makes up about 70% of the complex and Matheson sandy loam 20%. The remainder is composed of Bondfarm sandy loam and rock outcrop. The Malm soil is on convex side slopes, whereas the Matheson soil is on concave side slopes and some convex side slopes. The Malm soil has a brown sandy loam surface layer about 10 cm thick. The underlying material is pale brown and white sandy loam. Basalt occurs at a depth of about 61 cm. The soil is calcareous throughout and has a layer of lime at 30 cm. The Matheson soil has a light brownish gray sandy loam surface layer about 25 cm thick. The underlying material is light brownish gray and light gray sandy loam. Basalt occurs at a depth of 117 cm. The soil is calcareous throughout and has a layer of lime at 25 cm.
Matheson/Malm/Rock Outcrop Complex (1)	Similar to mapping unit 2, but in this case the Malm sandy loam makes up about 20% of the complex, while the Matheson sandy loam makes up 35%. Rock outcrop accounts for 20% of the complex. The remaining 25% is accounted for by Bondfarm loamy sand and Grassy Butte sand.

TABLE D.9 (Cont'd)

Name (Symbol)	Description
Whiteknob Gravelly Loam (12)	The surface layer is a pale brown gravelly loam about 13 cm thick. The upper part of the underlying material is pale brown gravelly loam about 23 cm thick. Below this is a pale brown, very gravelly sandy loam and very gravelly sand to a depth of 152 cm or more. This unit includes areas of Lidy sandy loam and a soil similar to this Whiteknob soil, but without a layer of lime accumulation.
Wolverine Sand (13)	The surface layer is a light brownish gray sand about 15 cm thick. The underlying material to a depth of 152 cm or more is stratified light brownish gray, light gray, and dark gray loamy coarse sand to fine sand.

Source: Breckenridge and McGrath 1989.

TABLE D.10 Physical Properties of Soils at the Proposed NPR Site

Name (Symbol)	Depth (cm)	Permeability (cm/h)	Available Water Capacity (cm/cm)	Erosion Factors		Wind Erodibility Group
				K	T	
Bockston (19)	0-23	5.0-15.0	0.28-0.33	--	--	3
	22-71	1.5-5.0	0.48-0.53	--	--	
	71-97	5.0-15.0	0.33-0.38	--	--	
	97-109	15.0-50.0	0.13-0.25	--	--	
	109-152	50.0	0.08-0.13	--	--	
Bondfarm (20)	0-10	5.0-15.0	0.28-0.33	0.17	1	3
	10-46	5.0-15.0	0.28-0.33	0.17		
Malm (2)	0-10	5.0-15.0	0.33-0.38	0.20	2	3
	10-61	5.0-15.0	0.33-0.38	0.20		
Matheson (2)	0-25	5.0-15.0	0.33-0.38	0.20	3	3
	25-152	5.0-15.0	0.28-0.33	0.20		
Whiteknow (12)	0-36	1.5-5.0	0.23-0.28	0.24	1	4L
	36-50	5.0-15.0	0.13-0.18	0.20		
	50-152	>50.0	0.08-0.13	0.10		
Wolverine (13)	0-152	>50.0	0.15-0.20	0.17	5	1

Source: Breckenridge and McGrath 1989.

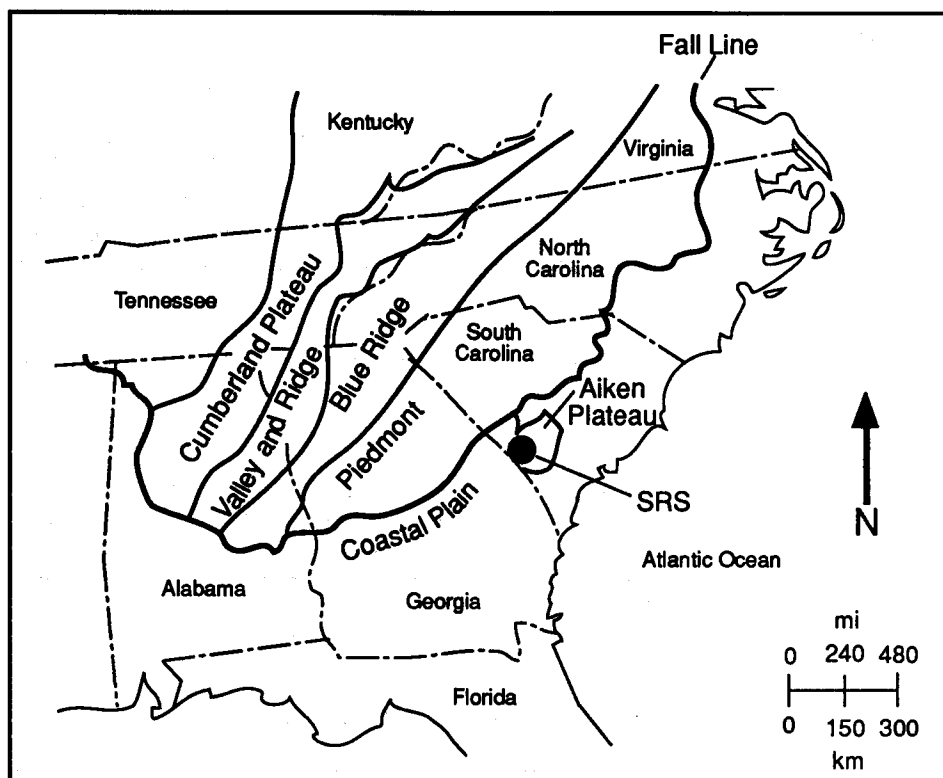


FIGURE D.12 Physiographic Provinces in the Vicinity of SRS (Source: Modified from DOE 1987)

(WSRC 1989a). With the exception of the Peedee Formation, this section is lithologically similar to its type section. These units were formerly described as the Tuscaloosa Formation, whose type locality is Late Cretaceous in age and not correlated via the fossil record to Cretaceous sediments in the southwestern South Carolina Coastal Plain. Generally, Lumbee Group deposits are variably colored sands and clays that become calcareous downdip from SRS (WSRC 1989a).

The Middendorf Formation has been correlated from its type locality to SRS by Nystrom and Willoughby (1982). It consists of moderately sorted quartz sand with occasional clay clasts. Over much of SRS, a clay zone up to 24 m thick caps the formation. The thickness of the Middendorf varies from 40 m at the northwestern SRS boundary to 55 m near the southeastern boundary (WSRC 1989a). At the proposed NPR site, the Middendorf Formation is approximately 43-46 m thick (WSRC 1989b).

The Black Creek Formation consists of moderately to poorly sorted quartz sands that grade upward into silts and clays. It is about 34 m thick at the northwestern SRS boundary and thickens to more than 76 m at the southeastern boundary (WSRC 1989a). The Black Creek has also been traced from its type locality (Colquhoun et al. 1983), where it unconformably overlies the Middendorf Formation and is approximately 88-95 m thick. At the proposed NPR site, the Black Creek consists of four zones: a lower sandy zone, an interbedded sand and clay zone, an upper sandy zone, and an upper clay zone (WSRC 1989b).

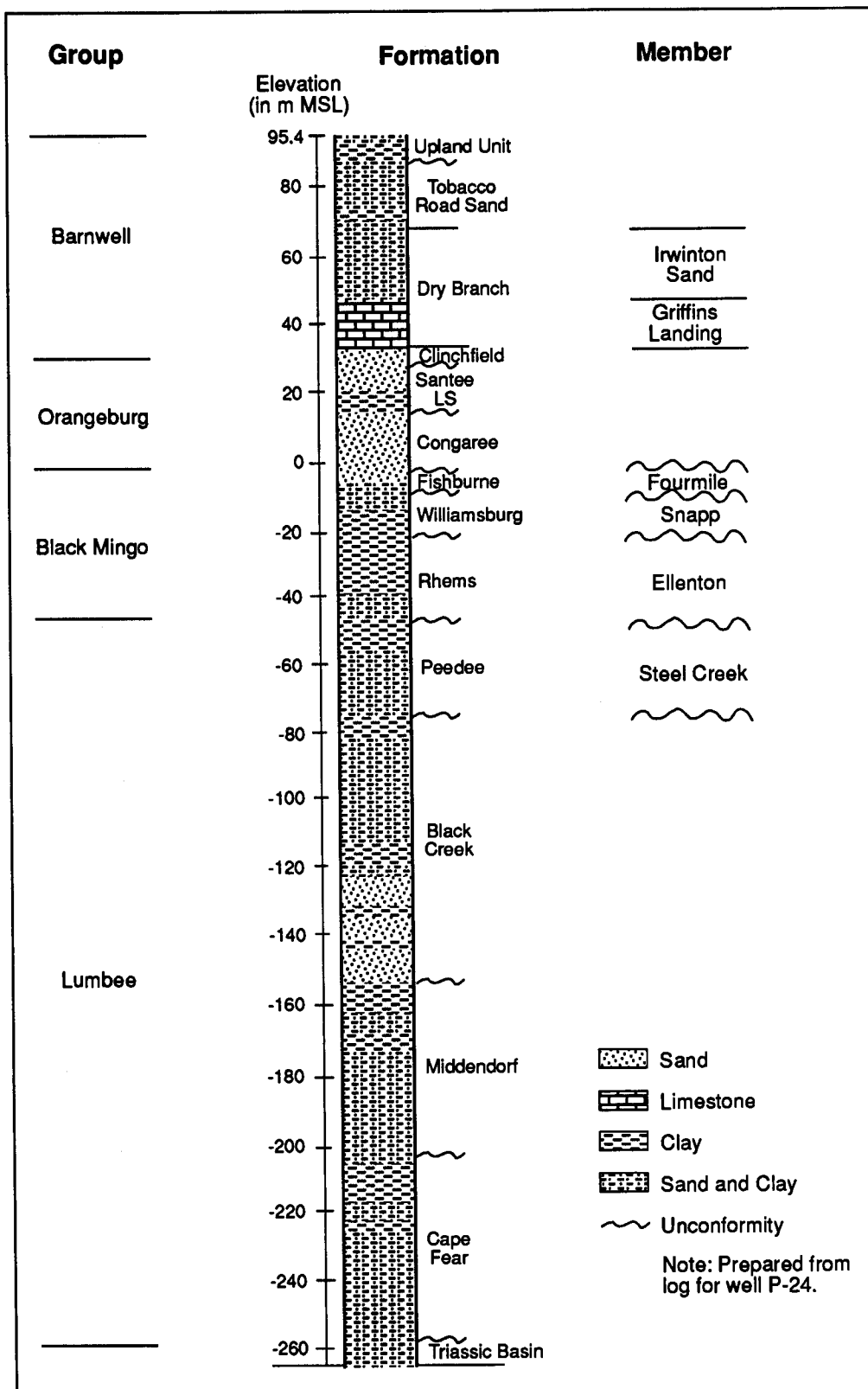


FIGURE D.13 Generalized Stratigraphic Column for the NPR Site at SRS (Source: Modified from WSRC 1989a)

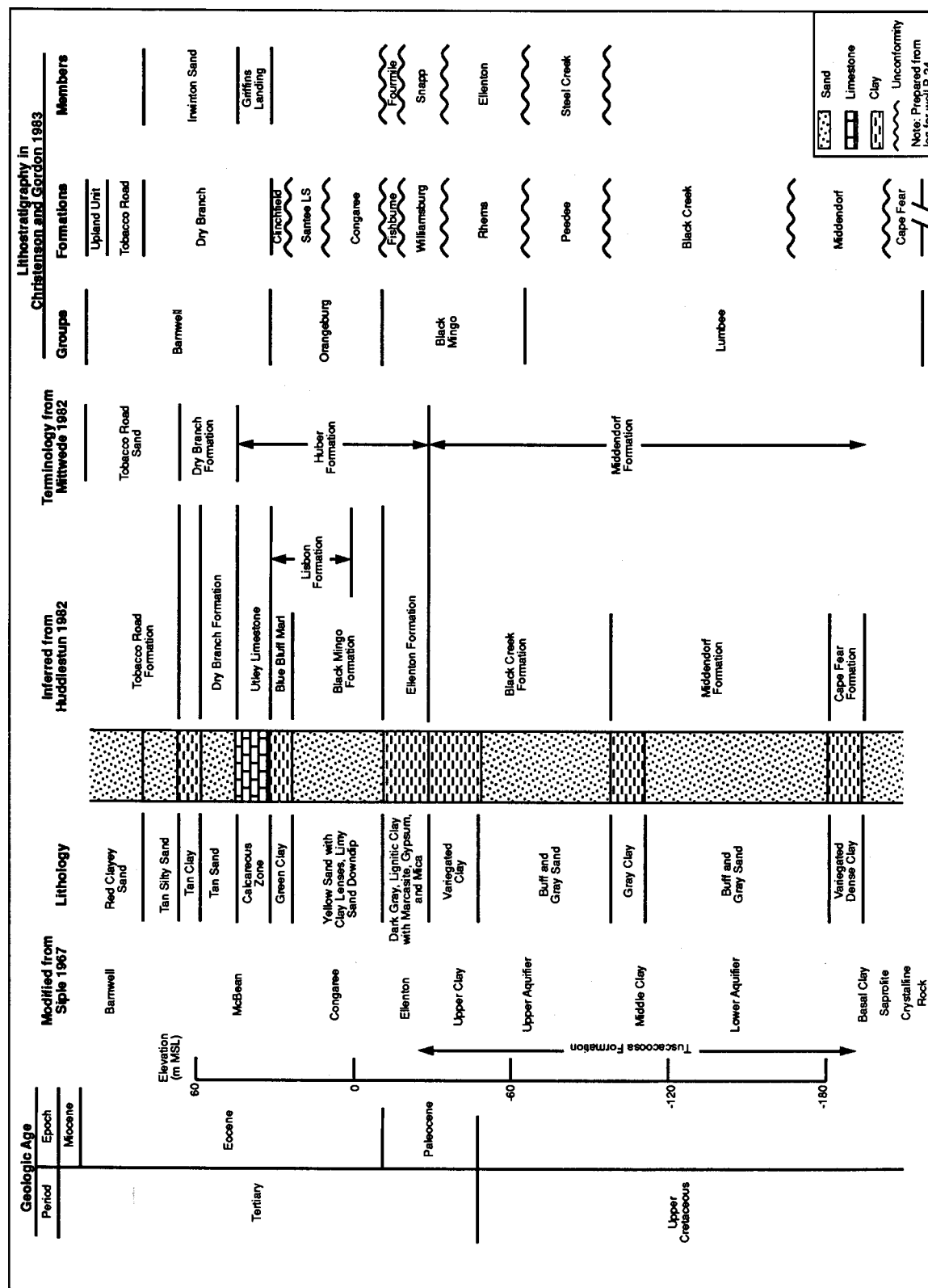


FIGURE D.14 Tentative Correlation of Stratigraphic Terms of the Southwestern South Carolina Coastal Plain (Source: Modified from Christenson and Gordon 1983)

The Peedee Formation, which contains marine fossils, consists of fine-grained sandstone and siltstone. Although it has been traced from its type locality to southwestern South Carolina (Colquhoun et al. 1983; Prowell et al. 1985), the lithologies at the type locality and SRS are different. Therefore, the time-equivalent facies at SRS is referred to as the Steel Creek Member of the Peedee Formation (WSRC 1989a). The Steel Creek Member consists of a sandy, pebble-rich zone that grades upward into a quartz sand and silty sand capped by clay. Portions of the member contain dinoflagellates, which indicates a marine affinity. The thickness of the Steel Creek Member ranges from 34 m at the northwestern SRS boundary to 40 m at the southeastern boundary (WSRC 1989a). At the proposed NPR site, the Steel Creek Member of the Peedee Formation unconformably overlies the Black Creek Formation. The Steel Creek Member is 21-26 m thick and consists of a lower sand and an upper clay (WSRC 1989b).

Overlying the Lumbee Group is the Paleocene-Eocene Black Mingo Group. This group comprises the Rhems, Williamsburg, and Fishburne formations, all of which have been traced in the subsurface to SRS (WSRC 1989a). The Rhems Formation, formerly designated the Ellenton Formation by Siple (1967), is called the Ellenton Member of the Rhems Formation because it is lithologically different from, but time-equivalent to, the Rhems (Colquhoun et al. 1983; Prowell et al. 1985). The Ellenton Member unconformably overlies Cretaceous units and consists of poorly sorted silty and clayey quartz sands interbedded with clays. It is approximately 12 m thick at the northwestern SRS boundary and thickens to about 30 m near the southeastern boundary (WSRC 1989a). The Ellenton Member is 18-21 m thick at the proposed NPR site.

The Williamsburg Formation is lithologically different from time-equivalent deposits at the type locality and has therefore been designated as the Snapp Member of the Williamsburg Formation (WSRC 1989a). The Snapp Member consists of a silty quartz sand interbedded with clay. It ranges from 9 m thick near the northwestern SRS boundary to 15 m near the southeastern boundary. The Snapp Member unconformably overlies the Ellenton Member and is approximately 9 m thick at the proposed NPR site (WSRC 1989b).

The Fishburne Formation is represented by the Fourmile Member, which is a moderately sorted sand with intermittent clay layers. The Fourmile Member is approximately 13 m thick in the central portion of SRS (WSRC 1989a).

The Middle Eocene Orangeburg Group, which overlies the Black Mingo Group, consists of the Congaree and Santee Limestone formations. The Congaree Formation consists of well-sorted quartz sands and has been traced to SRS from its type locality. The Congaree's time-equivalent unit has been mapped northwest of SRS as the Huber Formation (Nystrom and Willoughby 1982). The Congaree ranges from 18 m thick near the northwestern SRS boundary to 26 m thick near the southeastern boundary (WSRC 1989a). The Congaree Formation rests unconformably on the Fishburne Formation and is very similar lithologically. However, the Congaree lacks clay layering. At the proposed NPR site, it is 12-14 m thick (WSRC 1989b).

At SRS, the Santee Limestone Formation consists of lithologies varying from carbonates and calcareous quartz sands to quartz and glauconitic sands and clays. In other reports, this unit is referred to as the McBean and Lisbon formations. However, because of lithologic differences between these units at SRS and those at the type localities, the McBean Member of the Santee Limestone has been designated as the appropriate representation for the calcareous unit at SRS (WSRC 1989a). The glauconitic sands have been named the Warley-Hill Member, and the green clays are called the Caw-Caw Member. The stratigraphic sequence (from oldest to youngest) is Warley-Hill, Caw-Caw, and McBean. The thickness of the Santee Formation

ranges from 12 m near the northwestern SRS boundary to 24 m near the southeastern boundary (WSRC 1989a). It is 15 m thick at the proposed NPR site and varies in lithology from west to east across the site. In the western part, a basal sand unit is overlain by a glauconitic sand and clay unit, which is capped by the calcareous McBean Member of the Santee. The thickness of the unit in this area is about 6 m. In the central and eastern portions of the NPR site, where no carbonates have been found, a well-sorted sand is present (WSRC 1989b).

Late Eocene deposits of the Barnwell Group unconformably overlie the Orangeburg Group. The Barnwell Group consists of (from oldest to youngest) the Clinchfield, Dry Branch, and Tobacco Road Sand formations (Huddlestun and Hetrick 1985; WSRC 1989a). The Clinchfield Formation consists of a quartz sand that is sometimes calcareous. The Clinchfield at the proposed NPR site is approximately 2-5 m thick. In the western portion of the NPR site, the Clinchfield consists of a calcareous quartz sand. In the central and eastern portions of the site, however, no carbonate has been found, and the Clinchfield is a quartz sand (WSRC 1989b).

The Dry Branch Formation is composed of the Griffins Landing Member, which occurs sporadically, and the Irwinton Sand Member, which is a moderately sorted quartz sand with interlaminated clays. The Dry Branch Formation is about 15 m thick at the northwestern SRS boundary and increases to about 24 m thick at the southeastern boundary (WSRC 1989a). In the western portion of the NPR site, the Griffins Landing Member is overlain by a clay zone and the Irwinton Sand Member of the Dry Branch (WSRC 1989b). The Griffins Landing Member is absent in the central and eastern portions of the NPR site, and the Clinchfield is directly overlain by the Irwinton Sand, a moderately to well sorted sand (WSRC 1989b). At the NPR site, the Dry Branch is 21-24 m thick (WSRC 1989b).

The Tobacco Road Sand conformably overlies the Dry Branch Formation. The Late Eocene age of the Tobacco Road Sand has been established in Georgia (Huddlestun and Hetrick 1985), but parts of the formation may be younger (Colquhoun et al. 1983; Prowell et al. 1985; WSRC 1989a). The Tobacco Road Sand is a poorly to well sorted, medium to coarse sand with intermittent clay; it varies in thickness from 15 to 24 m at the NPR site.

The Upland Unit overlies parts of the Tobacco Road Sand Formation. "Upland Unit" is an informal stratigraphic term used to identify the deposits commonly occurring at higher elevations in the southwestern South Carolina Coastal Plain (Nystrom and Willoughby 1982; Nystrom, Willoughby, and Kite 1986). The sediments are poorly sorted, clayey to silty sands with lenses of conglomerate and clay. The lower surface is irregular because of erosion of the underlying Tobacco Road Formation before or during deposition of the Upland Unit. The age of the Upland Unit is uncertain but is postulated to be Miocene (Prowell et al. 1985; Nystrom, Willoughby, and Kite 1986). The maximum observed thickness at the proposed NPR site is about 23 m (WSRC 1989b).

Within the Upland Unit, and exposed at SRS roadside embankments and drainage ditches, are soil fractures associated with highly weathered zones in the upper soil horizons. These fractures have a leached appearance and contain clay minerals or, in some locations where infilling is present, bleached sands. The soil fractures at SRS lack a preferred azimuthal orientation and exhibit a wide range of dip angles. These fractures have not been reported to occur below the unsaturated zone.

The Aiken Plateau is characterized by interfluvial areas with narrow, steep-sided valleys. The SRS has more topographic relief than areas closer to the coast because of its proximity to the Savannah River and the Piedmont. On-site elevations range from 25.9 to 130 m MSL.

D.4.2 Geologic Structures

The Blue Ridge, the Valley and Ridge, and the Piedmont provinces are associated with the Appalachian orogen. Associated with these provinces are several fault systems located northwest of the Fall Line. In addition, several Triassic-Jurassic basins, which formed in response to tensional stresses in the early Triassic, have been identified in the Coastal Plain province of South Carolina and Georgia (Popenoe and Zietz 1977; Daniels, Zietz, and Popenoe 1983; WSRC 1989a). Table D.11 lists the significant geologic structures and fault systems in the region and gives the estimated age of last movement. Figure 4.3.1-1 shows the general locations of the structures that are within a 150-km radius of SRS.

The closest off-site fault system is the Belair Fault of the Augusta Fault Zone, which is approximately 40 km from SRS. Studies of this fault are inconclusive regarding movement in the last 35,000 years because the age of the oldest unfaulted sediments could be between 2,000 and 23,000 years (Prowell, O'Conner, and Rubin 1975; O'Conner and Prowell 1976; Case 1977). Case (1977) suggests that the Belair Fault is not "capable," citing the age relationship represented by the youngest faulted (65 million years) and oldest unfaulted sediments (2,000-23,000 years) and the absence of any correlation of macroseismicity with this fault zone.

Near Langley, South Carolina, the Langley Graben is associated with a pair of northwest-trending normal faults having an offset of about 1.2 m (Inden and Zupan 1975). This small graben is located about 27 km from SRS and is not considered to adversely affect on-site safety or seismic risk (du Pont 1980). There is no conclusive evidence of Recent (Holocene) displacement along any fault within 300 km of the site, with the possible exception of the geophysically inferred faults in the meizoseismal area of the 1886 Charleston earthquake, which occurred approximately 145 km from SRS (Lytle et al. 1979; Behrendt et al. 1981; Talwani 1982; du Pont 1982; Hamilton, Behrendt, and Ackermann 1983). Another possible on-site exception includes seismically inferred strike-slip or reverse motion on faults associated with the Dunbarton Basin (Stephenson, Talwani, and Rawlins 1985).

The Triassic Dunbarton Basin, a portion of which underlies SRS, was formed by laterally oriented extensional stresses that resulted in normal fault movement along a narrow, northeast-trending zone. In a geophysical study, Chapman and Di Stefano (1989) identified reverse motion on the Dunbarton Fault in the Late Cretaceous after the infilling, deposition, and erosion of Triassic sediments in the basin. In addition, they identified two faults transverse to the Dunbarton and one overlying the Dunbarton. Zoback, Moos, and Stephenson (1989) measured the in-situ state of stress in the crystalline basement rocks underlying the site and found that the maximum principal stress was oriented northeast-southwest. Reverse motion would occur preferentially on preexisting planes of weakness oriented perpendicular to this stress field (Zoback, Moos, and Stephenson 1989).

TABLE D.11 Geologic Structures in the SRS Region

Structural Feature	Closest Point to Site		Age of Last Movement
	Distance (km)	Direction	
Valley and Ridge province faults	350	NW	Late Paleozoic
Blue Ridge province faults (Cartersville, Whitestone, Fries-Hayesville-Altoona)	280	NW	Late Paleozoic
Cape Fear arch	250	NE	Pleistocene
Brevard fault zone	225	NW	Pre-Mesozoic
Westerfield fold-fault system	225	NE	Pre-Eocene
Deep River Basin (N.C. and S.C.)	215	NE	Triassic-Jurassic
Gold Hill Fault	210	NW	Late Paleozoic
Columbia Triassic Basin	155	NE	Pre-Cretaceous
Towaliga Fault, Kings Mtn. belt	135	NW	Late Paleozoic
Clubhouse Crossroads faults	115	SE	Pre-Miocene (?) ^a
Columbia reverse faults and clastic dikes	105	NE	Late Miocene
Charleston Triassic (?) Basin	180	SE	Triassic-Jurassic
Decatur-Coffee County (Ga.) graben and faults	65	SE	Pre-Paleocene
Eastern Piedmont fault system (Modoc, Flat Rock, Goat Rock, and Bartletts Ferry)	65	NW	Late Paleozoic
Augusta fault zone (Belair Fault)	40	NW	Pre-Miocene to Holocene ^b
Langley Graben	27	NW	Pre-Miocene (?)
Dunbarton Triassic (?) Basin	On-site	-	Pre-Late Cretaceous
Pen Branch Fault	On-site	-	Upper-Middle Eocene (?) ^c

^a(?) indicates uncertainty.

^bThe NRC has determined that, although the age of last movement is not precisely known, the Belair fault zone is not capable (Case 1977).

^cStudies to determine the age of the most recent movement on the Pen Branch Fault are ongoing.

Conversion factor: 1 km = 0.6214 mi.

Source: du Pont 1980.

The Pen Branch Fault (Figure 4.3.1-2) parallels the northwest-trending border fault of the Dunbarton Basin and may be the same fault as the one found paralleling the Dunbarton Basin border fault (Chapman and Di Stefano 1989). Snipes, Fallaw, and Price (1988) believe that the Pen Branch Fault is a growth fault and that normal movement began after deposition of Late Cretaceous sediments and continued intermittently through the early part of Late Eocene time. At this time, there is no evidence of strike-slip or oblique movement, although there is limited well control. The Pen Branch Fault is at least 32 km long, but its complete lateral extent is not yet defined (Snipes, Fallaw, and Price 1988). The South Carolina Geological Survey has tentatively mapped it as extending approximately 8 km east of the SRS boundary. Geophysical techniques and drilling are being used in an intensive investigation of the Pen Branch Fault recently initiated at SRS.

Investigations involving surface mapping, subsurface boring, and geophysical investigation have detected other surficial faults within Eocene-age strata at SRS (Figure 4.3.1-2) (du Pont 1980). These faults are generally less than 300 m long, and their displacements are less than 1 m. However, none of these faults is considered capable because the faults are overlain by younger sediments that show no evidence of faulting. Also, the faults exhibit no control over area topography or drainage.

D.4.3 Volcanism

There is no volcanic hazard at SRS. The SRS area has not experienced volcanism within the last 230 million years. Future volcanism is not expected because SRS is located along the passive eastern margin of the North American plate.

D.4.4 Seismicity

Two major earthquakes have occurred within 300 km of SRS. The first was the Charleston earthquake of 1886, which had an epicentral MMI of X and was located approximately 145 km from the site. An estimated peak horizontal acceleration of 0.10 g for the site during this quake has been calculated (URS/Blume 1982). The second major earthquake was the Union County, South Carolina, earthquake of 1913, which had an epicentral MMI of VII-VIII and was located about 160 km from the site (Langley and Marter 1973). Site intensities, accelerations, and available magnitudes for earthquakes of MMI V or larger are listed in Table D.12. Figure 4.3.1-3 shows the locations of the listed earthquakes that are within 150 km of SRS. Available data for earthquakes within the SRS region having MMIs larger than III or magnitudes of 2.0 or larger are presented in Table D.13. Figure 4.3.1-4 shows the locations of these listed earthquakes that occurred within 100 km of SRS. Earthquakes located in the Lake Thurmond area represent seismic events caused by elevated hydrostatic pressure resulting from increases in lake level (Talwani 1976).

During recent years, two earthquakes have occurred within SRS boundaries. An earthquake with a magnitude of 2.6 (maximum MMI of III) and a focal depth of 0.96 km occurred on June 8, 1985. The epicenter was just west of the C and K areas. The ground acceleration produced by this earthquake was less than 0.002 g. No aftershocks were recorded by the Site Seismic Network (Stephenson, Talwani, and Rawlins 1985). On August 5, 1988, an earthquake having a magnitude of 2.0 (maximum MMI of I-II) and a hypocentral depth of

TABLE D.12 Earthquakes within the SRS Region with MMI $\geq$ V

Date ^a	Location	Coordinates		Maximum Intensity	Distance from SRS (km)	Reported or Estimated Intensity at SRS	Magnitude	Estimated Acceleration at SRS (g)
		Lat. (°N)	Long. (°W)					
1811 Jan 13	Burke Co., Ga.	33.2	82.2	V	55	III-IV	NA ^b	0.02
1811-1812 (3 shocks)	New Madrid, Mo.	36.3	89.5	XI-XII	850	V-VI	NA	0.05
1875 Nov 02	Lincolnton, Ga.	33.8	82.5	VI	100	III-IV	NA	0.02
1886 Sep 02	Charleston, S.C.	32.9	80.0	X	145	VI	NA	0.10
1886 Oct 22	Charleston, S.C.	32.9	80.0	VII	155	III-IV	NA	0.02
1897 May 31	Giles, Co., Va.	37.3	80.7	VIII	455	III	NA	0.01
1912 Jun 12	Charleston, S.C.	33.0	80.2	VII	135	III-IV	NA	0.02
1913 Jan 01	Union Co., S.C.	34.7	81.7	VII-VIII	160	IV	NA	0.02
1920 Aug 01	Charleston, S.C.	33.1	80.2	VII	135	III-IV	NA	0.02
1972 Feb 03	Bowman, S.C.	33.5	80.4	V	115	IV	4.5	0.02
1974 Aug 02	Willington, S.C.	33.9	82.5	VI	105	IV	4.1	0.02
1974 Nov 22	Charleston, S.C.	32.9	80.1	VI	145	III-IV	4.3	0.02

^aBased on Greenwich mean time.

^bNA = data not available.

Conversion factor: 1 km = 0.6214 mi.

Source: DOE 1987.

TABLE D.13 Earthquakes within the SRS Region with MMI $\geq$ IV or Magnitude $\geq$ 2.0

Date ^a	Coordinates		Maximum Intensity	Distance from SRS (km)	Reported or Estimated Intensity at SRS		Magnitude	Estimated Acceleration at SRS (g)
	Lat. (°N)	Long. (°W)						
1811 Jan 13 ^b	33.2	82.2	V	55	III-IV		NA	0.02
1853 May 20	34.0	81.2	VI	102	NA		NA	NA
1945 Jul 26	33.8	81.4	V	77	NA		4.4	NA
1964 Mar 07	33.7	82.4	NA ^c	85	NA		3.3	NA
1964 Apr 20	33.8	81.1	V	96	NA		3.5	NA
1968 Sep 22	34.1	81.5	IV	102	NA		3.5	NA
1972 Aug 14	33.2	81.4	NA	27	NA		3.0	NA
1974 Oct 28	33.8	81.9	IV	72	NA		3.0	NA
1974 Nov 05	33.7	82.2	III	77	NA		3.7	NA
1976 Sep 15	33.1	81.4	NA	25	NA		2.5	NA
1977 Jun 05	33.1	81.4	NA	35	NA		2.7	NA
1982 Jan 28	32.9	81.4	NA	40	NA		3.4	NA
1985 Jan 08	33.2	81.7	III	On-site	III		2.6	NA
1988 Feb 17 ^d	33.6	81.7	III	45	NA		2.6	NA
1988 Aug 05	33.1	81.4	NA	On-site	II		2.0	NA

^aBased on Greenwich mean time.^bLocated in Burke County, Ga.^cNA = data not available.^dLocated at Aiken, S.C.

Conversion factor: 1 km = 0.6214 mi.

Sources: Visvanathan 1980; NOAA and USGS 1982; Stephenson 1988; Sibol and Bollinger 1989.

2.68 km occurred. The epicenter was located just northeast of K Area. Although the 1988 quake was not felt on-site, it was recorded by seismometers within 100 km of the site.

The hazards associated with seismic events at SRS are partly functions of magnitude, frequency, and attenuation. Bollinger et al. (1989) constructed a seismic event recurrence relation for the southeastern United States (Figure D.15) that predicts that approximately eight earthquakes of magnitude 2.5 are likely to occur per year in the region. Stephenson (1988) cites a number of recurrence relations for the region. Although based on different data sets, all of them predict several magnitude 2 (MMI intensity II) events per year in the South Carolina Coastal Plain. However, data of the type needed to construct recurrence rates are sparse, especially for the larger-magnitude events. For a return period of 10,000 years, the regression equation of Bollinger et al. (1989) gives an approximate magnitude of 8.4, but the maximum attainable magnitude may be less than this at SRS.

D.4.5 Soils

Soils close to the proposed NPR site at SRS are dominated by sands and sandy loams. Figure D.16 is a soils map for SRS (USDA 1990). The NPR site is dominated by the Fuquay-Blanton-Dothan series, which contains small inclusions of similar soils and several small (4,047- to 20,235-m²) areas of the Rembert series. Bordering the dominant Fuquay-Blanton-Dothan series at the NPR site is the Vacluse-Ailey complex, which is characterized by intermingled, loamy textured soils underlain by a brittle subsoil. Table D.14 describes the soils at the NPR site, including their engineering characteristics. Two members of the Dothan series -- Dothan A slope and Dothan B slope -- are classified as Prime Farmland (USDA 1990). None of the NPR site is classified as Unique Farmland, although some SRS areas are producing forest products.

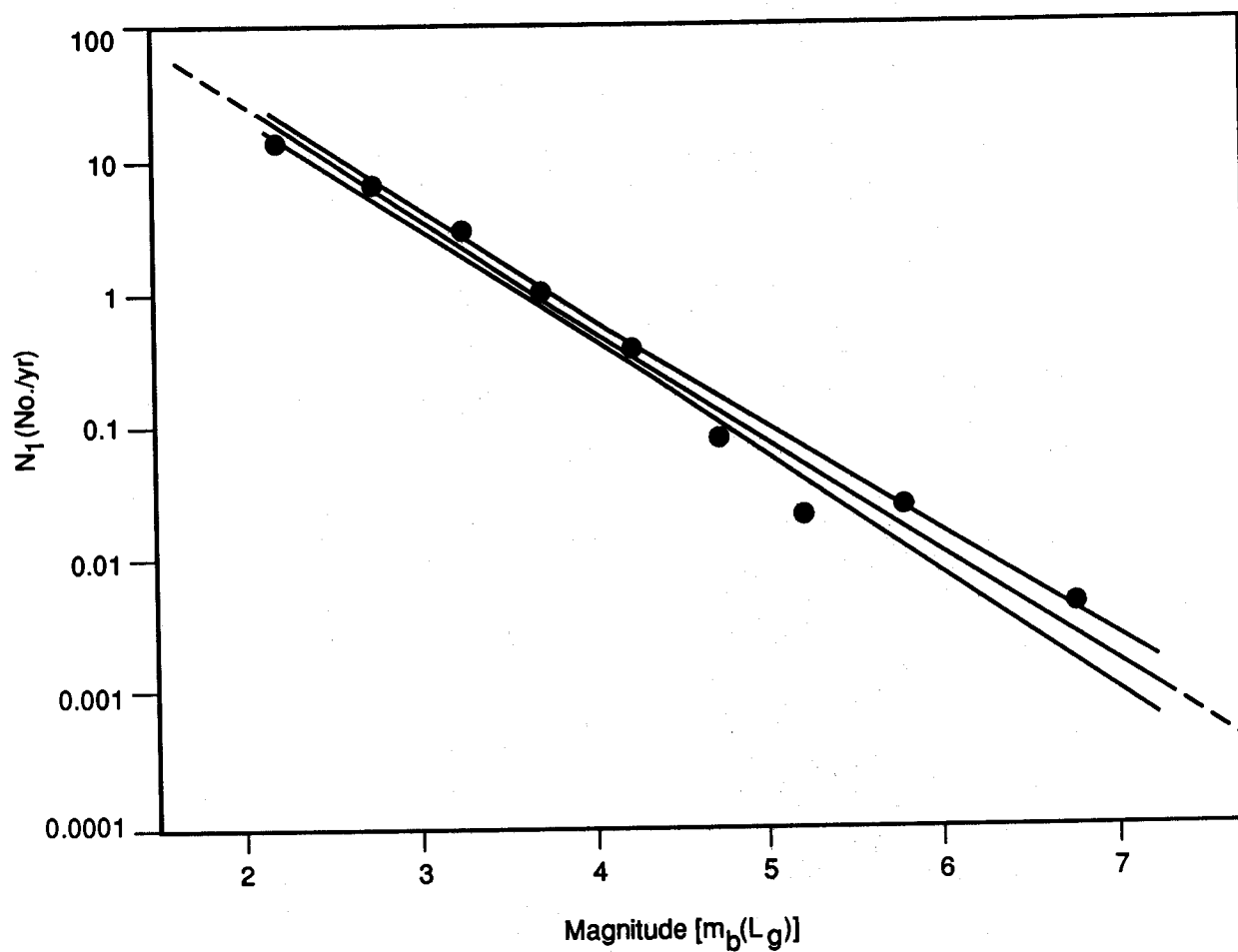


FIGURE D.15 Seismic Magnitude Recurrence Relationships for the Southeastern United States (plot derived from equation: $\log(N_1) = 3.126 \pm 0.024 m_b(L_g)$; maximum likelihood line includes 95% confidence intervals) (Source: Bollinger et al. 1989)

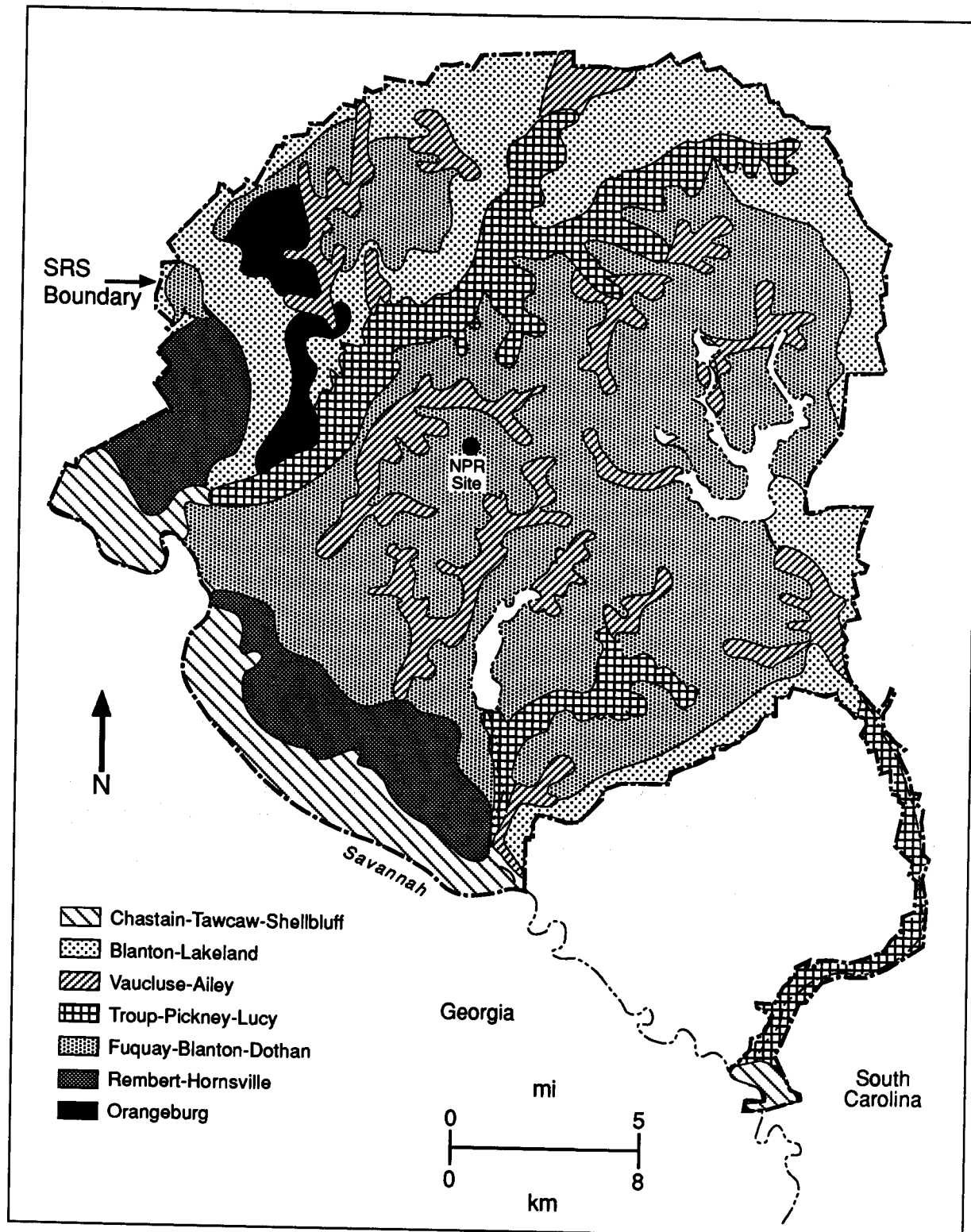


FIGURE D.16 Soils Map of SRS (Source: USDA 1990)

TABLE D.14 Characteristics of Soils at the Proposed NPR Site

Soil	Description	Depth (cm)	Erosion Potential	Available Water Capacity ^a (meters/meter of soil)	Permeability Range ^b (m/s)
Fuquay	Sand	0-56	Slight	4.19	4.23×10^{-7} - 1.41×10^{-6} (89-152 cm)
	Sandy loam to sandy clay loam	56-152			
Blanton	Sand or loamy sand	0-152	Slight	2.52	4.23×10^{-6} - 1.41×10^{-5} (178-203 cm)
	Sandy loam or sandy clay loam	152-203			
Dothan	Sand	0-18	Slight	7.24	1.41×10^{-6} - 4.23×10^{-6} (30-165 cm)
	Sandy clay loam	18-165			
Vauluse	Sandy loam	0-13	Moderate	6.73	4.23×10^{-7} - 4.23×10^{-6} (51-157 cm)
	Sandy clay loam	13-157			
	Sandy loam to loamy clay	157			
Ailey	Sand	0-66	Moderate	4.11	4.23×10^{-7} - 1.41×10^{-6} (102-165 cm)
	Sandy loam to sandy clay loam	66-140			
	Coarse sandy loam to sandy clay loam	140-165			
Rembert	Sandy loam	0-25	None to slight	7.65	4.23×10^{-7} - 1.41×10^{-6} (25-74 cm)
	Clay	25-74			
	Sandy loam to sandy clay loam	74-152			

^aAvailable water capacity of SRS soils computed for a 152-cm profile, beginning at the surface.^bPermeability (determined at saturation) given reflects the range for the most restrictive layer. Numbers in parentheses represent range of layer as depth in centimeters from the surface.

Source: USDA 1990.

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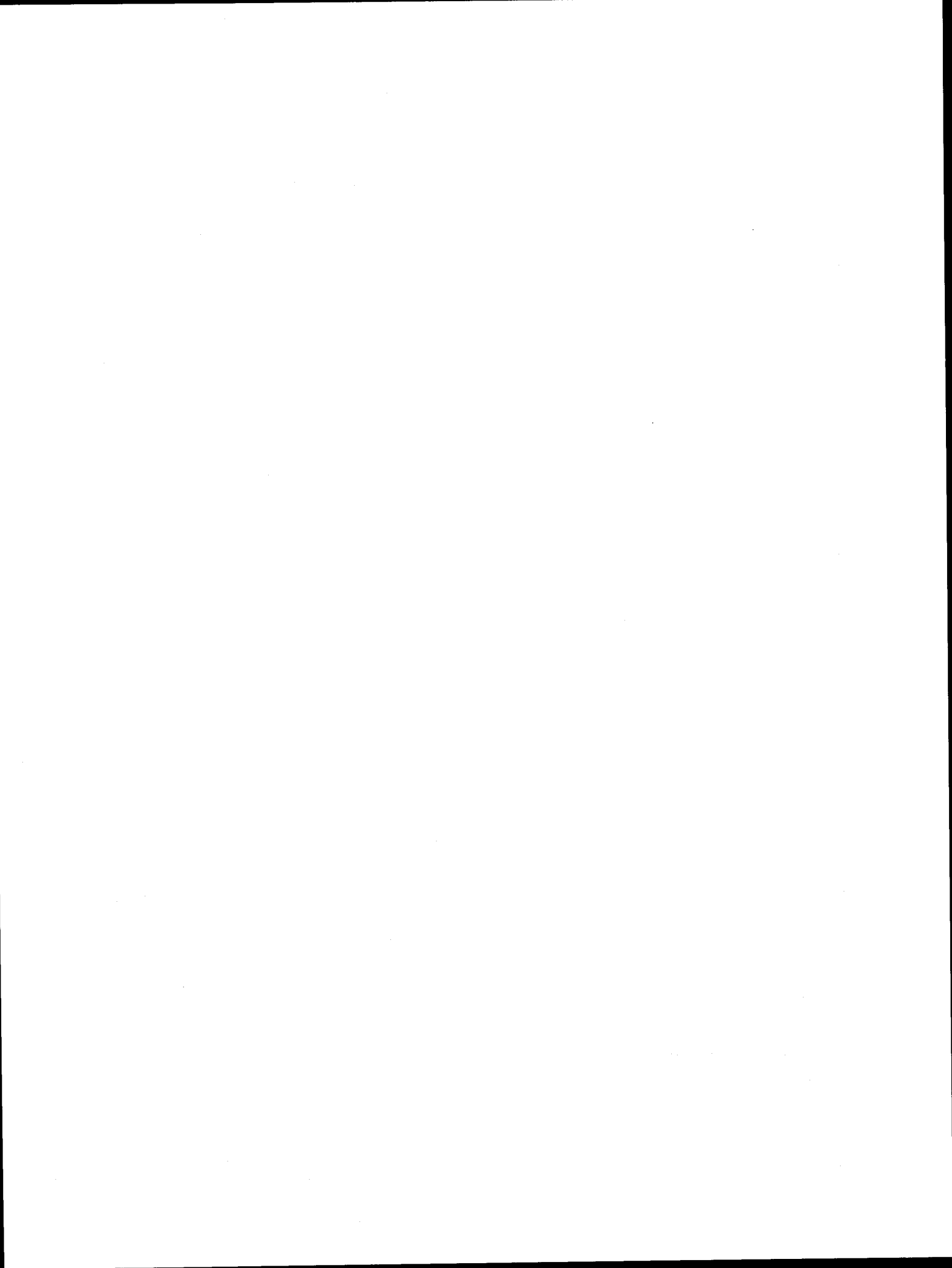
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APPENDIX E:
AIR RESOURCES AND NOISE



APPENDIX E: AIR RESOURCES AND NOISE

This appendix provides detailed information in support of the descriptions and evaluations presented in the air resources sections of the EIS.

E.1 AFFECTED ENVIRONMENT

Tables E.1 - E.15* provide detailed information in support of the descriptions of the affected environment in Sections 4.1.2, 4.2.2, and 4.3.2. The tables present applicable ambient air quality standards, existing ambient air quality data, and emission inventories. Figures E.1 - E.3 show the locations of air quality monitoring stations at and near Hanford, INEL, and SRS.

E.2 ENVIRONMENTAL CONSEQUENCES

This section provides additional information in support of the conclusions on environmental consequences related to ambient air quality, cooling tower operation, and noise, as presented in Sections 5.2.1, 5.3.1, and 5.4.1.

E.2.1 Air Quality Impacts

Potential environmental impacts of air pollutant emissions from activities associated with construction and operation of the NPR and its support facilities at the three candidate sites are described in this section. A detailed analysis was conducted of the potential ambient air quality impacts of the emissions from NPR construction and operation activities for the three NPR technologies at Hanford, INEL, and SRS (PNL 1991; EG&G 1991; NUS 1991). The air quality modeling analysis used (1) the Industrial Source Complex-Short Term (ISCST) model (EPA 1986a) recommended by EPA for point and area sources and (2) the CALINE3 model developed by the California State Department of Transportation for the Federal Highway Administration (CASDT 1979) for traffic sources. The ISCST model estimates dispersion of emissions from stack (point), area, and volume sources. It estimates the dispersion from stack and volume sources using a steady-state Gaussian plume equation and estimates dispersion from area sources using the equation for a continuous and finite cross-wind line source. The CALINE3 model also uses the finite cross-wind line source concept for each of a series of elements of an individual roadway link. (A finite cross-wind line source means that the emissions from an area source or a roadway element are represented by the same quantity of emissions condensed into a line, which is assumed to be perpendicular to wind direction.)

The performance of ISCST model has been evaluated with field data for its point source submodel (EPA 1977; Bowne and Longran 1983, 1985; Murray and Bowne 1988) and for its special features, such as gravitational settling/dry deposition option (Bowers and Anderson 1981;

*For sake of readability, all tables are presented at the end of textual portion of this appendix.

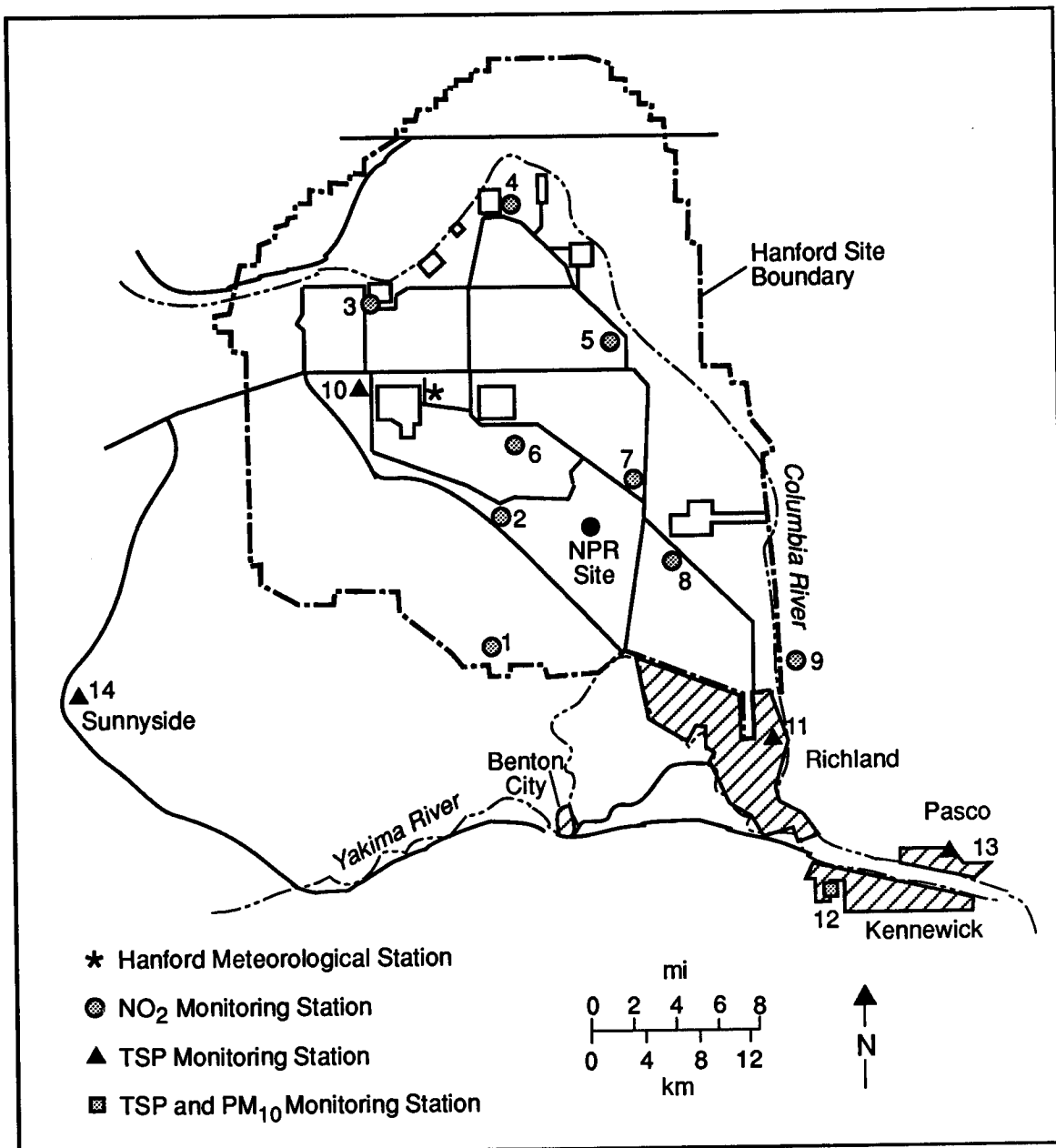


FIGURE E.1 Ambient Air Quality Monitoring Stations in and near Hanford, 1987-1988 (stations 1-10: Pacific Northwest Laboratory [stations 8-10 discontinued in 1988]; stations 11-14: Washington Department of Ecology) (Source: Modified from PNL 1990)

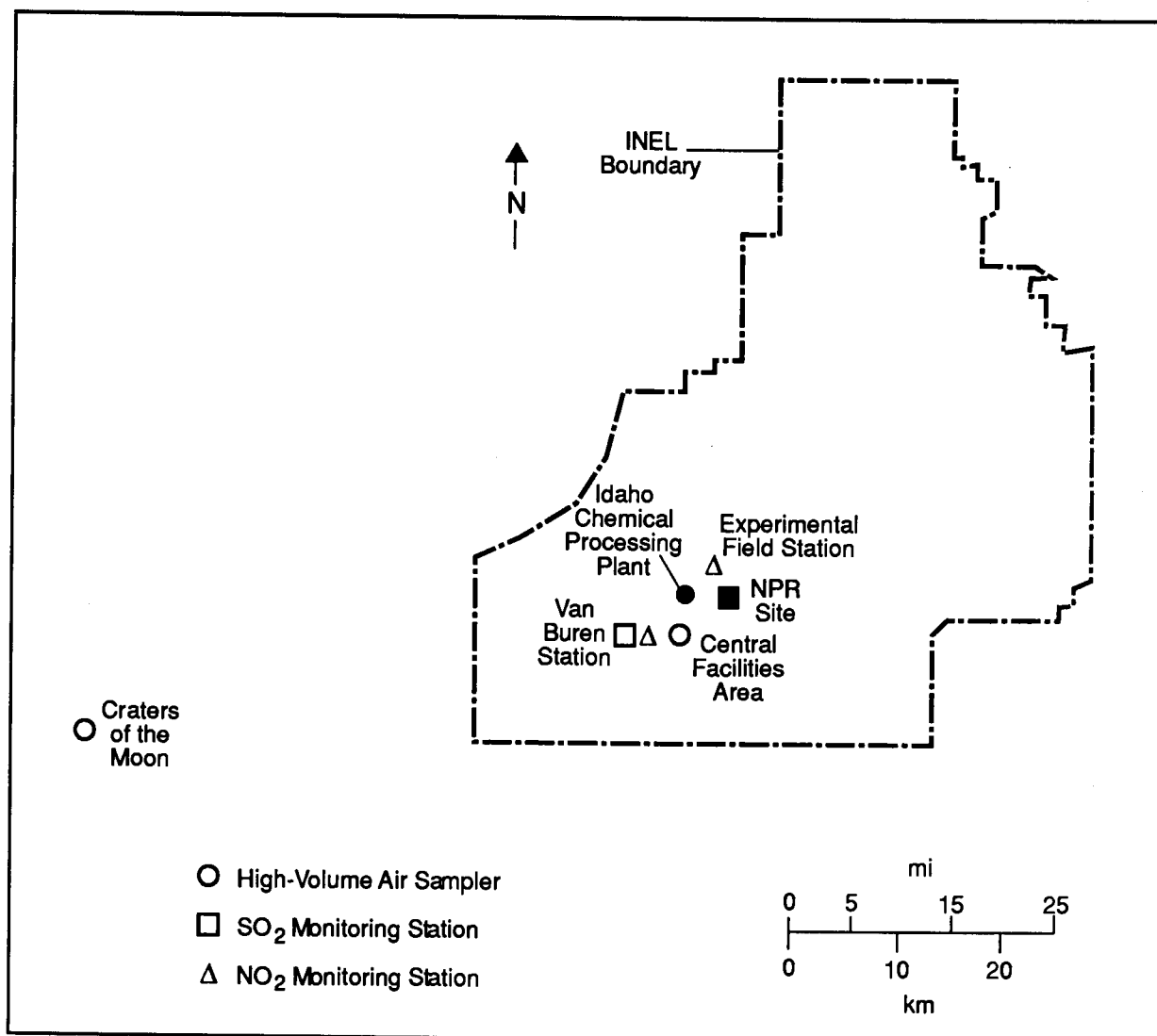


FIGURE E.2 Ambient Air Quality Monitoring Stations in and near INEL (Source: Modified from Hoff, Mitchell, and Moore 1989)

Bowers et al. 1982) and building downwash option (Bowers and Anderson 1981; Schulman and Hanna 1986). From the validation studies for the point source model (the CRSTER model) based on field data measured at four large power plants, it was concluded that the model acceptably predicts the upper percentile of the frequency distribution of 1-hour concentrations and of the corresponding distributions of 24-hour concentrations. Concentrations over the remainder of the frequency distributions are significantly underpredicted. The second-highest 1-hour concentrations were predicted within a factor of two at two-thirds of the field sampling sites for elevated power plant plumes. The second-highest 24-hour concentrations tended to be underpredicted by the model, with the ratio of predicted concentration to measured concentration ranging from about 0.2 to 2.7 at about 90% of the sampling sites (EPA 1977).

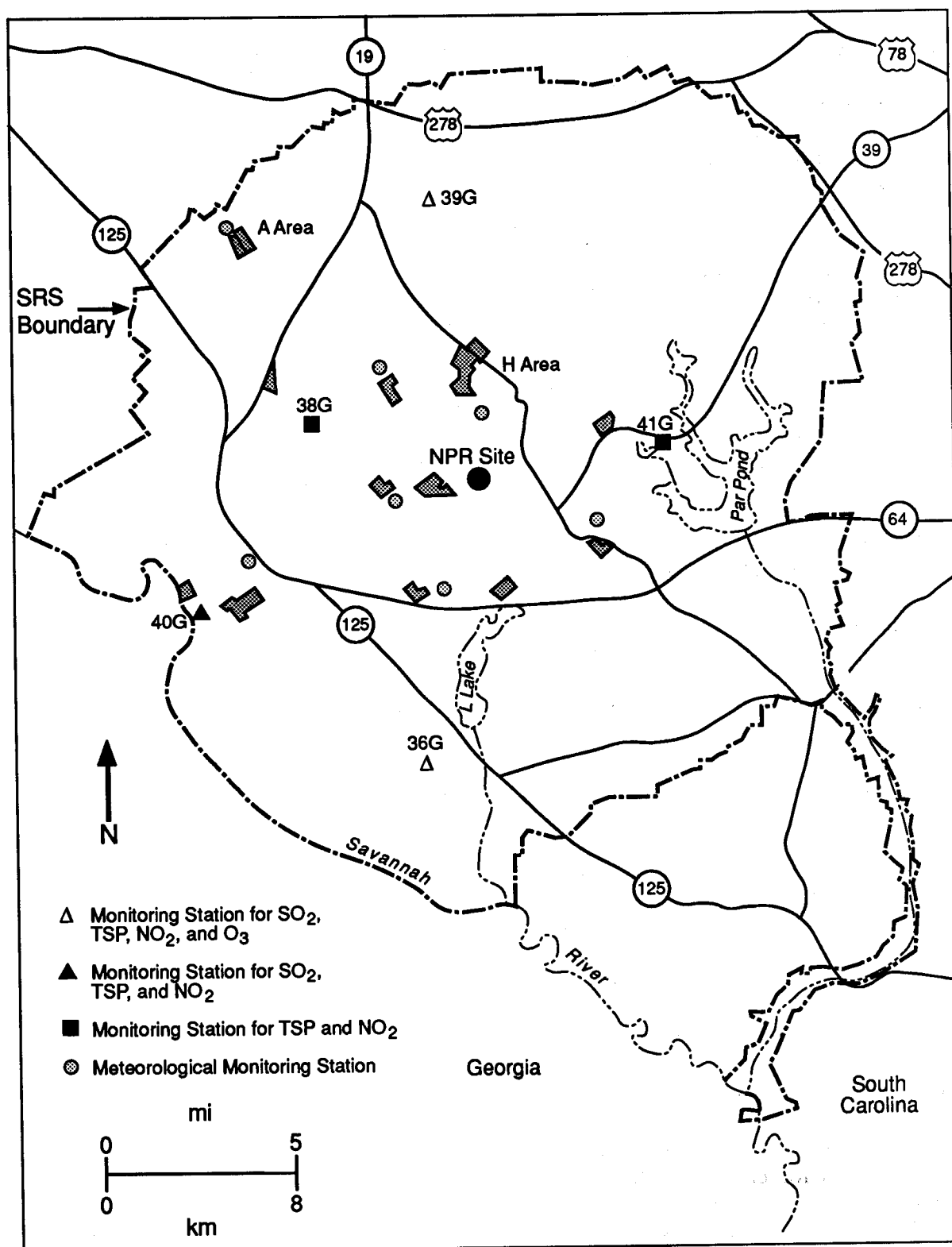


FIGURE E.3 Ambient Air Quality and Meteorological Monitoring Stations at SRS (Source: Modified from NUS 1991)

In other validation studies for the point source model, the CRSTER model predicted peak short-term (1-, 3-, and 24-hour) concentration values within 30-70% at a plain site (Bowne and Longran 1983). The CRSTER model predicted peak 1-hour concentrations within 2% and underpredicted peak 3-hour concentrations by about 30% at a moderately complex terrain site (Bowne and Longran 1985). The ISCST model overpredicted 1-hour concentrations by about 60% with better predictions for longer time periods at an urban site (Murray and Bowne 1988). Uses of gravitational settling/dry deposition and building downwash options were found to improve the model performance significantly over that of the model without such features (Bowers and Anderson 1981; Bowers et al. 1982; Schulman and Hanna 1986).

The CALINE3 model was validated with field data obtained from three separate locations: one in Michigan and two in California (CASDT 1979). In the validation study, it was concluded that the model can reliably predict pollutant concentrations for receptors located within 150 m of the roadway. The correlation coefficient for linear least-square regression of predicted versus measured concentrations was about 0.9 or greater for unstable and neutral atmospheric stability conditions, and about 0.7 or greater for stable conditions. The model tends to overpredict for cases with stable atmospheric conditions and for cases with wind directions parallel to the roadway. However, the ratio of predicted concentration to measured concentration was always less than three.

The ISCST modeling was performed in accordance with EPA's guideline on air quality modeling (EPA 1986b). The model input data included the emission inventories for the baseline, construction, and operation sources and for one full year of on-site, hourly meteorological data. The worst-case emission rates were assumed for the baseline, construction, and operation periods. These worst-case emission rates represented (1) full-capacity operation for the baseline sources, (2) peak construction activities, and (3) full-capacity operation for the new NPR sources under the production mode that results in the highest emission rate for each pollutant.

E.2.1.1 Hanford Site

Major sources of air pollutant emissions during NPR construction and operation are described in Section 5.1.2. Potential air quality impacts of the emissions from construction and operation of the NPR and its support facilities at Hanford were analyzed with the computer codes described in Section E.2.1. The model input data included the emission inventories for the baseline, construction, and operation sources summarized in Table E.16 (see Tables E.17 and E.18 for the emission rates during construction and operation by source category) and one full year (1986) of on-site, hourly meteorological data. The 1986 data set had the worst diffusion characteristics of the 7 years (1983-1989) of data examined (PNL 1991). The model options selected are described in PNL (1991). The short-term emission rates for criteria pollutants during the peak construction period are estimated to be the same for the three NPR technologies. However, the duration of peak construction activities for the LWR technology would be substantially shorter because construction of WNP-1 is 63% complete (WHC 1990). Although the emission rates for criteria pollutants during the NPR operation period would be low compared with the baseline emissions at Hanford, those for the construction period would be comparable in magnitude with the existing emissions, except for sulfur dioxide (SO₂), nitrogen oxides (NO_x), and volatile organic compounds (VOC) (Table E.16). However, the annual emission rates for the construction period are overestimated because the peak construction activities were assumed to continue throughout the year. The NO_x emission rates for the new stationary sources for the HWR (57.1 t/yr) and the MHTGR (101 t/yr) are greater than the

corresponding prevention of significant deterioration (PSD) significant emission rate (40 tons/yr, or 36.3 t/yr) (Table E.18). This means that the construction of HWR and MHTGR reactors would be considered major modifications subject to PSD regulations and best available control technology requirements. Estimated new emissions of toxic air pollutants from NPR support facilities are in a similar order of magnitude as the baseline emissions for acetone (C_3H_6O), ammonia (NH_3), and 1,1,1-trichloroethane (TCA). Emissions of Freon-113, nitric acid (HNO_3), and trichloroethylene (TCE) would be new additions for the Hanford Site (see Table E.16).

Maximum increases in ambient concentrations of criteria pollutants estimated for the Hanford boundary and along SR 240, a public highway that traverses the site (Figure 3.2-6), are given in Table E.20 for the baseline, baseline plus construction, operation, and baseline plus operation emission sources. Most of the highest increases in short-term pollutant concentrations due to NPR construction or operation sources for the HWR and MHTGR would occur along SR 240 west-southwest of the NPR sources. The highest increases in pollutant concentrations for the LWR were estimated to occur along the eastern site boundary east to east-southeast of the NPR sources.

The ambient concentration levels of SO_2 , nitrogen dioxide (NO_2), and carbon monoxide (CO) calculated from the baseline, baseline plus construction, and baseline plus operation source emissions are all small fractions of applicable national and state ambient air quality standards for all averaging time periods. These levels would remain well below the standards after the addition of background concentrations (see Table E.20). The ambient concentration levels of PM_{10} * and total suspended particulate (TSP) calculated from the baseline, baseline plus construction, and baseline plus operation source emissions are also small fractions of applicable ambient standards for the annual period. These levels are expected to remain well below applicable ambient standards after the addition of the background concentrations. The ambient levels of 24-hour PM_{10} and TSP concentrations that would result from the baseline and baseline plus operation source emissions are small fractions of applicable standards. Those that would result from the baseline plus construction source emissions are larger fractions of the standards, but they would not exceed the standards by themselves. After the addition of background levels, the 24-hour PM_{10} concentrations would remain below applicable ambient standards, but the 24-hour TSP concentrations could exceed the state's ambient standard ($150 \mu g/m^3$). However, such exceedances would only occur occasionally, when the background TSP concentration level was very high. (The background 24-hour TSP concentrations at the Hanford Site are as high as $356 \mu g/m^3$ and therefore exceed the applicable ambient standard by themselves.)

The PM_{10} and TSP emissions rates for the NPR construction activities listed in Table E.17 are based on the assumption that, according to normal construction industry practice, water would be sprayed to reduce by half the fugitive dust emissions from construction activities, such as site preparation and excavation. The frequency of water spraying would be increased during extremely dry and windy conditions to minimize fugitive dust emissions from construction activities.

The projected increases in traffic along the major roadways leading to the proposed NPR site at Hanford both during NPR construction and operation would contribute to the ambient air pollutant concentrations. The results of air quality modeling conducted with the CALINE3

* PM_{10} is particulate matter with a diameter $\leq 10 \mu m$.

model indicate that the maximum emissions contributions from increased traffic volume would occur along SR 240 during the HWR construction period (PNL 1991). The maximum estimated ambient concentration contributions from the increased traffic volume are all small fractions of applicable ambient standards and would not significantly contribute to any exceedance of the standards (PNL 1991). These maximum increases due to the increased traffic volume are estimated to occur along SR 240 at the southeastern Hanford boundary.

Electric power requirements at Hanford are estimated to be about 10 MW during construction and about 100 MW during operation for the three NPR technologies and their support facilities. An NPR facility based on the LWR or MHTGR technology would be a net producer of electricity during operation, that is, more electricity would be produced than consumed. Electric power needed (1) during NPR construction, (2) during operation of the HWR, and (3) during operation of the LWR or MHTGR while the reactor's electric generating system is out of service would be purchased from the Bonneville Power Administration (BPA) system. The BPA obtains 75% of its power (15,000 MW) from hydroelectric plants, 10% from coal-fired power plants, and the remaining 15% from miscellaneous sources (e.g., nuclear power plants and imports). Thus, only a small portion of the electrical power requirements for the NPR would be met by regional coal-fired power plants. Emissions associated with meeting the relatively small power demands of the NPR would be released over a large area and would have little effect on ambient air quality at any location.

Combustion of fossil fuels during NPR construction and operation activities would result in emissions of carbon dioxide (CO_2), a greenhouse gas contributing to global warming. The highest estimated CO_2 emission rates during the NPR construction and operation periods are on the order of 10^4 t/yr for an LWR during construction and 10^6 t/yr for an HWR during operation at Hanford (PNL 1991). These emission rates correspond to about one-six-millionth and one-twenty-thousandth, respectively, of the estimated annual, global CO_2 emission rate due to fossil fuel combustion (Rotty, Marland, and Treat 1984).

The estimated increases in ambient concentrations of toxic air pollutants (maximum 24-hour concentration at the Hanford boundary) resulting from emissions from proposed new sources are given in Table E.21 by reactor technology. These maximum 24-hour concentrations (baseline and baseline plus new NPR emissions) are all substantially less than the corresponding ambient standards established by the Washington State Department of Ecology (WDOE 1991).

Table E.21 shows that small amounts of ozone (O_3) depleting gases (Freon-113 and TCA) would be emitted from the NPR support facility process operations. However, production of these O_3 -depleting gases would be phased out before or shortly after the start of NPR operation, in the year 2000. Industry is developing substitute materials for these gases, and when suitable substitutes become available, the O_3 -depleting gases would be replaced.

E.2.1.2 Idaho National Engineering Laboratory

Major sources of air pollutant emissions during NPR construction and operation are described in Section 5.1.2. Potential ambient air quality impacts of the emissions due to NPR construction and operation for the three NPR technologies at INEL were analyzed with the computer codes described in Section E.2.1 (EG&G 1991). The model input data included the emission inventories for the baseline, construction, and operation sources summarized in Table E.22 (see Tables E.23 and E.24 for the emission rates during construction and operation by

source category) and one full year (1986) of on-site, hourly meteorological data. Among 7 years (1982-1988) of data, the 1986 data set exhibited the worst diffusion characteristics (EG&G 1991). Model options selected are described by EG&G (1991). The short-term emission rates for criteria pollutants during the peak construction period (about 18 months) are estimated to be similar for the three NPR technologies. The duration of peak construction activities would be substantially shorter if an alternative means of disposing of the cooling tower blowdown is selected and a large evaporation pond is eliminated from the plans. Although the emission rates for criteria pollutants during the NPR operation period would be low compared with baseline emissions at INEL, those for the construction period would be comparable in magnitude to the existing emissions, except for SO₂ (Table E.22). However, the annual emission rates for the construction period are overestimates because peak construction activities were assumed to continue throughout the year. The SO₂ emission rate for an HWR during the operation period would be higher than those for the other two NPR technologies because of the emissions from six continuously operating diesel generators (Table E.24). The NO_x emission rates for the new stationary sources for the MHTGR (101 t/yr) and the HWR (75.2 t/yr) would be greater than the PSD significant emission rate for NO_x (40 tons/yr, or 36.3 t/yr) (Table E.24). This means that construction of the HWR or MHTGR would be considered a major modification subject to PSD regulations and best available control technology requirements. New emissions of toxic air pollutants from the NPR support facilities would be similar in order of magnitude to the baseline emissions for NH₃ and TCA. Emissions of HNO₃ would be substantially larger than the baseline emissions, and the emissions of C₃H₆O and Freon-113 would be new additions at INEL (Table E.26).

Maximum increases in ambient concentrations of criteria pollutants estimated for the INEL boundary and along US 26/20, a public highway that traverses the southern portion of INEL (Figure 3.2-8), are given in Table E.25 for the baseline, baseline plus construction, operation, and baseline plus operation emission sources. The highest increase in pollutant concentrations due to NPR construction or operation sources would occur along US 26/20, southwest of the proposed NPR site.

The calculated ambient concentration levels of SO₂, NO₂, and CO resulting from the baseline, baseline plus construction, and baseline plus operation source emissions are all small fractions of applicable national and state ambient air quality standards for all averaging time periods. These levels would remain well below the standards after the addition of background concentrations (see Table E.25). The ambient concentration levels of PM₁₀ and TSP resulting from the baseline, baseline plus construction, and baseline plus operation source emissions are also calculated to be small fractions of applicable ambient standards for the annual period. These levels are also expected to remain well below applicable ambient standards after the addition of background concentrations. The estimated ambient levels of 24-hour PM₁₀ and TSP concentrations resulting from the baseline and baseline plus operation source emissions are also small fractions of applicable standards. However, the concentrations resulting from the baseline plus construction source emissions are larger fractions of the standards, but by themselves would not exceed the standards. After the addition of background levels, the 24-hour PM₁₀ concentrations would remain below applicable ambient standards, but the 24-hour TSP concentrations could exceed the state's ambient standard (150 µg/m³). (The background 24-hour TSP concentrations at INEL site are estimated at 88 µg/m³.) However, such exceedances would only occur occasionally, during periods when the background TSP concentration level was very high.

The PM_{10} and TSP emissions rates for the NPR construction activities listed in Table E.23 are based on the assumption that, in accordance with normal construction industry practice, water would be sprayed to reduce by half the fugitive dust emissions from construction activities, such as site preparation and excavation. The frequency of water spraying would be increased during extremely dry and windy conditions to minimize the fugitive dust emissions during construction.

Increases in traffic along the major roadways leading to the proposed NPR site at INEL during the NPR construction and operation periods would contribute to the ambient air pollutant concentrations. The results of air quality modeling conducted with the CALINE3 model and the projected increases in traffic volume along those roadways indicate that the maximum ambient concentration contributions due to increased traffic volume would occur along SR 20 during the LWR construction period. The maximum estimated ambient contributions due to the increased traffic volume are all small fractions of applicable ambient standards and would not significantly contribute to exceedance of the ambient standards (Chun and Chang 1991a).

Electric power requirements for each of the three NPR technologies and their support facilities at INEL are estimated to be about 10 MW during construction and about 100 MW during operation. An NPR facility based on the LWR or MHTGR technology would be a net producer of electricity during operation, that is, more electricity would be produced than consumed. Electric power needed (1) during NPR construction, (2) during operation of the HWR, and (3) during operation of the LWR or MHTGR while the reactor's electric generating system is out of service would be purchased from the Utah Power and Light Company, Idaho Power Company, or the BPA system. These utilities obtain their power from a mix of facilities that includes hydroelectric, nuclear, and fossil-fueled power plants situated over a large area. The emissions associated with generation of electricity for an NPR would be released over a large area and would have little effect on ambient air quality at any location.

A detailed analysis was performed for a possible visibility impairment at the Craters of the Moon National Monument and Wilderness Area (Class I area), about 43 km west-southwest of the Idaho Chemical Processing Plant (ICPP), due to NO_x emissions from the ICPP main stack (Engineering-Sciences 1988). The results of level III visibility impairment analysis showed that only very slight visibility impairments would occur (0.7% reduction in visual range, barely perceptible plume with slightly yellow coloration, and a plume slightly darker than background) at the point of the monument closest to ICPP with the current NO_x emission level of 1,542 t/yr. No visibility impairment was predicted at locations inside the monument farther away from the ICPP and at higher elevation than the boundary point. The ICPP emissions of NO_x are to be reduced by about 90% in 1992 to 150.1 t/yr (WINCO 1990). Even after the addition of the new NO_x emissions from the NPR support facilities (93.8 t/yr for the MHTGR, the highest among the three NPR technologies), the site's NO_x emissions would be about 16% of the current total. Therefore, no visibility impairment is anticipated at the Craters of the Moon National Monument and Wilderness Area as a result of the additional NO_x emissions from the NPR support facility process operations.

Combustion of fossil fuels during NPR construction and operation would result in emissions of CO_2 , a greenhouse gas contributing to global warming. The highest estimated CO_2 emission rates during the NPR construction and operation periods are on the order of 10^4 t/yr for an LWR during construction and 10^6 t/yr for an HWR during operation at INEL (Chun and Chang 1991b). These emission rates correspond to about approximately one-millionth and

one-twenty-thousandth, respectively, of the estimated annual global CO₂ emission rate attributable to fossil fuel combustion (Rotty, Marland, and Treat 1984).

The estimated increases in ambient concentrations of toxic air pollutants (maximum 8-hour concentration at the INEL boundary) resulting from emissions from proposed new NPR sources are given in Table E.26 by reactor technology. These maximum 8-hour concentrations (baseline and baseline plus new NPR emissions) are all substantially less than the corresponding ambient limits (to be used by the state of Idaho as one of the criteria in evaluating construction permit application for a new source) as proposed by the state of Idaho (IDHW 1990).

Table E.26 shows that small amounts of O₃-depleting gases (Freon-113 and TCA) would be emitted from the NPR support facility process operations. However, production of these O₃-depleting gases would be phased out before or shortly after the start of NPR operation, in the year 2000. Industry is developing substitute materials for these gases, and when suitable substitutes become available, the O₃-depleting gases would be replaced.

E.2.1.3 Savannah River Site

Major sources of air pollutant emissions during NPR construction and operation are described in Section 5.1.2. Potential ambient air quality impacts of the emissions from NPR construction and operation activities for the three NPR technologies at SRS were analyzed with the computer codes described in Section E.2.1 (NUS 1991). The model input data included the emission inventories for the baseline, construction, and operational sources summarized in Table E.27 (see Tables E.28 and E.29 for the emission rates during construction and operation by source category) and one full year (1985) of on-site, hourly meteorological data. Among 5 years (1982-1986) of data, the 1985 data set exhibited the worst diffusion characteristics (NUS 1991). Model options selected are described in NUS (1991). The short-term emission rates for criteria pollutants during the peak construction periods are estimated to be similar for the three NPR technologies. The estimated emission rates for criteria pollutants during the NPR construction and operation periods are quite low compared with baseline emissions at SRS, except for CO during the construction period (Table E.27). The annual emission rates for the construction period are overestimates because peak construction activities were assumed to continue throughout the year. The NO_x emission rates for the new stationary sources for the MHTGR (97.4 t/yr) and the HWR (117 t/yr) would be greater than the PSD significant emission rate for NO_x (40 tons/yr, or 36.3 t/yr) (Table E.29). This means that the construction of the HWR or MHTGR would be considered a major modification subject to PSD regulations and best available control technology requirements. New emissions of toxic air pollutants from the NPR support facilities would be similar in order of magnitude to the baseline emissions for HNO₃, but those for TCA would be a small fraction of the baseline emissions. The C₃H₆O and NH₃ would be new emissions at SRS (Table E.31).

Maximum increases in ambient concentrations of criteria pollutants estimated for the SRS boundary are given in Table E.30 for baseline, baseline plus construction, operation, and baseline plus operation emission sources. Most of the highest increases in pollutant concentrations at the SRS boundary due to NPR construction or operation sources would occur at the southwestern boundary. High concentrations occur in this area because more than 50% of the baseline and new point source emissions of SO₂, NO_x, and PM₁₀ are released from boilers at D-Area powerhouses near the southwestern SRS boundary (Figure 3.2-10).

The estimated ambient concentration levels of NO_2 , CO , PM_{10} , and TSP resulting from the baseline, baseline plus construction, and baseline plus operation source emissions are all small fractions of applicable national and state ambient air quality standards for all averaging time periods. The corresponding concentration levels for SO_2 are significant fractions of applicable ambient standards. However, these levels would remain below the standards after the addition of background concentrations (Table E.30).

The PM_{10} and TSP emissions rates listed in Table E.28 for the NPR construction activities are based on the assumption that, in accord with normal construction industry practice, water would be sprayed to reduce by half the fugitive dust emissions from construction activities, such as site preparation and excavation. The frequency water spraying would be increased during extremely dry and windy conditions to minimize the fugitive dust emissions from construction activities.

Increased traffic along the major roadways leading to the proposed NPR site at SRS during the NPR construction and operation periods would contribute to the ambient air pollutant concentrations. The results of air quality modeling conducted with the CALINE3 model indicate that the maximum emissions contributions attributable to increased traffic would occur along SR 125N during the LWR construction period (NUS 1991). These maximum estimated ambient contributions along SR 125N at the SRS boundary are all small fractions of applicable ambient standards and would not significantly contribute to exceedance of the ambient standards (NUS 1991).

Electric power requirements at SRS for each of the three NPR technologies and support facilities are estimated to be about 10 MW during construction and 100 MW during operation. An NPR facility based on the LWR or MHTGR technology would be a net producer of electricity during operation (more electricity would be produced than consumed). Electric power needed (1) during NPR construction, (2) during operation of the HWR, and (3) during operation of the LWR or MHTGR while the reactor's electric generating system is out of service would be purchased from the South Carolina Gas Co. South Carolina Gas obtains its power from a mix of facilities that includes nuclear and fossil-fueled power plants located across a large area. The resulting emissions from this mix would not significantly affect ambient air quality at any location.

Combustion of fossil fuels for NPR construction and operation activities would result in emissions of CO_2 , a greenhouse gas contributing to global warming. The highest estimated CO_2 emission rates during the NPR construction and operation periods at SRS are on the order of 10^4 t/yr for an LWR during construction and 10^6 t/yr for an HWR during operation (NUS 1991). These emission rates correspond to about one-six-millionth and one-twenty-thousandth, respectively, of the estimated annual global CO_2 emission rate due to fossil fuel combustion (Rotty, Marland, and Treat 1984).

The estimated increases in maximum 24-hour concentrations of toxic air pollutants at the SRS boundary resulting from emissions from the proposed new NPR sources are given in Table E.31. These maximum 24-hour concentrations (baseline and baseline plus new NPR emissions) are all lower than the corresponding ambient air quality standards proposed by the South Carolina Department of Health and Environmental Control (SCDHEC 1989).

Table E.31 shows that small amounts of O_3 -depleting gases (Freon-113 and TCA) would be emitted from the NPR support facility process operations. However, production of these

O₃-depleting gases would be phased out before or shortly after the start of NPR operation, in the year 2000. Industry is developing substitute materials for these gases, and when suitable substitutes become available, the O₃-depleting gases would be replaced.

E.2.2 Impacts of Operating Cooling Towers

The potential environmental impacts of operating the cooling towers proposed for the three NPR technologies at the three NPR candidate sites are described in this section. A detailed analysis of potential environmental impacts of operating proposed cooling towers for the three NPR technologies was conducted for the three NPR candidate sites (PNL 1991; EG&G 1991; NUS 1991). The analysis was performed with the Seasonal/Annual Cooling Tower Impact (SACTI) model licensed by the Electric Power Research Institute (Policastro, Coke, and Wastag 1984; Dunn, Coke, and Policastro 1987). The potential environmental impacts analyzed include visible plume, plume-induced ground fogging, rime icing, and drift-related glaze-ice formation and salt deposition.

The SACTI model incorporates the latest theoretical developments in the field, as well as highly efficient computational schemes. The performance and accuracy of the SACTI model were determined by comparing the predictions of the submodels of the SACTI model dealing with specific aspects of cooling tower plume impacts with those of other available cooling tower impact models, and by verification with field and laboratory data (Policastro, Dunn, and Carhart 1981a-e; Policastro, Coke, and Wastag 1984). The comparison and verification included both mechanical- and natural-draft cooling towers in single- and multiple-tower configurations. The SACTI plume submodel predicted the visible plume length within a factor of 2.5 for 77% of the field data for the single natural-draft cooling tower, and for 80% of the field data for the multiple mechanical-draft cooling towers. In addition, the SACTI plume submodel demonstrated the best predictive capability among the 12 available cooling tower impact models.

The SACTI's drift submodel predicted the salt deposition rate within a factor of 3 in 92% of the field data collected within 1.0 km of a single mechanical-draft cooling tower and in 50% of the field data taken near a multiple mechanical-draft cooling tower system. The drift submodel's predictive capability equaled that of the best-performing model among the 12 available cooling tower impact models.

Quantitative data do not exist for verification of the mathematical plume model for ground fogging and deposition of ice. Long-term field data also do not exist for verifying the seasonal and annual predictions of the SACTI model.

The input data for the SACTI model included the cooling tower design and operating parameters and 1 to 5 years of hourly meteorological data. The meteorological data used were collected either on-site or at a nearby National Weather Service Station. The worst-case conditions for thermal discharge to the environment were assumed (i.e., full-power operation of HWR and full-power operation of the NPR with the electricity generating system not functioning for the LWR and MHTGR). It also was assumed that the drift droplet size distribution for a circular, mechanical-draft cooling tower at the Arizona Public Service's Palo Verde Nuclear Generating Station (Wilbur and Vercauteren 1986) approximates the drift droplet size distribution for the circular mechanical-draft cooling towers for the NPR.

E.2.2.1 Hanford Site

Circular, mechanical-draft, wet cooling towers are proposed for dissipation of the waste heat generated from operation of the NPR and its support facilities at Hanford. Three cooling towers are required for the HWR and LWR technologies and two cooling towers for the MHTGR technology (see Figures 3.3.2-4, 3.3.3-4 and 3.3.4-4 for the locations of these cooling towers with respect to other NPR facilities and Table E.32 for the dimensions of these cooling towers). Potential environmental impacts from operating the mechanical-draft cooling towers for the NPR at Hanford were analyzed using the SACTI model licensed by the Electric Power Research Institute (PNL 1991). The model input data included the cooling tower design and operating parameters (Table E.32) and one full year (1986) of hourly meteorological data collected on the site. The 1986 data set was selected because it exhibited the most restrictive dispersion characteristics of the 7 years (1983-1989) of data examined (PNL 1991).

Table E.33 summarizes (1) the estimated water and salt deposition rates as a function of distance from the cooling towers and (2) the frequency of occurrence of visible plume as a function of plume length. The maximum deposition of water associated with drift is estimated to occur within a few hundred meters of the cooling towers. At 0.5 km from the towers, the average water deposition rate is less than 10% of the annual average precipitation at Hanford (160 mm/yr). During an extended period when the ground temperature is below freezing, glaze ice could build up on surfaces as the deposited water freezes. However, the average thickness of glaze ice formed at 0.5 km from the towers would be about 1 mm or less, even if such buildup were continued for an entire month (see Table E.33 for the estimated rate of water deposition in millimeters per month). Therefore, no glaze ice impact is anticipated beyond a few hundred meters from the towers. The nearest public highway (SR 240) is about 6 km southwest of the NPR site for the HWR or MHTGR and about 12 km southwest of the LWR (WNP-1) site.

The maximum deposition of salt from drift emissions is also estimated to occur within a few hundred meters of the cooling tower (Table E.33). The rate of salt deposition would decrease rapidly from averages of 1,900, 1,600, and 2,000 kg/km²·mo at a distance of 0.3 km to averages of 540, 400, and 1,200 kg/km²·mo at a distance of 0.5 km from the cooling towers for the HWR, LWR, and MHTGR technologies, respectively. Potential impacts of drift salt deposition at Hanford on terrestrial vegetation are discussed in Section 5.2.4.

Table E.33 shows that visible plumes would dissipate between 3 and 5 km from the cooling towers, indicating that visible plumes would seldom extend to the nearby public highways or beyond the Hanford Site boundary. The nearest public highway (SR 240) is about 6 km southwest of the proposed HWR/MHTGR site, and the nearest site boundary is about 4 km east of the LWR (WNP-1) site. Potential visual impacts of the cooling tower plumes are discussed in Section 5.2.3.

E.2.2.2 Idaho National Engineering Laboratory

Circular, mechanical-draft, wet cooling towers are proposed for dissipation of the waste heat generated from the operation of the NPR and its support facilities at INEL. Three cooling towers would be required for the HWR and LWR technologies and two for the MHTGR technology (see Table E.34 for the dimensions of these cooling towers). Potential environmental impacts of operating the mechanical-draft cooling towers for the NPR at INEL were analyzed using the SACTI model licensed by the Electric Power Research Institute (EG&G 1991). The

model input data included the cooling tower design parameters (Table E.34) and one full year (1989) of on-site, hourly meteorological data.

Table E.35 summarizes (1) the estimated water and salt deposition rates as a function of distance from the cooling towers and (2) the frequency of occurrence of visible plume as a function of plume length. The maximum deposition of water associated with drift is estimated to occur within a few hundred meters of the cooling towers. At 0.5 km from the towers, the average water deposition rate is less than 12% of the annual average precipitation at INEL (221 mm/yr). During an extended period when the ground temperature is below freezing, glaze ice could build up on surfaces as the deposited water freezes. However, at 0.5 km from the towers, the glaze ice so formed would average about 2 mm or less thick, even if such buildup were continued for an entire month. (Table E.35 gives the estimated rate of water deposition in millimeters per month.) Therefore, no glaze ice impact is anticipated beyond several hundred meters from the towers. The nearest public highways (US 20, 26, and 26/20) are about 8 km south to southwest of the NPR site.

The maximum deposition of salt due to drift emission is also estimated to occur within several hundred meters of the cooling towers (see Table E.35). The rate of salt deposition would decrease rapidly from averages of 2,900, 5,200, and 2,400 kg/km²·mo at a distance of 0.5 km to averages of 430, 550, and 590 kg/km²·mo at a distance of 1.0 km from the cooling towers for the HWR, LWR and MHTGR technologies, respectively. Potential impacts of drift salt deposition at INEL on terrestrial vegetation are discussed in Section 5.3.4.

Results of the cooling tower plume modeling analysis indicate that visible plumes about 15 km long would occur occasionally during operation of the mechanical-draft cooling towers for the NPR at INEL and that the visible plumes would dissipate at a distance of 21 km (see Table E.35). A visible plume with a length of about 10 km would occur with a total frequency for all directions 9%-25% of the time, depending on the NPR technology, and a maximum frequency to the northeast of the towers of 4%-6% of the time. These long visible plumes would occur during cool periods of the year -- mostly at night during spring and fall, and at night and early morning during winter. The analysis shows that (1) visible plumes would eventually dissipate within a distance of 21 km; (2) visible plumes from the LWR cooling towers would not extend beyond the INEL boundary, but those from the HWR and MHTGR cooling towers would occasionally extend beyond the boundary in a south-southeast to southwest direction; and (3) visible plumes would extend above the public highways traversing INEL south of the NPR site (i.e., U.S. 20, 26, and 26/20) with a frequency of about 1%-4% of the time, depending on specific direction.

However, these predictions of visible plumes are probably overestimates because of the unrealistically high relative humidity values used in modeling (Start 1990). The relative humidity data at INEL are derived from dry bulb and dew point temperatures measured at the Central Facility Area location during 1989. Elevated deviations of dew point temperature are suspected during the wintertime, when the air is near saturation. Calculated relative humidity could be 2%-8% higher than actual humidities if the dew point were measured at 0.1-0.5°C higher than actual values. Taking into consideration the upward bias in the 1989 relative humidity values at INEL, it is estimated that the visible plume is not likely to extend beyond the INEL boundary and might only occasionally extend to the public highways south of the NPR site. Potential visual impacts of the cooling tower plumes at INEL are discussed in Section 5.3.3.

E.2.2.3 Savannah River Site

Circular, mechanical-draft, wet cooling towers are proposed for dissipation of the waste heat generated from the operation of an NPR and its support facilities at SRS. Eight mechanical-draft cooling towers would be required for an HWR, 12 for an LWR, and 9 for an MHTGR. Natural-draft, wet cooling towers are also proposed at SRS as an alternative to the mechanical-draft towers. Two natural-draft cooling towers would be required for each of the three NPR technologies (see Table E.36 for the dimensions of these cooling towers).

Potential environmental impacts of operating the mechanical- and natural-draft cooling towers for the NPR at SRS were analyzed using the SACTI model licensed by the Electric Power Research Institute (NUS 1991). The model input data included the cooling tower design parameters (Table E.36) and 5 years (1982-1986) of hourly meteorological data from the National Weather Service (NWS) station at the Bush Field Airport in Augusta, Georgia, about 33 km west-northwest of the NPR site. (The NWS station data were used because they were more complete than on-site data for the parameters needed in the modeling.) For the natural-draft cooling tower alternative, it was assumed that the drift droplet size distribution data for a natural-draft cooling tower at Chalk Point Power Plant in Maryland (Meyer and Stanbro 1977) approximates the size distribution for the natural-draft cooling towers for the NPR at SRS.

Tables E.37 and E.38 summarize the calculated (1) water and salt deposition rates as a function of distance from the cooling towers and (2) the frequency of occurrence of visible plume as a function of plume length for the mechanical- and natural-draft cooling towers for the three NPR technologies. The maximum deposition of water associated with drift is estimated to occur within a few hundred meters of the cooling towers. At 0.5 km from the towers, the average water deposition rate would be less than 1% of the annual average precipitation at SRS (1,220 mm/yr). During an extended period when the ground temperature is below freezing, glaze ice could build up on surfaces as the deposited water freezes. However, even if such buildup were continued for an entire month, glaze ice formed at 0.5 km from the mechanical-draft cooling towers would average less than 1 mm thick for the mechanical-draft cooling towers and less than 0.1 mm for the natural-draft cooling towers (Tables E.37 and E.38). Therefore, no glaze-ice impact from drift emissions is anticipated from the natural-draft towers, and no such impact is anticipated beyond several hundred meters from the mechanical-draft cooling towers. The highway closest to the NPR site where public access is not limited (SR 125) is about 8 km to the southwest.

The maximum deposition of salt due to drift emission is also estimated to occur within several hundred meters of the cooling towers (Tables E.37 and E.38). The rate of salt deposition would decrease rapidly from averages of 65, 680, and 160 kg/km²·mo at a distance of 0.3 km to averages of 14, 26, and 160 kg/km²·mo at a distance of 1.0 km from the mechanical-draft cooling towers for the HWR, LWR, and MHTGR technologies, respectively. The rates of salt deposition due to drift emissions from the natural-draft cooling towers are substantially smaller than those for mechanical-draft towers within 1 km from the towers. Potential impacts of drift salt deposition on terrestrial vegetation are discussed in Section 5.4.4.

Visible plumes would dissipate between 5 and 10 km from the mechanical-draft cooling towers (Table E.37), indicating that visible plumes would seldom extend to the nearby public highways or beyond the SRS boundary. The nearest public highway (SR 125) is about 8 km southwest of the proposed NPR site. Visible plumes from the natural-draft towers are estimated to occasionally extend beyond 10 km from the towers (Table E.38). It is calculated that visible

plumes would occur above SR 125 (about 8 km southwest of the NPR site) only about 0.5% of the time. Potential visual impacts of the cooling tower plumes at SRS are discussed in Section 5.4.3.

E.2.3 Noise Impacts

This section summarizes the results of a detailed analysis of the potential noise impacts of activities associated with construction and operation of the NPR and its support facilities for the three NPR technologies at each of the three NPR candidate sites (PNL 1991; EG&G 1991; NUS 1991). The results of an analysis of the potential impacts of ground vibration and airblast overpressure due to blasting at INEL for the NPR construction (EG&G 1991) are also summarized. The analysis to estimate increases in noise levels at receptor locations was performed with (1) the Environmental Noise Model (ENM 1.2) (Tonin 1985; RTA 1989) for stationary sources and sources such as construction equipment and machines at a construction site and (2) the Federal Highway Administration's (FHWA's) Highway Noise Prediction Model (STAMINA) (FHWA 1978, 1979, 1982) for highway traffic. The potential impacts of noise level increases were assessed in terms of responses from communities and sensitive receptors with the Composite Noise Rating (CNR) model (Stevens, Rosenblith, and Bolt 1953; Parrack 1957; Bolt, Beranek, and Newman 1984) and the Individual Annoyance Prediction (IAP) model (Fidell et al. 1987, 1988), as further described in Section E.2.3.4. To analyze potential impacts of ground vibration and airblast overpressure due to blasting, the empirical relationships described in Rose (1981) and Construction Engineering Research Laboratory (Schomer 1973; Schomer 1981), respectively, were utilized (see Section E.2.3.5 for further details).

In addition to the assumptions of the worst-case noise emissions described in Section 5.1.2, the noise modeling for stationary sources was for very unfavorable meteorological conditions at each site (i.e., summer seasonal average ambient temperature; 80% relative humidity at Hanford and INEL and 90% at SRS; 2.5 m/s wind from source to receptor; and an inversion condition with a temperature gradient of 1°C/100 m). The noise level increases due to all stationary noise sources (intrusive noise level) were estimated at the site boundary or at the sensitive receptor location closest to the site boundary. Noise level increases from highway traffic were estimated at a distance of 15 m from the edge of the roadway. Ambient noise levels, combining both the background and intrusive noise levels, were also determined for stationary noise sources. The predicted ambient noise levels were compared with the EPA guideline for protection of the public from the effect of broadband environmental noise in typically quiet outdoor and residential areas (55 dBA in L_{dn}) and the EPA guideline for protection against hearing loss (70 dBA in L_{eq})*.

For a rural community, the modified CNR model was used to predict community responses to intrusive noise levels. The IAP model was used to estimate responses on the part of individuals engaged in ongoing activities indoors. The following sections describe the results of noise analysis conducted for the three NPR candidate sites.

* L_{dn} is the day-night weighted equivalent sound level. L_{eq} is the equivalent steady sound level that, if continuous during a specified time period, would contain the same total energy as the actual time-varying sound. For example, $L_{eq}(1h)$ is the 1-hour equivalent sound level.

E.2.3.1 Hanford Site

The major noise sources, along with their sound power levels and estimated noise impacts, are listed in Tables E.39 and E.40 for the period of peak NPR construction and for the NPR operation period at Hanford. Noise impacts for the LWR technology at Hanford in terms of individual responses were estimated at a farmhouse nearest to the WNP-1 site. The farmhouse is located about 6 km east of the WNP-1 site and about 2 km east of the Hanford boundary. Community responses were estimated for the town of Matthew's Corner, 9.6 km east of the WNP-1 site. For the HWR and MHTGR technologies, noise impacts were evaluated at the Hanford boundary nearest to the proposed NPR site. That boundary location is about 8.5 km southwest of the NPR site. Noise impacts are expressed in terms of noise level increases (in L_{eq}) and overall noise levels, including the background levels (in L_{eq} and L_{dn}). The most probable level of individual response estimated from the IAP model and the community response predicted using the modified CNR model are also provided.

The number of pieces of construction equipment used and the associated noise emissions would be greater at the NPR site than at the WNP-1 site. The duration of the peak noise emission period would be shorter at the WNP-1 site because the construction of the LWR at Hanford is already 63% complete (WHC 1990). However, Table E.39 shows that noise level increases due to construction equipment would be greater at the farmhouse east of the WNP-1 site for the LWR than at the southwestern site boundary for the HWR or MHTGR. This result can be attributed to the fact that the farmhouse is closer to the WNP-1 site than the southwestern boundary is to the NPR site. The estimated maximum noise levels at the town of Matthew's Corner and the farmhouse site for the peak construction period are 36.3 and 23.8 dBA in L_{eq} and 38.2 and 22.4 dBA in L_{dn} , respectively. The mechanical-draft cooling tower fans would be the source of the greatest noise during NPR operation. Table E.40 shows that the maximum noise levels estimated for the town of Matthew's Corner and the farmhouse east of the WNP-1 site during the NPR operation period are 32.4 and 18.0 dBA in L_{eq} and 38.8 and 24.4 dBA in L_{dn} , respectively.

The estimated noise levels listed in Tables E.39 and E.40 are all less than the EPA guideline for protection of the public from the effect of broadband environmental noise in typically quiet outdoor and residential areas (55 dBA in L_{dn}) and the EPA guideline for protection against hearing loss (70 dBA in L_{eq}). Because the estimated noise level increases at the receptor locations are small, no individual annoyance or community reaction is predicted during construction or operation of an NPR.

Estimated peak and off-peak hourly traffic volumes and associated noise levels along the major roadways leading to the NPR site at Hanford are listed in Table E.41 for the baseline, NPR construction, and NPR operation periods. Estimated noise levels due to traffic are presented for the three NPR technologies, along with the largest increases above the baseline noise levels among the three NPR technology cases. The maximum noise level increases, equal to or less than 3.3 dBA (L_{eq}), are less than the 5-dBA increase below which no significant change in community reaction is expected (see Section E.2.3.4).

E.2.3.2 Idaho National Engineering Laboratory

The major noise sources, along with their sound power levels and estimated noise impacts, are provided in Tables E.42 and E.43 for the peak NPR construction period and the NPR

operation period at INEL. Noise impacts for the three NPR technologies were estimated at the nearest INEL site boundary, which is located about 14.5 km south of the proposed NPR site. Noise impacts are expressed in terms of noise level increases (in L_{eq}) and overall noise levels, including background levels (in L_{eq} and L_{dn}). Also provided are the most probable level of individual response (estimated with the IAP model) and community response (predicted with the CNR model).

It was assumed that the same types and number of pieces of construction equipment would be used at INEL for each of the three NPR technologies (Table E.42). Peak construction activities involving operation of these pieces of equipment and machines and their associated noise emissions would last for about 18 months, a period that includes construction of a large evaporation pond. For the mitigative alternative of building mechanical evaporators instead of the evaporation pond for disposal of blowdown from the secondary cooling system, peak construction activities and associated noise emissions would last for about six months.

Table E.42 shows that the estimated maximum noise levels at the southern INEL boundary during the peak construction period are 30.6 and 12.2 dBA (L_{eq}) and 36.5 and 18.1 dBA (L_{dn}) for an outdoor and indoor receptor, respectively. During NPR operation, the dry cooling tower (air-blast heat exchanger) fans for an MHTGR would be the largest noise source at INEL. The air-blast heat exchangers are an alternative to the wet cooling towers for the MHTGR technology to mitigate other environmental impacts. Table E.43 shows that the estimated maximum noise levels at the southern INEL boundary during NPR operation are 30.4 and 12.0 dBA (L_{eq}) and 36.5 and 18.1 dBA (L_{dn}) for an outdoor and indoor receptor, respectively.

The estimated noise levels listed in Tables E.42 and E.43 are all less than the EPA guideline for protection of the public from the effect of broadband environmental noise in typically quiet outdoor and residential areas (55 dBA, L_{dn}) and the EPA guideline for protection against hearing loss (70 dBA, L_{eq}). Because the estimated increases in noise levels at the southern INEL boundary are small, no individual annoyance or community reaction is predicted during NPR construction and operation.

Estimated peak and off-peak hourly traffic volumes and associated noise levels along the major roadways leading to the NPR site at INEL are listed in Table E.44 for the baseline, NPR construction, and NPR operation periods. Traffic noise levels are presented for the three NPR technologies, along with the largest increases above the baseline noise levels among the three NPR technologies. The estimated maximum noise level increases, equal to or less than 5.3 dBA (L_{eq}), are within the range of the 5-dBA increase below which no significant change in community reaction is expected.

The potential impacts of ground vibration and airblast overpressure due to blasting were estimated at the ICPP, a major INEL facility located 5.3 km west of the NPR site. The following assumptions were made in the analysis: (1) a blasting event called a "shot" consists of a sequence of blasts set off at 25-millisecond intervals; (2) blasting occurs in well-covered holes at depths ranging from 3 to 9 m; and (3) maximum charge weight per delay is 114 kg.

The results of the analysis using the procedures described in Section E.2.3.5 indicate that the peak particle velocity (PPV) at receptor locations beyond about 100 m from the blast would not exceed 5.1 cm/s, a value considered safe for poor plaster. The PPV at the ICPP is estimated to be no greater than about 0.001 cm/s, and, therefore, no impacts due to ground vibration are anticipated.

The airblast overpressure is estimated to be equal to or less than the threshold value that may cause damage to farm or wild animals (7.0 kg/m^2) at receptors located about 600 and 1,100 m from the blast for the base and focus zone, respectively (see Section E.2.5.3). At ICPP, the airblast overpressure would be about 0.8 kg/m^2 for the base zone and 1.4 kg/m^2 for the focus zone and, therefore, no impacts due to air blast overpressure are expected.

E.2.3.3 Savannah River Site

The major noise sources, along with their sound power levels and estimated noise impacts, are provided in Tables E.45 and E.46 for the peak NPR construction period and the NPR operation period at SRS. Noise impacts for the three NPR technologies were estimated at the nearest SRS boundary, located about 12 km north of the proposed NPR site. Noise impacts are expressed in terms of noise level increases in (L_{eq}) and the overall noise levels, including the background levels in (L_{eq} and L_{dn}). Also provided are the most probable level of individual response (estimated with the IAP model) and community response (predicted with the CNR model).

The types and number of pieces of equipment employed at SRS for the NPR construction were assumed to be the same for the three NPR technologies. Table E.45 shows that the estimated maximum noise levels at the northern SRS boundary during the peak construction period are 50.0 and 38.8 dBA (L_{eq}) and 57.0 and 45.2 dBA (L_{dn}) for an outdoor and indoor receptor, respectively. During NPR operation, the mechanical-draft cooling tower fans would be a greater noise source than the alternative natural-draft cooling towers. Therefore, only the noise impacts from the mechanical-draft cooling towers were considered in the analysis. Table E.46 indicates that the estimated maximum noise levels at the northern SRS boundary location during NPR operation are 50.0 and 38.8 dBA (L_{eq}) and 57.0 and 45.2 dBA (L_{dn}) for an outdoor and indoor receptor, respectively.

Although the estimated L_{eq} noise levels listed in Tables E.45 and E.46 are all less than the EPA guideline for protection against hearing loss (70 dBA, L_{eq}), the estimated L_{dn} values (57 dBA) are higher than the EPA guideline for protection of the public from the effect of broadband environmental noise in typically quiet outdoor and residential areas (55 dBA, L_{dn}). However, these L_{dn} values are due principally to background noise (NUS 1991). Because the estimated increases in noise level at the northern SRS site boundary due to NPR activities are small, no individual annoyance or community reaction is predicted during NPR construction or operation (Tables E.45 and E.46).

Estimated peak and off-peak hourly traffic volumes and associated noise levels along the major roadways leading to the NPR site at SRS are listed in Table E.47 for the baseline, NPR construction, and NPR operation periods. Noise levels due to traffic are presented for the three NPR technologies, along with the largest increases above the baseline noise levels among the three NPR technology cases. The estimated maximum noise level increases during the construction period, which are equal to or less than 5.2 dBA (L_{eq}), are within the range of the 5-dBA increase below which no significant change in community reaction is expected.

E.2.3.4 Noise Level and Noise Impact Prediction Models

This section describes the noise-level prediction models and noise impact prediction models used in this analysis. The performance of these models is also discussed. The computer code used to estimate the noise levels due to NPR construction activities and normal operation was the Environmental Noise Model (ENM 1.2) (Tonin 1985; RTA 1989). The model is highly sophisticated, incorporating sound propagation models developed by the Edison Electric Institute (Bolt, Beranek, and Newman 1984) and the European Oil Companies Organization for Environmental and Health Protection (Manning 1981). The model is capable of accounting for the effects of meteorological conditions, ground cover, topography, and barriers.

To estimate the noise levels resulting from highway traffic during the periods of NPR construction and operation, a computerized version (STAMINA) of the FHA's Traffic Noise Prediction Model (FHWA 1978, 1979, 1982) was used. The model calculations take into account the traffic volume and mix (i.e., automobiles, medium trucks, and heavy trucks), as well as the speed and direction of traffic flow. Both L_{eq} and L_{dn} noise levels were estimated for the peak traffic hour and for an off-peak daytime hour.

Criteria for evaluating the annoyance level of potential noise impacts are discussed below. The criteria for more severe impacts (i.e., speech and sleep interference) are not discussed because the predicted noise levels, including noises from NPR construction and operation, would not result in any level of annoyance at sensitive receptor locations (Sections E.2.3.1, E.2.3.2, and E.2.3.3). Criteria for evaluating the degree of annoyance due to nonintermittent noise are based on the classification of annoyance provided in two models: (1) the Composite Noise Rating (CNR) model (Stevens, Rosenblith, and Bolt 1953; Parrack 1957; Bolt, Beranek, and Newman 1984) and the Individual Annoyance Prediction (IAP) model (Fidell et al. 1987).

The CNR is an empirical model that provides a multistep classification of annoyance, expressed in terms of probable community reaction in response to a newly introduced noise. The CNR model response categories are listed in Table E.48. The model evaluates relative noise levels in relation to community reaction levels and attitudes toward noise sources. The IAP model provides five levels of annoyance based on intrusion levels calculated using the Probabilistic Noise Audibility (PNA) model (Fidell et al. 1988) (see Table E.49 for the IAP model response categories). The intrusion levels are calculated by comparing the spectrum of newly introduced noise with an audibility spectrum that the PNA model generates as a composite of masking effect of the baseline ambient spectrum and the threshold of hearing. Like the CNR model, the IAP model includes an evaluation of the nonacoustic aspects of resident attitude toward noise.

The criterion of evaluating the degree of impacts due to mostly intermittent traffic noise is based on the fact that the fluctuations of noise levels, both in time and space, are often as high as 5 dB. Previous experience shows that a change in noise level of less than 5 dB would not produce a significant change in community reaction to the noise (Stevens, Rosenblith, and Bolt 1955).

Performance of the ENM model has been checked with noise measurements made near a steel plant in a suburban environment in northern Italy (Cerrato, Elia, and Renzullo 1990). The model performance was checked by comparing noise-level predictions made by the model against actual measurements made near the steel plant. Although the authors did not provide detailed information on the test conditions (e.g., sound power levels of sources and measurement

locations), their results showed very good agreement between the measured data and the values predicted by the ENM. The best agreement was in the 125-2000-Hz range, where the sound pressure level ranged from about 40 to 60 dB. In the low-frequency range (31-126 Hz), the model underpredicted the measured values by 1 to 3 dB. In the high-frequency range (2,000-8,000 Hz), the model underpredicted the actual noise level by as much as 15 dB. The authors attributed these results to the fact that the measured values include contributions from background sources near the measurement locations. The effects of background noise sources dominate in the high-frequency range because of greater atmospheric absorption of the high-frequency noise from the steel plant.

The performance of the STAMINA model has been checked with actual noise measurements from a large number of cases in Florida (FHWA 1978). The overall correlation coefficient for linear least-square regression of 66 predicted versus measured values at distances of 15, 30, and 60 m from the centerline of the roadway was greater than 0.9. The coefficient was higher (0.93) at 15 m, but lower (0.86) at 60 m. No measured value exceeded the predicted value by more than 3 dB, while the predicted value was sometimes as much as 5 dB greater than the measured value. The model tends to overpredict the higher end of the noise levels at a given distance from the roadway centerline.

Case histories of responses to noise from various source types as summarized by Rosenblith et al. (1953) and Hoover (1976) show that community response predictions made using the CNR model closely match the actual responses. Because the CNR model has been documented only for applications to populated communities, the IAP model also was used in this analysis because of the model's applicability to individual residents in rural and wilderness areas. Field verification of the IAP model has been documented only in limited cases dealing with electrical transformer noises (Vèr and Anderson 1977). Good agreement was found between the predicted and actual responses.

E.2.3.5 Ground Vibration and Airblast Overpressure Prediction Models

This section describes the empirical procedures used in this analysis to estimate potential impacts of ground vibration and airblast overpressure due to blasting. The performance of these procedures is also discussed. Potential impacts of ground vibration were estimated using a monograph available in Rose (1981) that relates PPV, charge weight per delay, and the distance between the blast and receptor (Figure E.4). The PPV or the velocity of ground movement is generally accepted as the best indicator for the potential for structural damage.

Figure E.4 is typically used to design a "first try" charge weight per delay for a given blasting work. Then, a distance is determined using that charge weight and a PPV corresponding to a level of no damage to structure (e.g., poor plaster). This distance is used as a radius for a sphere of impact around the source of vibration. If a human or structural receptor is located inside the sphere, the charge weight is reduced until all receptors are outside the sphere of impact. Figure E.4 can also be used to determine the value of PPV for a given distance between the blast and receptor and charge weight.

Potential impacts of airblast overpressure were estimated using empirical formulas described by the Construction Engineering Research Laboratory of the U.S. Army Corps of Engineers (Schomer 1973, 1981). The empirical formulas allow computation of peak overpressure

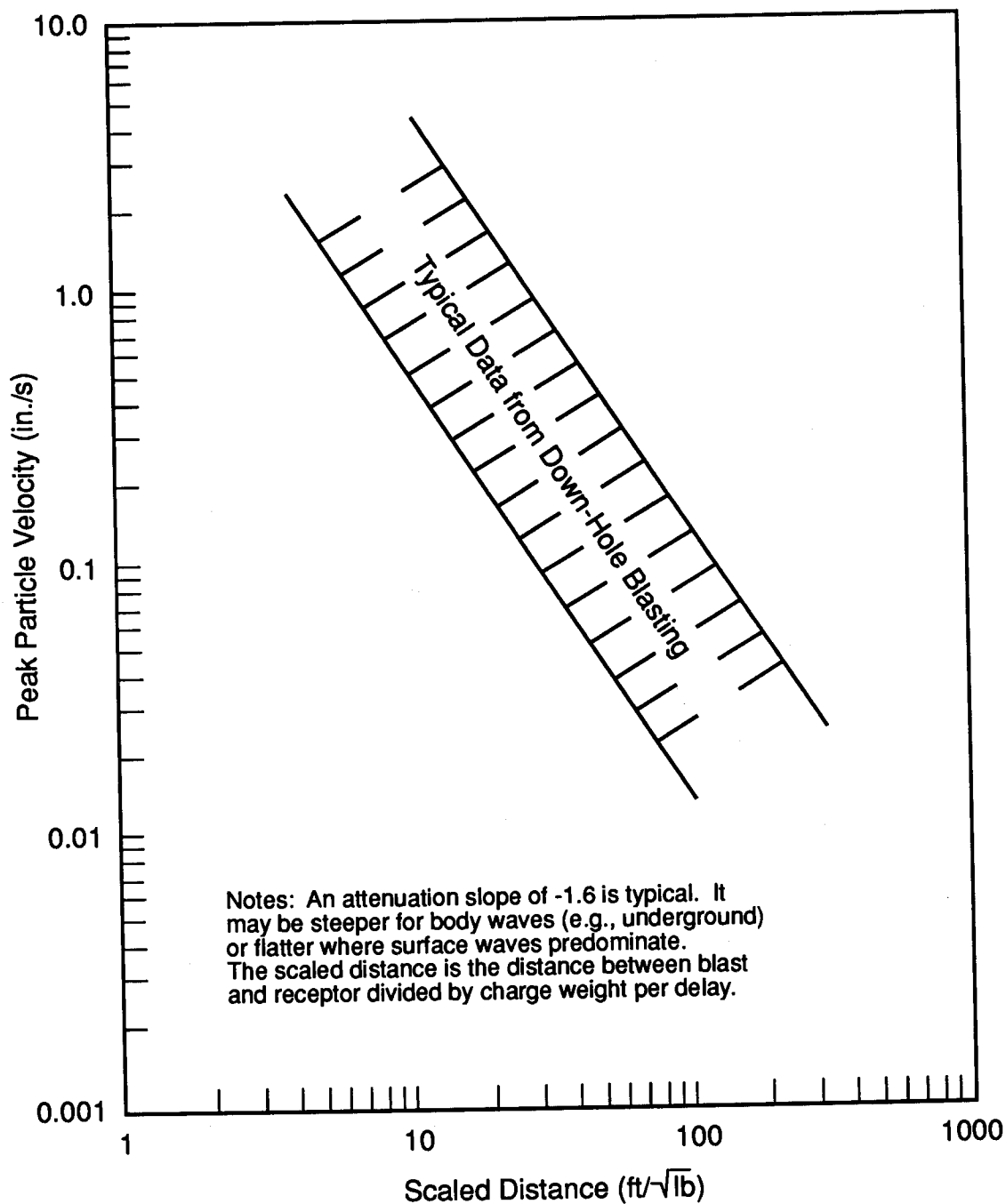


FIGURE E.4 Peak Particle Velocity as a Function of Scaled Distance (Source: Modified from Rose 1981)

at a given receptor location for a blast of a given charge weight per delay by correcting the peak overpressure at the reference distance of 61 m from 0.454 kg of charge exploded at ground level. The corrections are made for actual charge weight per delay, burial depth, sequential blasts, and distance between the blast and receptor. The distance correction is made for two typical cases, termed "zones," along the path of propagation where the meteorological conditions cause either an intensified or reduced peak overpressure level. The worst-case zone is called the "focus zone," in which focusing of the sound waves due to atmospheric refraction occurs. The zone most likely to occur along the propagation path is called the "base zone."

The criterion of evaluating the degree of impacts from ground vibration is based on PPV. A PPV of 5.1 cm/s is generally accepted as safe for poor plaster (DOE 1990). Window pane damage would not be caused at an overpressure level of 70-350 kg/m², although such overpressures may cause public annoyance (DOE 1990). Overpressures somewhat above 7.0 kg/m² were found to cause no damage to farm or wild animals (Fletcher and Busnel 1978). Therefore, an upper limit of 7.0 kg/m² was chosen as the criterion for evaluating potential impacts of overpressure.

Since the procedures used in estimating potential impacts of ground vibration and airblast overpressure are based on empirical data, they would be expected to provide fairly accurate estimates. The procedure for estimating the airblast overpressure used in this analysis tends to overpredict. The overpressure levels predicted for the base zone at two different distances for two field test cases using two different levels of charge weight were higher than the measured values up to 6 db (Raspet and Bobak 1988).

E.2.3.6 Threshold Limit Values for Noise

Threshold limit values for noise are established for the workplace to protect workers from hearing loss. The values are set by the American Conference of Governmental Industrial Hygienists (ACGIH 1990). The threshold values for continuous or intermittent noise are listed in Table E.50, and those for impulsive or impact noise are listed in Table E.51.

TABLE E.1 Estimated Seasonal and Annual Morning and Afternoon Mixing-Layer Depths for the Hanford Region

Period	Morning Mixing-Layer Depth (m)	Afternoon Mixing-Layer Depth (m)
Spring	400	1,800
Summer	200	2,200
Fall	300	1,200
Winter	400	600
Annual	350	1,400

Source: Holzworth 1972.

TABLE E.2 Ambient Air Quality Standards Applicable to Hanford and Maximum Estimated Background Concentrations (standards and concentrations are in $\mu\text{g}/\text{m}^3$)

Pollutant ^a	Averaging Time	Washington State Standards ^b	Primary NAAQS ^c	Secondary NAAQS ^c	Maximum Background Concentration
SO ₂	Annual	52	80	d	0.5
	24 h	260	365	d	6
	3 h	d	d	1,300	20
	1 h	1,018	d	d	49
	1 h	655 ^e	d	d	49
PM					
TSP ^f	Annual ^g	60	d	d	56
	24 h	150	d	d	356
PM ₁₀	Annual ^h	50 ⁱ	50	j	27
	24 h	150 ^k	150	j	81
CO	8 h	10,000	10,000	j	6,500
	1 h	40,000	40,000	j	11,800
O ₃	1 h	235 ^l	235	j	m
NO ₂	Annual	100	100	j	36
Pb PM	Calendar quarter	d	1.5	j	m

^aSO₂ is sulfur dioxide, PM is particulate matter, TSP is total suspended particulate, PM₁₀ is particulate matter $\leq 10 \mu\text{m}$, CO is carbon monoxide, O₃ is ozone, NO₂ is nitrogen dioxide, and Pb PM is lead particulate matter.

^bThe Washington State annual standards are never to be exceeded; short-term standards are not to be exceeded more than once per year, unless otherwise noted.

^cThe National Ambient Air Quality Standards (NAAQS), other than those for O₃ and PM₁₀ and those based on annual averages, are not to be exceeded more than once per year. The O₃ standard is attained when the expected number of days per calendar year with maximum hourly average concentrations above the standard is less than or equal to one. The 24-hour PM₁₀ standard is attained when the expected number of days with a 24-hour average concentration above the standard is less than or equal to one. The annual arithmetic mean PM₁₀ standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.

^dThere is no standard.

TABLE E-2 (Cont'd)

^eThe standard is not to be exceeded more than twice in any seven consecutive days.

^fThe TSP standards have been replaced by the PM₁₀ standards, but the former are serving as interim standards.

^gGeometric mean of all values reported during the year.

^hArithmetic mean of the quarterly arithmetic means for the four calendar quarters of the year.

ⁱThe attainment condition is the same as that for the annual NAAQS for PM₁₀.

^jSame as for the primary standard.

^kThe attainment condition also includes that for the 24-hour NAAQS for PM₁₀.

^lThe attainment condition is the same as that for the NAAQS for O₃.

^mNot estimated.

Sources: Washington Administrative Code, Chapters 173-470, -474, and -480; Code of Federal Regulations, Title 40, Part 50 (40 CFR 50); PNL 1990.

TABLE E.3 Baseline Source Emission Inventory at Hanford, by Source Category^a

Source Category	Emission Rate (t/yr) ^b					
	SO ₂	NO _x	VOC ^c	CO	PM ₁₀	TSP
Point						
Combustion	1,061	594	239	79	36	239
Process	0	759	0	0	0	0
Subtotal	1,061	1,353	239	79	36	239
Area						
Fugitive	0	0	0	0	34	68
Vehicular (on-site)						
Exhaust	7	97	84	482	5	5
Road Dust	d	d	d	d	395	1,056
Subtotal	7	97	84	482	435	1,130
Total	1,070	1,450	323	562	471	1,370

^aFor the baseline year 1997 when NPR construction activity is at its peak.^bThe individual values may not add up to the total because of rounding.^cVOC is volatile organic compounds.^dNot applicable.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: PNL 1991.

TABLE E.4 Maximum Allowable Prevention of Significant Deterioration (PSD) Concentration Increments

Pollutant	Averaging Time	PSD Increment ($\mu\text{g}/\text{m}^3$) ^a		
		Class I	Class II	Class III
SO ₂	Annual	2	20	40
	24 h	5	91	182
	3 h	25	512	700
TSP	Annual	5	19	37
	24 h	10	37	75
NO ₂	Annual	2.5	25	50

^aShort-term increments are not to be exceeded more than once per year.

Source: 40 CFR 52.21 (c).

TABLE E.5 PSD Sources Permitted in the Hanford Site Region and PSD Concentration Increments Consumed at the Hanford Boundary

PSD Source ^a		PSD Concentration Increment Consumed at the Hanford Boundary
Description	Location	
NO _x emissions from plutonium-uranium extraction (PUREX) and uranium oxide plants (DOE)	200 Area inside the Hanford boundary	Annual average NO ₂ concentration increment = 0 $\mu\text{g}/\text{m}^3$ ^b
SO ₂ emissions from wood-fired boilers (Boise Cascade Corporation)	Walla Walla County, Wash., about 40 km southeast of the Hanford boundary	Annual average SO ₂ concentration increment, 1 $\mu\text{g}/\text{m}^3$ at about 5 km northwest of the Boise Cascade facility; estimated to be negligible at the Hanford boundary

^aTSP not included because it is below threshold value.

^bPSD permit issued before the PSD increment for NO₂ was established.

Sources: Kinzer 1979; Sherman 1977.

TABLE E.6 Ambient Air Quality Data in and near the Hanford Boundary, 1988
($\mu\text{g}/\text{m}^3$)

Monitoring Station No. ^b	Ambient Concentration ^a				
	NO ₂	PM ₁₀		TSP	
	Annual	Max. 24-h	Annual ^c	Max. 24-h	Annual ^c
1	<10	d	d	d	d
2	< 8	d	d	d	d
3	<10	d	d	d	d
4	<10	d	d	d	d
5	< 6	d	d	d	d
6	<10	d	d	d	d
7	<10	d	d	d	d
8	<10 ^e	d	d	d	d
9	<14 ^e	d	d	d	d
10	d	d	d	91 ^e	33 ^e
11	d	d	d	120	38
12	d	211	35	317	66
13	d	d	d	256	53
14	d	d	d	140	60

^a1988 data, unless otherwise noted.^bSee Figure E.1 for the locations. Stations 1-10 are operated by Pacific Northwest Laboratory and stations 11-14 by the Washington Department of Ecology.^cGeometric mean.^dNot monitored.^e1987 data (station closed in 1988).

Sources: PNL 1988, 1989; WDOE 1989.

TABLE E.7 State of Washington Regulations on Maximum Allowable Noise Limits Applicable to Hanford

Environmental Designation for Noise Abatement		Maximum Allowable Noise Level in L_{eq} (dBA)
Source Area	Receptor Area	
Class C ^a (industrial)	Class A (residential)	60 (day) 50 (night)
	Class B (commercial)	65 (day)
	Class C (industrial)	70 (day)

^aHanford Site belongs to Class C.

Source: Washington State Administration Code,
Chapter 173-60, 1982.

TABLE E.8 Estimated Seasonal and Annual Morning and Afternoon Mixing-Layer Depths for the INEL Region

Period	Morning Mixing-Layer Depth (m)	Afternoon Mixing-Layer Depth (m)
Spring	480	2,330
Summer	260	2,900
Fall	330	1,550
Winter	400	730
Annual	370	2,090

Source: Holzworth 1972.

TABLE E.9 Ambient Air Quality Standards Applicable to INEL and Maximum Estimated Background Concentrations (standards and concentrations are in $\mu\text{g}/\text{m}^3$)

Pollutant ^a	Averaging Time	Idaho State Standards ^b	Primary NAAQS ^c	Secondary NAAQS ^c	Maximum Background Concentration
SO ₂	Annual	80	80	d	0.2
	24 h	365	365	d	8
	3 h	1,300	d	1,300	14
PM					
TSP ^e	Annual ^f	60	d	d	40
	24 h	150	d	d	88
PM ₁₀	Annual ^g	50 ^h	50	i	14
	24 h	150 ^j	150	i	32
CO	8 h	10,000	10,000	i	k
	1 h	40,000	40,000	i	k
O ₃	1 h	235 ^l	235	i	k
NO ₂	Annual	100	100	i	0.6
Pb PM	Calendar quarter	1.5	1.5	i	k

^aSee Table E.2 for pollutant abbreviations.^bThe Idaho State annual standards are never to be exceeded; short-term standards are not to be exceeded more than once per year, unless otherwise noted.^cThe NAAQS, other than those for O₃ and PM₁₀ and those based on annual averages, are not to be exceeded more than once per year. The O₃ standard is attained when the expected number of days per calendar year with maximum hourly average concentrations above the standard is less than or equal to one. The 24-hour PM₁₀ standard is attained when the expected number of days with a 24-hour average concentration above the standard is less than or equal to one. The annual arithmetic mean PM₁₀ standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.^dThere is no standard.^eThe TSP standards have been replaced by the PM₁₀ standards, but the former are serving as interim standards.^fGeometric mean of all values reported during the year.

TABLE E.9 (Cont'd)

^gArithmetic mean of the quarterly arithmetic means for the four calendar quarters of the year.

^hThe attainment condition is the same as that for the annual NAAQS for PM₁₀.

ⁱSame as for the primary standard.

^jThe attainment condition also includes that for the 24-hour NAAQS for PM₁₀.

^kData are unavailable.

^lThe attainment condition is the same as that for the NAAQS for O₃.

Sources: IDHW 1987; Code of Federal Regulations, Title 40, Part 50; EG&G 1991.

TABLE E.10 Baseline Source Emission Inventory at INEL, by Source Category^a

Source Category	Emission Rate (t/yr) ^b					
	SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
Point						
Combustion	1,889	1,054	35	142	132	132
Process	0	153	0	0	0	0
Subtotal	1,889	1,207	35	142	132	132
Area						
Fugitive	0	0	0	0	2	6
Vehicular						
Exhaust	4	57	47	280	3	3
Road Dust	c	c	c	c	231	625
Subtotal	4	57	47	280	236	634
Total	1,893	1,264	82	422	368	766

^aFor the baseline year 1997 when NPR construction activity is at its peak.

^bThe individual values may not add up to the total because of rounding.

^cNot applicable.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: EG&G 1991.

TABLE E.11 PSD Sources at INEL and PSD Concentration Increments Consumed at the INEL Boundary and Nearby Class I Area

PSD Source		PSD Concentration Increment Consumed ^a	
		INEL Boundary	Craters of the Moon
Coal-fired steam generating plant	Next to the Idaho Chemical Processing Plant ^b	<10% of Class II allowable SO ₂ and TSP PSD increments	<20% of Class I allowable SO ₂ and TSP PSD increments
Fuel processing restoration facility	Idaho Chemical Processing Plant	<1% of Class II allowable NO ₂ PSD increment	<1% of Class I allowable NO ₂ PSD increment

^aSee Table E.4 for the maximum allowable PSD increments.

^bSee Figure 3.2-8 for the location of the Idaho Chemical Processing Plant.

Sources: DOE 1980; Engineering-Sciences 1988.

TABLE E.12 Ambient Air Quality Data in and near the INEL Boundary, 1987-1989
($\mu\text{g}/\text{m}^3$)

Monitoring Station ^b	Ambient Concentration ^a					
	SO ₂			NO ₂	TSP	
	Max. 3-h	Max. 24-h	Annual	Annual	Max. 24-h	Annual
Van Buren Experimental Field Station (EFS)	14 ^c	8 ^c	0.2 ^c	4.2 ^d	e	e
Central Facility Area (CFA)	e	e	e	4.7 ^d	e	e
Craters of the Moon ^f	e	e	e	e	88	40
	e	e	e	e	35 ^g	15 ^g

^a1989 data, unless otherwise noted.

^bSee Figure E.2 for the locations. The on-site stations are operated by INEL and the off-site station (Craters of the Moon) by the Idaho Department of Health and Welfare.

^cData are for the fourth quarter of 1989.

^dHighest quarterly data from the third and fourth quarters of 1989.

^eNot monitored.

^fLocated about 40 km west of the southwestern corner of INEL.

^g1987 data.

Sources: INEL 1990; IDHW 1988.

TABLE E.13 Estimated Seasonal and Annual Morning and Afternoon Mixing-Layer Depths for the SRS Region

Period	Morning Mixing-Layer Depth (m)	Afternoon Mixing-Layer Depth (m)
Spring	375	1,700
Summer	400	1,800
Fall	300	1,400
Winter	350	1,000
Annual	375	1,450

Source: Holzworth 1972.

TABLE E.14 Ambient Air Quality Standards Applicable to SRS, Maximum Ambient Concentrations in and near SRS, 1988, and Maximum Estimated Background Concentrations for SRS (standards and concentrations are in $\mu\text{g}/\text{m}^3$)

Pollutant ^a	Averaging Time	South Carolina and Georgia State Standards	Primary NAAQS ^b	Secondary NAAQS ^b	Maximum Ambient Concentrations for 1986-1988		
					Within SRS	Outside SRS ^c	Maximum Background Concentration
SO ₂	Annual 24 h 3 h	80 365 1,300	80 365 d	d d 1,300	12 139 383	9 53 103	5 34 48
PM							
TSP ^e	Annual ^f	75	d	d	38	44	27
PM ₁₀ ^g	Annual ^h 24 h	50 150	50 150	i i	23 135	27 86	27 47
CO	8 h 1 h	10,000 40,000	10,000 40,000	i i	j j	j j	k k
O ₃ ⁱ	1 h	235	235	i	196	243 ^m	k
NO ₂	Annual	100	100	i	8	11	6
Pb PM	Calendar quarter	1.5	1.5	i	j	0.08	k

TABLE E.14 (Cont'd)

Pollutant ^a	Averaging Time	South Carolina and Georgia State Standards	Primary NAAQS ^b	Secondary NAAQS ^b	Maximum Ambient Concentrations for 1986-1988		
					Within SRS	Outside SRS ^c	Maximum Background Concentration
Gaseous	1 mo	0.8	d	d	j	j	k
Fluoride	1 wk	1.6	d	d	j	j	k
(as HF) ^a	24 h	2.9	d	d	j	j	k
	12 h	3.7	d	d	j	j	k

^aSee Table E.2 for pollutant abbreviations. Annual standards are not to be exceeded; short-term standards are not to be exceeded more than once per year, unless otherwise noted.

^bThe NAAQS, other than those for O₃ and PM₁₀ and those based on annual averages, are not to be exceeded more than once per year. The O₃ standard is attained when the expected number of days per calendar year with maximum hourly average concentrations above the standard is less than or equal to one. The 24-hour PM₁₀ standard is attained when the expected number of days with a 24-hour average concentration above the standard is less than or equal to one. The annual arithmetic mean PM₁₀ standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.

^cAt Barnwell, Beech Island, and Jackson monitoring stations.

^dThere is no standard.

^eThe TSP standards have been replaced by the PM₁₀ standards, but the annual TSP standard is retained in South Carolina.

^fGeometric mean; all other means are arithmetic means.

TABLE E.14 (Cont'd)

^sThe 24-hour PM_{10} standard is attained when the expected number of days with 24-hour average concentrations above the standard is less than or equal to one. The annual arithmetic mean PM_{10} standard is attained when the expected annual arithmetic mean concentration is less than or equal to the standard.

^bArithmetic mean of the quarterly arithmetic means for the four calendar quarters of the year.

^cSame as for the primary standard.

^dNot monitored because the potential release is insignificant compared with the standard.

^kNot estimated.

^lThe O_3 standard is met when the expected number of days per calendar year with the maximum hourly average concentration above the standard is less than or equal to one.

^mThis was the only measurement above $235 \mu g/m^3$ during the period 1986-1988.

ⁿApplicable to South Carolina only.

Sources: Code of Federal Regulations, Title 40, Part 50; duPont 1987, 1988; Davis, Martin, and Todd 1989; SCDHEC 1987a, 1988, 1989; NUS 1991.

TABLE E.15 Baseline Source Emission Inventory at SRS, by Source Category^a

Source Category	Emission Rate (t/yr) ^b					
	SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
Point						
Combustion	7,920	5,158	14	460	593	857
Process	0	614	66	0	0	0
Subtotal	7,920	5,772	80	460	593	857
Area						
Fugitive	0	0	124	140	2,525	2,683
Vehicle						
Exhaust	9	105	194	723	8	8
Road Dust	c	c	c	c	482	1,237
Subtotal	9	105	318	863	3,015	3,928
Total	7,929	5,877	398	1,323	3,608	4,785

^aFor the baseline year 1997 when NPR construction activity is at its peak.

^bThe individual values may not add up to the total because of rounding.

^cNot applicable.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: NUS 1991.

TABLE E.16 Emission Inventories for Baseline Sources at Hanford, and Construction and Operation Sources for an NPR and Support Facilities at Hanford

Emission Source	NPR Technology	Emission Rate (t/yr)										
		Criteria Pollutant							Toxic Pollutant ^a			
		SO ₂	NO _x	VOC ^b	CO	PM ₁₀ ^b	TSP	C ₃ H ₆ O ^b	NH ₃	Freon-113	HNO ₃	TCA ^b TCE ^b
Hanford Site baseline ^c		1,070	1,450	323	562	471	1,370	1.7	8.8	^d	-	0.8 -
Construction	HWR	21	292	58	323	529	1,430	-	-	-	-	-
	LWR	21	289	54	303	185	477	-	-	-	-	-
	MHTGR	20	280	39	215	371	1,010	-	-	-	-	-
Operation	HWR	8	70	7	49	29	75	-	-	20.0	2.9	0.5 -
	LWR	5	27	16	63	43	109	1.8	0.9	-	12.9	2.5 -
	MHTGR	5	118	11	140	42	106	-	-	-	-	0.2 -

^aOnly those toxic pollutants projected to be released by NPR and its support facilities are listed.

^bVOC is volatile organic compounds, PM₁₀ is particulate matter of diameter ≤10 μm, C₃H₆O is acetone, TCA is 1, 1, 1-trichloroethane, and TCE is trichloroethylene.

^cBaseline emissions for the peak construction year (1997) and the first year of NPR operation (2000) are estimated to be essentially the same.

^d "-" denotes no emissions.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: PNL 1991.

TABLE E.17 NPR Construction Source Emission Inventory for Criteria Pollutants at Hanford,
by NPR Technology and Source Category

NPR Tech- nology	Source Category	Emission Rate (t/yr) ^a					
		SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
HWR	Vehicular and machine exhausts						
	Construction site ^b	18.5	264	22.6	118	19.9	55.2
	Other on-site ^c	2.4	27.9	35.3	206	2.2	2.2
	Fugitive and road dusts						
	Construction site ^b	d	d	d	d	390	1,080
	Other on-site ^c	d	d	d	d	117	292
	Total	20.9	292	57.9	323	529	1,430
	Vehicular and machine exhausts						
	Construction site ^b	18.5	264	22.6	118	19.9	55.2
LWR	Other on-site ^c	2.2	24.9	31.7	185	2.0	2.0
	Fugitive and road dusts						
	Construction site ^b	d	d	d	d	57.1	154
	Other on-site ^c	d	d	d	d	106	265
	Total	20.7	289	54.3	303	185	477
	Vehicular and machine exhausts						
	Construction site ^b	18.5	265	22.6	118	19.9	55.2
	Other on-site ^c	1.0	15.3	16.6	96.9	1.0	1.0
	Fugitive and road dusts						
MHTGR	Construction site ^b	d	d	d	d	292	811
	Other on-site ^c	d	d	d	d	57.7	146
	Total	19.5	280	39.2	215	371	1,010
	Vehicular and machine exhausts						
	Construction site ^b	18.5	265	22.6	118	19.9	55.2
	Other on-site ^c	1.0	15.3	16.6	96.9	1.0	1.0
	Fugitive and road dusts						
	Construction site ^b	d	d	d	d	292	811
	Other on-site ^c	d	d	d	d	57.7	146
	Total	19.5	280	39.2	215	371	1,010

^aIndividual values may not add up to the total because of rounding.^bIncludes concrete mix delivery.^cConstruction material delivery and worker commuting.^dNot applicable.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: PNL 1991.

TABLE E.18 NPR Operating Source Emission Inventory for Criteria Pollutants at Hanford, by NPR Technology and Source Category

NPR Tech- nology	Source Category	Emission Rate (t/yr) ^a					
		SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
HWR	Stationary sources						
	Existing sources ^b	3.8	9.0	0	0.2	0.2	1.2
	New sources						
	Diesel generator (continuous)	3.5	52.8	0	11.5	3.8	10.5
	Diesel generator (emergency)	0.2	0.3	0	0.7	0.1	0.2
	Process	0	4.0	0	0	0	0
	Subtotal	3.7	57.1	0	12.2	3.9	10.7
	Mobile sources (on-site)						
	Exhaust	0.2	4.2	6.6	36.7	0.4	0.4
	Road dust	c	c	c	c	24.4	62.5
	Total	7.7	70.3	6.6	49.1	28.9	74.8
LWR	Stationary sources						
	Existing sources ^b	3.8	9.0	0	0.2	0.2	1.2
	New sources						
	Diesel generator (emergency)	0.4	5.9	0	0.6	0.4	1.3
	Process	0	4.6	4.3	0	0	0
	Subtotal	0.4	10.5	4.3	0.6	0.4	1.3
	Mobile sources (on-site)						
	Exhaust	0.6	7.8	11.4	62.3	0.7	0.7
	Road dust	c	c	c	c	41.5	106
	Total	4.8	27.3	15.7	63.1	42.8	109
MHTGR	Stationary sources						
	Existing sources ^b	3.8	9.0	0	0.2	0.2	1.2
	New sources						
	Diesel generator (emergency)	0.5	7.2	0	1.6	0.5	1.4
	Process	0	93.8	0	76.0	0	0
	Subtotal	0	101	0	77.6	0.5	1.4

TABLE E.18 (Cont'd)

NPR Tech- nology	Source Category	Emission Rate (t/yr) ^a					
		SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
MHTGR (Cont'd)	Mobile sources (on-site)						
	Exhaust	0.6	8.1	11.2	62.2	0.7	0.7
	Road dust	c	c	c	c	40.6	103
	Total	4.9	118	11.4	140	42.0	106

^aIndividual values may not add up to the total because of rounding.

^bIncreased use of existing boilers and incinerators under the existing permit conditions.

^cNot applicable.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: PNL 1991.

TABLE E.19 Significant Emission Rates under PSD Regulation

Pollutant	PSD Significant Emission Rate (t/yr)	Metric Equivalent Emission Rate (t/yr)
Carbon monoxide	100	90.7
Nitrogen oxides	40	36.3
Sulfur dioxide	40	36.3
Particulate matter	25	22.7
Reduced sulfur compounds (including hydrogen sulfide)	10	9.1
Ozone (volatile organic compounds)	40 ^a	36.3
Lead	0.6	0.54
Asbestos	0.007	0.0064
Beryllium	0.0004	0.00036
Mercury	0.1	0.091
Vinyl chloride	1	0.91
Fluorides	3	2.7
Sulfuric acid mist	7	6.4
Hydrogen sulfide	10	9.1
Total reduced sulfur (including hydrogen sulfide)	10	9.1
Particulate matter $\leq 10 \mu\text{m}$	15	13.6
Each regulated pollutant	Emission rate that causes an air quality impact of 1 $\mu\text{g}/\text{m}^3$ or greater (24-hour basis) in any Class I area located within 10 km of the source	

^aOf VOC.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: 40 CFR 52.21 (b)(23).

TABLE E.20 Estimated Ambient Concentration Contributions of Criteria Pollutants from Baseline Sources at Hanford, and Construction and Operation Sources for an NPR and Support Facilities at Hanford

Maximum Contributions to Ambient Concentration (µg/m ³) ^a												
Pollutant	Averaging Time	Baseline Sources	Baseline Plus Construction Sources			New Sources			Baseline Plus New Sources			
			HWR	LWR	MHTGR	HWR	LWR	MHTGR	HWR	LWR	MHTGR	
SO ₂	Annual	2	2	2	2	<0.1	<0.1	<0.1	<0.1	2	2	2
	24 h	15	19	27	19	1	2	2	2	16	17	17
	3 h	55	83	151	84	42	115	94	94	97	170	149
	1 h	120	202	299	204	127	345	282	282	247	465	402
NO ₂	Annual	3	4	7	4	0.2	<0.1	2	2	3	3	5
CO	8 h	2	42	140	43	24	23	50	26	25	52	
	1 h	9	321	699	333	411	492	929	420	501	938	
PM ₁₀	Annual	<0.5	1	<0.5	<0.5	<0.1	<0.1	<0.1	<0.5	<0.5	<0.5	<0.5
	24 h	2	34	17	26	0.7	2	2	3	4	4	
TSP	Annual	<0.5	2	1	1	<0.1	<0.1	<0.1	<0.5	<0.5	<0.5	<0.5
	24 h	4	90	46	70	2	6	5	6	10	9	

TABLE E.20 (Cont'd)

Pollutant	Averaging Time	Background Concentration ($\mu\text{g}/\text{m}^3$)	National and State ^c Ambient Air Quality Standards ($\mu\text{g}/\text{m}^3$)		Class II Maximum Allowable PSD Increment ($\mu\text{g}/\text{m}^3$)
			Primary	Secondary	
SO ₂	Annual	0.5	80	d	20
	24 h	6	365	d	91
	3 h	20	e	1,300	512
	1 h	49	1,018 ^c	c	d
NO ₂	Annual	36	100	e	25
CO	8 h	6,500	10,000	e	d
	1 h	11,800	40,000	e	d
PM ₁₀	Annual	27	50	e	d
	24 h	81	150	e	d
TSP	Annual	56	60 ^c	c	19
	24 h	356	150 ^c	c	37

^aThe concentrations and concentration increments are the maximum values among the highest values along the SR 240 or at the Hanford Site boundary. The maximum concentration levels due to total emissions from many sources are not necessarily the sum of the maximum concentration contributions from individual sources because the latter do not necessarily occur at the same receptor locations.

^bThe maximum concentration increments for HWR and MHTGR occur on SR 240, and most of the maximum concentration increments for LWR occur at the eastern Hanford boundary.

^cWashington state ambient standards are the same as the national standards except for TSP. The state standards have no distinction between the primary and secondary standards.

^dThere is no standard.

^eSame as for the primary standard.

Source: PNL 1991.

TABLE E.21 Baseline and Future Emission Rates and Maximum Site Boundary Concentration Contributions of Toxic Pollutants from NPR Support Facilities at Hanford

NPR Technology	Toxic Air Pollutant	Annual Emission Rate (kg/yr)			Maximum 24-h Source Contributions at Site Boundary ($\mu\text{g}/\text{m}^3$)			Washington State ^a Ambient Standard ($\mu\text{g}/\text{m}^3$)
		Baseline Sources	New Sources	Baseline Plus New Sources	Baseline Sources	New Sources	Baseline Plus New Sources	
HWR	Freon-113	0	20,000	20,000	0	4.5	4.5	b
	HNO ₃	0	2,900	2,900	0	0.4	0.4	16.7
	TCA	790	480	1,270	0.4	0.1	0.5	6,327
LWR	C ₃ H ₆ O	1,710	1,800	3,510	0.8	0.1	0.9	5,927
	NH ₃	8,750	930	9,680	3.9	0.1	4.0	59.9
	HNO ₃	0	12,900	12,900	0	0.6	0.6	16.7
	TCE	0	2,500	2,500	0	0.2	0.2	0.8
MHTGR	TCA	790	150	940	0.4	0.01	0.4	6,327

^a24-hour time-weighted average concentration.^bNo standard has been established.

Sources: PNL 1991; WDOE 1991.

TABLE E.22 Emission Inventories for Baseline Sources at INEL and Construction and Operation Sources for an NPR and Support Facilities at INEL

Emission Source	NPR Technology	Emission Rate (t/yr)											
		Criteria Pollutant						Toxic Pollutant ^a					
		SO ₂	NO _x	VOC ^b	CO	PM ₁₀ ^b	TSP	C ₃ H ₆ O ^b	NH ₃	Freon-113	HNO ₃	TCA ^b	TCE ^b
INEL baseline ^c		1,893	1,264	82	422	368	766	0	2.7	- ^d	2.0	0.9	e
NPR and support facilities													
Construction	HWR	49	732	107	472	470	1,170	-	-	-	-	-	-
	LWR	48	716	111	495	491	1,230	-	-	-	-	-	-
	MHTGR	41	606	78	309	359	899	-	-	-	-	-	-
Operation	HWR	7	93	15	81	53	133	-	-	20.0	2.9	0.5	-
	LWR	2	35	17	108	66	173	1.8	0.9	-	12.9	-	2.5
	MHTGR	3	118	9	147	45	119	-	-	-	-	0.2	-

^aOnly those toxic pollutants projected to be released by NPR and its support facilities are listed.

^bVOC is volatile organic compounds, PM₁₀ is particulate matter of diameter ≤10 μm, C₃H₆O is acetone, TCA is 1,1,1-trichloroethane, and TCE is trichloroethylene.

^cBaseline emissions for the peak construction year (1997) and the first year of NPR operation (2000) are estimated to be essentially the same.

^d "-" denotes no emissions.

^eData not available.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: EG&G 1991.

TABLE E.23 NPR Construction Source Emission Inventory for Criteria Pollutants at INEL, by NPR Technology and Source Category

NPR Tech- nology	Source Category	Emission Rate (t/yr) ^a					
		SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
HWR	Vehicular and machine exhausts						
	Construction site ^b	45.3	688	55.5	155	50.2	50.2
	Other on-site ^c	3.6	43.8	51.1	316	3.3	3.3
	Fugitive and road dusts						
	Construction site ^b	d	d	d	d	217	604
	Other on-site ^c	d	d	d	d	199	516
	Total	48.9	732	107	471	470	1,170
LWR	Vehicular and machine exhausts						
	Construction site ^b	44.3	668	54.5	145	48.2	48.2
	Other on-site ^c	3.9	48.5	56.4	350	3.6	3.6
	Fugitive and road dusts						
	Construction site ^b	d	d	d	d	222	615
	Other on-site ^c	d	d	d	d	217	563
	Total	48.2	716	111	495	491	1,230
MHTGR	Vehicular and machine exhausts						
	Construction site ^b	38.3	578	47.5	126	42.2	42.2
	Other on-site ^c	2.4	28.0	30.0	183	1.9	1.9
	Fugitive and road dusts						
	Construction site ^b	d	d	d	d	192	534
	Other on-site ^c	d	d	d	d	122	320
	Total	40.7	606	77.5	309	359	899

^aIndividual values may not add up to the total because of rounding.^bIncludes concrete mix delivery.^cConstruction material delivery and worker commuting.^dNot applicable.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: EG&G 1991.

TABLE E.24 NPR Operating Source Emission Inventory for Criteria Pollutants at INEL, by NPR Technology and Source Category

NPR Tech- nology	Source Category	Emission Rate (t/yr) ^a					
		SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
HWR	Stationary sources						
	Existing sources ^b	1.1	10.2	0	13.3	0	0
	New sources						
	Diesel generator (continuous)	4.6	67.7	5.5	14.7	4.9	4.9
	Diesel generator (emergency)	0.2	2.2	0.1	0.6	0.2	0.2
	Process	0	5.3	0	0	0	0
	Subtotal	4.8	75.2	5.6	15.3	5.1	5.1
	Mobile sources (on-site)						
	Exhaust	0.6	7.8	9.1	52.8	0.6	0.6
	Road dust	c	c	c	c	47.6	127
	Total	6.5	93.2	14.7	81.4	53.3	133
LWR	Stationary sources						
	Existing sources ^b	0.7	12.5	0	32.4	0	0
	New sources						
	Diesel generator (emergency)	0.6	5.3	0.1	1.4	0.5	0.5
	Process	0	7.0	4.3	1.1	0	0
	Subtotal	0.6	12.3	4.4	2.5	0.5	0.5
	Mobile sources (on-site)						
	Exhaust	0.8	10.5	12.5	73.1	0.9	0.9
MHTGR	Road dust	c	c	c	c	64.5	172
	Total	2.1	35.3	16.9	108	65.9	173
	Stationary sources						
	Existing sources ^b	1.6	9.0	0	18.2	0	0
	New sources						
	Diesel generator (emergency)	0.4	3.7	0.1	1.0	0.4	0.4
	Process	0	97.1	0	77.5	0	0
	Subtotal	0.4	101	0.1	78.5	0.4	0.4

TABLE E.24 (Cont'd)

NPR Tech- nology	Source Category	Emission Rate (t/yr) ^a					
		SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
MHTGR (Cont'd)	Mobile sources (on-site)						
	Exhaust	0.7	8.2	8.8	50.4	0.6	0.6
	Road dust	c	c	c	c	44.1	118
	Total	2.7	118	8.9	147	45.1	119

^aIndividual values may not add up to the total because of rounding.

^bIncreased use of existing boilers and incinerators under the existing permit conditions.

^cNot applicable.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: EG&G 1991.

Maximum Contributions to Ambient Concentration ($\mu\text{g}/\text{m}^3$)^a

Maximum Contributions to Ambient Concentration (µg/m ³) ^a											
Pollutant	Averaging Time	Baseline Sources	Baseline Plus Construction Sources			New Sources ^b			Baseline Plus New Sources		
			HWR	LWR	MHTGR	HWR	LWR	MHTG R	HWR	LWR	MHTGR
SO ₂	Annual 24 h	4	4	5	4	<0.1	<0.1	<0.1	4	4	4
		38	46	46	46	0.3	8	6	38	46	44
	3 h	120	141	142	142	(^c <0.1)	(^c 0.5)	(^c 0.3)	122	184	165
NO ₂	Annual	3	6	6	6	(^c 0.1)	(^c 3.3)	(^c 2.3)	3	3	3
CO	8 h 1 h	6	57	58	57	3	60	44	9	66	50
		21	95	104	108	16	294	211	37	315	232
PM ₁₀	Annual 24 h	0.3	2	2	2	<0.1	<0.1	<0.1	0.3	0.3	0.3
		2.5	53	53	47	0.3	7	5	3	10	8
TSP	Annual 24 h	0.3	4	4	4	<0.1	<0.1	<0.1	0.3	0.3	0.3
		3	124	127	112	0.3	7	5	3	10	8

TABLE E.25 (Cont'd)

Pollutant	Averaging Time	Background Concentration ($\mu\text{g}/\text{m}^3$)	National and State ^d Ambient Air Quality Standards ($\mu\text{g}/\text{m}^3$)		Maximum Allowable PSD Increment ($\mu\text{g}/\text{m}^3$)	
			Primary	Secondary	Class I	Class II
SO ₂	Annual	0.2	80	e	2	20
	24 h	8	365	e	5	91
	3 h	14	e	1,300	25	512
NO ₂	Annual	5	100	f	2.5	25
CO	8 h	g	10,000	f	e	e
	1 h	g	40,000	f	e	e
PM ₁₀	Annual	14	50	f	e	e
	24 h	32	150	f	e	e
TSP	Annual	40	75 ^h	60 ^h	5	19
	24 h	88	260 ^h	150 ^h	10	37

^aThe concentrations and concentration increments are maximum values; all maximum concentrations occur on US 20/26. The maximum concentration levels due to total emissions from many sources are not necessarily the sum of the maximum concentration contributions from individual sources because the latter do not necessarily occur at the same receptor location.

^bThe maximum concentration increments occur on US 20/26.

^cThe numbers in parentheses are the concentration increments at the Craters of the Moon National Monument (Class I area).

^dThe Idaho State ambient air quality standards are the same as the NAAQS, except for total suspended particulate (TSP).

TABLE E.25 (Cont'd)

^cThere is no standard.

^fSame as for the primary standard.

^gData are unavailable.

^hInterim state standards.

Source: EG&G 1991.

TABLE E.26 Baseline and Future Emission Rates and Maximum Site Boundary Concentration Contributions of Toxic Pollutants from NPR Support Facilities at INEL

NPR Technology	Toxic Air Pollutant	Annual Emission Rate (kg/yr)			Maximum 8-h Source Contributions at Site Boundary ($\mu\text{g}/\text{m}^3$) ^a			Proposed Idaho State ^b Ambient Concentration Limit ($\mu\text{g}/\text{m}^3$)
		Baseline Sources	New Sources	Baseline Plus New Sources	Baseline Sources	New Sources	Baseline Plus New Sources	
HWR	Freon-113	c	20,000	c	c	2.1	c	d
	HNO ₃	2,020	2,900	2,770	10	0.1	10	50
	TCA	876	480	1,356	0.2	0.02	0.2	d
LWR	C ₃ H ₆ O	22	1,800	1,822	0.3	0.03	0.3	d
	NH ₃	2,720	930	3,650	30	0.02	30	180
	HNO ₃	2,020	12,900	14,920	10	0.2	10	50
	TCE	c	2,500	c	c	0.04	c	2,700
MHTGR	TCA	876	151	1,027	0.2	0.005	0.2	d

^aAt US 20/26, about 8 km southwest of the proposed NPR site.^b8-hour time-weighted average concentration, which is equal to one-hundredth of the occupational exposure level. These are ambient concentration limits proposed for use as one of the criteria in evaluating construction permit application for a new source.^cData not available.^dNo limit has been proposed.

Sources: EG&G 1991; IDHW 1990.

TABLE E.27 Emission Inventories for Baseline Sources at SRS and Construction and Operation Sources for an NPR and Support Facilities at SRS

Emission Source	NPR Technology	Emission Rate (t/yr)									
		Criteria Pollutant						Toxic Pollutant ^a			
		SO ₂	NO _x	VOC ^b	CO	PM ₁₀ ^b	TSP	C ₃ H ₆ O ^b	NH ₃	Freon-113	HNO ₃ TCA ^b
SRS baseline ^c		7,930	5,880	398	1,320	3,610	4,790	^d	-	5.0	8.6 9.7
NPR and support facilities											
Construction											
HWR	13	124	36	803	222	563	-	-	-	-	-
LWR	14	148	43	838	275	705	-	-	-	-	-
MHTGR	16	172	25	776	205	502	-	-	-	-	-
Operation											
HWR	157	226	16	52	36	80	-	-	-	40	3.7 0.6
LWR	150	123	13	31	30	78	0.6	0.3	20	12	0.4
MHTGR	150	210	7	102	26	68	-	-	20	-	-

^aOnly those toxic pollutants projected to be released by NPR and its support facilities are listed.

^bVOC is volatile organic compounds, PM₁₀ is particulate matter of diameter ≤10 μm, C₃H₆O is acetone, TCA is 1,1,1-trichloroethane, and TCE is trichloroethylene.

^cBaseline emissions for the peak construction year (1997) and the first year of NPR operation (2000) are estimated to be essentially the same.

^d "-" denotes no emissions.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: NUS 1991.

TABLE E.28 NPR Construction Source Emission Inventory for Criteria Pollutants at SRS, by NPR Technology and Source Category

NPR Tech- nology	Source Category	Emission Rate (t/yr) ^a					
		SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
HWR	Vehicular and machine exhausts						
	Construction site ^b	11.3	110	5.0	692	24.1	26.0
	Other on-site ^c	1.4	14.3	30.7	111	1.2	1.2
	Fugitive and road dusts						
	Construction site ^b	d	d	d	d	120	340
	Other on-site ^c	d	d	d	d	76.2	196
	Total	12.7	124	35.7	803	222	563
LWR	Vehicular and machine exhausts						
	Construction site ^b	12.5	129	5.0	696	29.1	32.6
	Other on-site ^c	1.8	18.6	38.2	142	1.5	1.5
	Fugitive and road dusts						
	Construction site ^b	d	d	d	d	150	430
	Other on-site ^c	d	d	d	d	94.1	241
	Total	14.3	148	43.2	838	275	705
MHTGR	Vehicular and machine exhausts						
	Construction site ^b	14.7	161	5.0	703	31.3	35.1
	Other on-site ^c	1.0	10.9	19.7	73.2	0.8	0.8
	Fugitive and road dusts						
	Construction site ^b	d	d	d	d	120	330
	Other on-site ^c	d	d	d	d	52.5	136
	Total	15.7	172	24.7	776	205	502

^aIndividual values may not add up to the total because of rounding.^bIncludes concrete mix delivery.^cConstruction material delivery and worker commuting.^dNot applicable.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: NUS 1991.

TABLE E.29 NPR Operating Source Emission Inventory for Criteria Pollutants at SRS, by NPR Technology and Source Category

NPR Tech- nology	Source Category	Emission Rate (t/yr) ^a					
		SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
HWR	Stationary sources						
	Existing sources ^b	149	109	0.3	1.9	4.4	11.7
	New sources						
	Diesel generator (continuous)	7.3	111	8.8	23.9	7.9	7.9
	Diesel generator (emergency)	0.2	2.2	0.1	0.6	0.2	0.2
	Process	0	4.0	0	0	0	0
	Subtotal	7.5	117	8.9	24.5	8.1	8.1
	Mobile sources (on-site)						
	Exhaust	0.3	3.2	7.2	25.1	0.3	0.3
	Road dust	c	c	c	c	22.8	59.7
	Total	157	226	16.4	51.5	35.6	79.8
LWR	Stationary sources						
	Existing sources ^b	149	109	0.3	1.9	4.4	11.7
	New sources						
	Diesel generator (emergency)	0.6	5.5	0.1	1.4	0.6	0.6
	Process	0	4.6	4.3	0	0	0
	Subtotal	0.6	10.1	4.4	1.4	0.6	0.6
	Mobile sources (on-site)						
	Exhaust	0.4	3.5	7.9	27.5	0.3	0.3
	Road dust	c	c	c	c	24.6	65.1
	Total	150	123	12.6	30.8	29.9	77.7
MHTGR	Stationary sources						
	Existing sources ^b	149	109	0.3	1.9	4.4	11.7
	New sources						
	Diesel generator (emergency)	0.4	3.6	0.1	0.9	0.4	0.4
	Process	0	93.8	0	76.0	0	0
	Subtotal	0.4	97.4	0.1	76.9	0.4	0.4

TABLE E.29 (Cont'd)

NPR Tech- nology	Source Category	Emission Rate (t/yr) ^a					
		SO ₂	NO _x	VOC	CO	PM ₁₀	TSP
MHTGR (Cont'd)	Mobile sources (on-site)						
	Exhaust	0.3	3.2	6.5	23.5	0.3	0.3
	Road dust	c	c	c	c	20.7	55.2
	Total	150	210	6.9	102	25.8	67.6

^aIndividual values may not add up to the total because of rounding.

^bIncreased use of existing boilers and incinerators under the existing permit conditions.

^cNot applicable.

Conversion factor: 1 t/yr = 1.102 tons/yr.

Source: NUS 1991.

TABLE E.30 Estimated Ambient Concentration Contributions of Criteria Pollutants from Baseline Sources at SRS and Construction and Operation Sources for an NPR and Support Facilities at SRS

Maximum Contributions to Ambient Concentration ($\mu\text{g}/\text{m}^3$) ^a												
Pollutant	Averaging Time	Baseline Sources	Baseline Plus Construction Sources				New Sources				Baseline Plus New Sources	
			HWR	LWR	MHTGR		HWR	LWR	MHTGR		HWR	MHTGR
SO ₂	Annual	11	13	11	11		<0.5	<0.5	<0.5		11	11
	24 h	232	232	232	232		<0.5	<0.5	<0.5		232	232
	3 h	1,074	1,074	1,074	1,074		<0.5	<0.5	<0.5		1,074	1,074
NO ₂	Annual	16	16	16	16		1	<1	<1		16	16
CO	8 h	38	44	138	115		<35	<35	35		38	127
	1 h	154	195	195	461		<240	<240	240		195	381
PM ₁₀	Annual	1	2	2	2		<0.5	<0.5	<0.5		1	1
	24 h	17	98	84	87		<0.5	<0.5	<0.5		17	17
TSP	Annual	2	2	2	2		<0.5	<0.5	<0.5		2	2
	24 h	c	c	c	c		c	c	c		c	c

TABLE E.30 (Cont'd)

Pollutant	Averaging Time	Background Concentration ($\mu\text{g}/\text{m}^3$)	National and State ^c Ambient Air Quality Standards ($\mu\text{g}/\text{m}^3$)		Class II Maximum Allowable PSD Increment ($\mu\text{g}/\text{m}^3$)
			Primary	Secondary	
SO ₂	Annual	5	80	e	20
	24 h	34	365	e	91
	3 h	48	e	1,300	512
NO ₂	Annual	6	100	f	25
CO	8 h	c	10,000	f	e
	1 h	c	40,000	f	e
PM ₁₀	Annual	27	50	f	e
	24 h	47	150	f	e
TSP	Annual	27	75 ^d	d	19
	24 h	c	e	e	37

^aThe concentrations and concentration increments are the maximum values among the highest values at the SRS boundary. The maximum concentration levels due to total emissions from many sources are not necessarily the sum of the maximum concentration contributions from individual sources because the latter do not necessarily occur at the same receptor locations.

^bMost of the maximum concentration increments occur at the southwestern SRS boundary near the D-Area power houses.

^cNot estimated.

TABLE E.30 (Cont'd)

^dSouth Carolina ambient standards are the same as the national standards except for TSP. The state standards have no distinction between the primary and secondary standards.

^eThere is no standard.

^fSame as for the primary standard.

Source: NUS 1991.

TABLE E.31 Baseline and Future Emission Rates and Maximum Site Boundary Concentration Contributions of Toxic Pollutants from NPR Support Facilities at SRS

NPR Technology	Toxic Air Pollutant	Annual Emission Rate (kg/yr)				Maximum 24-h Source Contributions at Site Boundary ($\mu\text{g}/\text{m}^3$) ^a				Proposed South Carolina ^a Ambient Standard ($\mu\text{g}/\text{m}^3$)
		Baseline Sources	New Sources	Baseline Plus New Sources		Baseline Sources	New Sources	Baseline Plus New Sources		
HWR	Freon-113	182,800	20,000	202,800		23.3	16.0	39.3		b
	HNO ₃	8,636	2,900	11,536		1.1	1.0	2.1		125
	TCA	10,100	480	10,580		1.2	0.1	1.3		b
LWR	C ₃ H ₆ O	0	600	600		0	0.4	0.4		b
	NH ₃	0	310	310		0	0.1	0.1		b
	Freon-113	182,800	20,000	202,800		23.3	10.1	33.4		b
	HNO ₃	8,636	12,000	20,636		1.1	0.7	1.8		125
	TCA	10,100	300	10,400		1.3	0.1	1.4		b
MHTGR	Freon-113	182,800	20,000	202,800		23.3	11.3	34.6		b

^a24-hour average concentration.^bNo standard has been proposed.

Sources: NUS 1991; SCDHEC 1987b.

TABLE E.32 Design Parameters for Mechanical-Draft Recirculating Cooling Towers for an NPR at Hanford

Parameter	NPR Technology		
	HWR	LWR	MHTGR
Maximum heat dissipation rate (MWt)	3,300	3,800	2,800
Number of towers	3	3	2
Effective tower height (m)	19.8	19.8	19.8
Exit diameter of tower (m)	74.1	74.1	74.1
Number of fans ^a per tower	19	19	19
Circulating water flow rate (kg/s) ^{b,c}	18,700	39,900	30,100
Tower influent air flow rate (kg/s) ^c	12,500	32,700	16,100
Design ΔT (range) ($^{\circ}C$)	42.2	22.8	22.2

^aThe power rating per fan is 150 hp.

^bThe total dissolved solids concentration of the circulating water based on five cycles of concentration is 849 mg/L, and the drift rate for the cooling tower is 0.02% of the circulating water flow rate.

^cThe flow rates are for the tower system consisting of two or three towers.

Conversion factors: 1 m = 3.281 ft; 1 kg/s = 2.205 lb/s; $^{\circ}F = (^{\circ}C \times 9/5) + 32$;
 $\Delta^{\circ}F = \Delta^{\circ}C \times 9/5$.

Source: Chun and Coke 1991.

TABLE E.33 Estimated Environmental Impacts of Operating Mechanical-Draft Cooling Towers for an NPR at Hanford

NPR Technology	Annual Rate of Drift Water Deposition by Distance from the Towers in Kilometers (mm/mo)					Annual Rate of Drift Salt Deposition by Distance from the Towers in Kilometers (kg/km ² -mo)					Annual Frequency of Visible Plume Occurrence by Plume Length in Kilometers (% of time) ^a				
	0.3	0.5	1.0	2.0	3.0	0.3	0.5	1.0	2.0	3.0	0.5	1.0	2.0	3.0	5.0
HWR															
Maximum ^b	1.8	0.8	0.4	0.2		2,900(S)	970	440	290		11	6	4	1(S) ^c	0
Average	0.6	0.4	0.1	<0.1		900	540	170	110		4	2	2	1	0
Sum ^d	e	e	e	e		e	e	e	e		66	37	25	8	0
LWR															
Maximum ^b	1.7(S) ^d	0.5	0.2	0.3		3,100	830(S)	310	370		7	4	3	0.1(SE)	0
Average	0.8	0.3	0.1	0.1		1,600	400	150	160		3	2	1	<0.1	0
Sum ^d	e	e	e	e		e	e	e	e		46	28	19	0.3	0
MHTGR															
Maximum ^b	3.6	2.2	0.9	0.2		4,000	2,700	1,100	360		8	7	4	1(S)	0
Average	1.6	1.0	0.4	<0.1		2,000	1,200	540	100		3	2	1	0.3	0
Sum ^d	e	e	e	e		e	e	e	e		51	31	23	5	0

^aUp to the length indicated.^bMaximum value occurs north of the cooling towers unless otherwise noted.^cLetter(s) in parentheses indicates the direction from the cooling towers where the maximum value occurs.^dFor all directions.^eNot applicable.Conversion factors: 1 km = 0.6213 mi; 1 kg/km²-mo = 5.707 lb/mi²-mo.

Source: Chun and Coke 1991.

TABLE E.34 Design Parameters for Mechanical-Draft Recirculating Cooling Towers for an NPR at INEL

Parameter	NPR Technology		
	HWR	LWR	MHTGR
Maximum heat dissipation rate (MWt)	3,300	3,800	2,800
Number of towers	3	3	2
Effective tower height (m)	19.8	19.8	19.8
Exit diameter of tower (m)	74.1	74.1	74.1
Number of fans ^a per tower	19	19	19
Circulating water flow rate (kg/s) ^{b,c}	18,700	44,500	33,600
Tower influent air flow rate (kg/s) ^c	10,600	26,600	14,000
Design ΔT (range) ($^{\circ}\text{C}$)	42.2	22.8	22.2

^aThe power rating per fan is 150 hp.

^bThe total dissolved solids concentration of the circulating water based on cycles of concentration is 2,565 mg/L, and the drift rate for the cooling towers is 0.02% of the circulating water flow rate.

^cThe flow rates are for the tower system consisting of two or three towers.

Conversion factors: 1 m = 3.281 ft; 1 kg/s = 2.205 lb/s; $^{\circ}\text{F} = (^{\circ}\text{C} \times 9/5) + 32$;
 $\Delta^{\circ}\text{F} = \Delta^{\circ}\text{C} \times 9/5$.

Source: Chun and Coke 1991.

TABLE E.35 Estimated Environmental Impacts of Operating Mechanical-Draft Cooling Towers for an NPR at INEL

NPR Technology	Annual Rate of Drift Water Deposition by Distance from the Towers in Kilometers (mm/mo)					Annual Rate of Drift Salt Deposition by Distance from the Towers in Kilometers (kg/km ² -mo)					Annual Frequency of Visible Plume Occurrence by Plume Length in Kilometers (% of time) ^a				
	0.3	0.5	1.0	2.0		0.3	0.5	1.0	2.0		1.0	5.0	10	15	21
HWR															
Maximum ^b	2.6	2.6	0.3	0.1		7,100	7,200	1,100	510		10	6	4	4	0
Average	0.9	1.0	0.1	<0.1		2,500	2,900	430	170		3	2	0.6	0.6	0
Sum ^c	d	d	d	d		d	d	d	d		56	27	9	9	0
LWR															
Maximum ^b	2.6	5.2	0.6	0.2		7,000	14,000	1,700	810		9	7	5	0.4	0
Average	1.1	1.9	0.2	<0.1		3,100	5,200	550	240		3	2	1	<0.1	0
Sum ^c	d	d	d	d		d	d	d	d		51	35	19	0.7	0
MHTGR															
Maximum ^b	4.6	5.3	0.9	0.2		5,300	6,400	1,600	230		9	7	6	2	0
Average	1.8	2.1	0.4	<0.1		2,000	2,400	590	75		3	2	2	0.3	0
Sum ^c	d	d	d	d		d	d	d	d		52	31	25	5	0

^aUp to the length indicated.^bMaximum value occurs northeast of the cooling towers.^cFor all directions.^dNot applicable.Conversion factors: 1 km = 0.6213 mi; 1 kg/km²-mo = 5.707 lb/mi²-mo.

Source: Chun and Coke 1991.

TABLE E.36 Design Parameters for Mechanical-Draft and Natural-Draft Recirculating Cooling Towers for an NPR at SRS

Parameter	Mechanical-Draft			Natural-Draft		
	HWR	LWR	MHTGR	HWR	LWR	MHTGR
Maximum heat dissipation rate (MW/t)	3,300	3,800	2,800	3,300	3,800	2,800
Number of towers	8	12	9	2	2	2
Effective height of tower (m)	20	20	20	138	155	127
Exit diameter of tower (m)	61 ^a	61 ^a	61 ^a	96	102	93
Number of fans ^b per tower	12	12	12	c	c	c
Circulating water flow rate (kg/s) ^{d,e}	18,700	39,900	30,100	18,700	40,000	30,100
Tower influent air flow rate (kg/s) ^e	23,600	33,800	25,300	f	f	f
Design ΔT (range) (°C)	42.2	22.8	22.2	42.2	22.8	22.2

^aThe length of a side of square which is the shape of tower housing at the top.

^bThe power rating per fan is 255 hp.

^cNot applicable.

^dThe total dissolved solids concentration of the circulating water based on 2.5 cycles of concentration is 157 mg/L, and the drift rate for the cooling towers is 0.02% of the circulating water flow rate.

^eThe flow rates are for the tower system consisting of two or three towers.

^fFor natural-draft cooling towers, the rate is a function of the wet-bulb temperature and relative humidity because the density difference between the inlet and outlet air is the driving force and the exit air is generally unsaturated.

Conversion factors: 1 m = 3.281 ft; 1 kg/s = 2.205 lb/s; °F = (°C × 9/5) + 32; Δ°F = Δ°C × 9/5.

Source: NUS 1991.

TABLE E.37 Estimated Environmental Impacts of Operating Mechanical-Draft Cooling Towers for an NPR at SRS

NPR Technology	Annual Rate of Drift Water Deposition by Distance from the Towers in Kilometers (mm/mo)					Annual Rate of Drift Salt Deposition by Distance from the Towers in Kilometers (kg/km ² ·mo)					Annual Frequency of Visible Plume Occurrence by Plume Length in Kilometers (% of time) ^a				
	0.3	0.5	1.0	2.0		0.3	0.5	1.0	2.0		0.5	1.0	5.0	8.0	10.0
HWR															
Maximum ^b	1.4 (ENE) ^c	0.9 (NE)	0.1 (E)	<0.1 (E)		240 (ENE)	270 (ENE)	25 (ENE)	19 (ENE)		3	2	2	0	0
Average	0.4	0.3	<0.1	<0.1		65	77	14	8		2	1	1	0	0
Sum ^d	e	e	e	e		e	e	e	e		27	20	16	0	0
LWR															
Maximum ^b	10 (ENE)	1.5 (NE)	0.2 (N)	<0.1 (E)		3,200 (ENE)	330 (NE)	74 (ENE)	13 (E)		2	2	2	0.7	0
Average	2.2	0.4	<0.1	<0.1		680	80	26	8		1	1	0.7	0.3	0
Sum ^d	e	e	e	e		e	e	e	e		19	17	11	4	0
MHTGR															
Maximum ^b	2.2 (N)	2.4 (ENE)	0.2 (N)	<0.1 (N)		540 (N)	550 (ENE)	32 (N)	19 (N)		3	2	2	0.7	0
Average	0.7	0.8	0.1	<0.1		160	160	18	11		1	1	1	0.3	0
Sum ^d	e	e	e	e		e	e	e	e		19	16	11	5	0

^aUp to the length indicated.^bMaximum value occurs south of the cooling towers unless otherwise noted.^cLetter(s) in parentheses indicates the direction from the cooling towers where the maximum value occurs.^dFor all directions.^eNot applicable.Conversion factors: 1 km = 0.6213 mi; 1 kg/km²·mo = 5.707 lb/mi²·mo.

Source: NUS 1991.

TABLE E.38 Estimated Environmental Impacts of Operating Natural-Draft Cooling Towers for an NPR at SRS

NPR Technology	Annual Rate of Drift Water Deposition by Distance from the Towers in Kilometers (mm/mo)					Annual Rate of Drift Salt Deposition by Distance from the Towers in Kilometers (kg/km ² -mo)					Annual Frequency of Visible Plume Occurrence by Plume Length in Kilometers (% of time) ^a				
	0.3	0.5	1.0	2.0		0.3	0.5	1.0	2.0		0.5	1.0	5.0	8.0	10.0
HWR															
Maximum ^b	<0.1 (N) ^c	<0.1 (N)	<0.1 (N)	<0.1 (E)		7 (N)	7 (N)	4 (N)	1 (E)		6	4	3	1	0.6 (E)
Average	<0.1	<0.1	<0.1	<0.1		5	5	3	1		3	2	2	0.6	0.4
Sum ^d	e	e	e	e		e	e	e	e		54	32	27	9	6
LWR															
Maximum ^b	<0.1	<0.1	<0.1	<0.1		19	20	13	3 (N)		6	4	3	1	0.5 (SSE)
Average	<0.1	<0.1	<0.1	<0.1		13	15	9	2		4	2	1	0.7	0.2
Sum ^d	e	e	e	e		e	e	e	e		56	32	23	11	4
MHTGR															
Maximum ^b	0.1	0.1	<0.1 (E)	<0.1 (E)		23	29	13 (E)	3 (E)		5	3	3	1	0.6 (E)
Average	<0.1	<0.1	<0.1	<0.1		16	20	9	1		3	2	1	0.7	0.4
Sum ^d	e	e	e	e		e	e	e	e		46	28	23	11	6

^aUp to the length indicated.

^bMaximum value occurs south of the cooling towers unless otherwise noted.

^cLetter(s) in parentheses indicates the direction from the cooling towers where the maximum value occurs.

^dFor all directions.

^eNot applicable.

Conversion factors: 1 km = 0.6213 mi; 1 kg/km²-mo = 5.707 lb/mi²-mo.

Source: NUS 1991.

TABLE E.39 Major Noise Sources during the Peak NPR Construction Period at Hanford and Their Estimated Sound Power Levels and Maximum Noise Impacts

Source	Capacity (hp/unit)	Sound Power Level (dBA)	Number of Units		Outdoors		Indoors ^b	
					HWR/ MHTGR	LWR (WNP-1)	HWR/ MHTGR	LWR (WNP-1)
			HWR/ MHTGR	LWR (WNP-1)	Noise Level in L_{eq} (1 h) at the Nearest Receptor Location (dBA) ^a			
Crawler crane	700	124.8	4	1	25.5	26.5	7.1	15.9
Crawler crane	460	123.8	2	2	21.5	28.7	3.1	18
Bulldozer	460	123.8	2	1	21.5	25.7	3.1	15
Bulldozer	335	121.8	1	0	16.5	--	-1.9	--
Truck or equivalent	270	114.0	130	35	17.2	30.2	-1.2	19.1
Miscellaneous equipment ^d	50-270 ^c	100- 117 ^e	160	45	18.2	25.4	-0.2	15
Intrusive noise levels due to all sources					29.1	34.7	10.7	23.5
CNR model, predicted community response					-----No reaction-----			
IAP model								
Intrusion levels (dB)							1.0	14.5
Predicted individual response							-----No annoyance-----	
Ambient noise levels ^e					32.6 (36.7) ^f	36.3 (38.2)	14.2 (18.4)	23.8 (22.4)

TABLE E.39 (Cont'd)

^a L_{eq} (1 h) is the hourly L_{eq} and is predicted for a receptor located at the Hanford Site boundary 8.5 km southwest from the NPR site (for the HWR/MHTGR technology), and at the town of Mattew's Corner 9.6 km east of the WNP-1 site (for indoor) and at a farmhouse 5.8 km east of the WNP-1 site and 2.2 km east of the Hanford Site boundary (for indoor) (for the LWR technology).
^b Indoor intrusive noise levels are determined for a house in a cold climate with the windows open (Sutherland, Braden, and Colman 1973).
^c Range of values.
^d Includes bulldozers, portable electric generators, pumps, backhoes, light trucks, scrapers, forklifts, air breathing machines, air compressors, paver, grader, front-end loaders, and tractors.
^e The ambient noise level combining both the background and intrusive noise levels. The background noise level is 30.0 dBA for the site boundary and 31.1 dBA for the farmhouse near WNP-1 (PNL 1991).
^f The values in parentheses are in L_{dn} .
Source: Chun and Rabchuk 1991.

TABLE E.40 Major Noise Sources during NPR Operation Period at Hanford and Their Estimated Sound Power Levels and Maximum Noise Impacts

Source	Capacity (hp/unit)	Sound Power Level (dBA)	Number of Units			Noise Level in $L_{eq}(1h)$ at the Nearest Receptor Location (dBA) ^a			
			Indoors ^b			Outdoors			
			HWR	LWR (WNP-1)	MHTGR	HWR	LWR (WNP-1)	MHTGR	MHTGR
Mechanical-draft cooling tower fan	150/fan	105.5	57	57	38	20.7	25.5	18.9	2.3
Diesel generator	150	102.1	6	0	0	23.0	--	--	4.6
Miscellaneous equipment ^d	50-9,100 ^c	95-110 ^c	18	19	20	12.1	21.3	12.1	-6.3
-----No reaction-----									
Intrusive noise levels due to all sources						24.4	26.9	19.1	6.0
CNR model, predicted community response									16.5
IAP model									0.7
Intrusion levels (dB)									1.0
Predicted individual response									10.0
Ambient noise levels ^e									-----No annoyance-----
						31.1	32.4	30.3	12.7
						(37.5) ^f	(38.8)	(36.7)	(19.1)
									18.0
									(24.4)
									(18.3)

^a $L_{eq}(1h)$ is the hourly L_{eq} and is predicted for a receptor located at the Hanford Site boundary 8.5 km southwest of the NPR site (for the HWR/MHTGR technology), and at the town of Mattew's Corner 9.6 km east of the WNP-1 Site (for outdoor) and at a farmhouse 5.8 km east of the WNP-1 site and 2.2 km east of the Hanford Site boundary (for indoor) (for the LWR technology).

^bIndoor intrusive noise levels are determined for a house in a cold climate with the windows open (Sutherland, Braden, and Colman 1973).

^cRange of values.

^dIncludes diesel generators, water pumps, transformers, and air compressors.

^eThe ambient noise level combining both the background and intrusive noise levels. The background noise level is 30.0 dBA for the site boundary and 31.1 dBA for the farmhouse near WNP-1 (PNL 1991).

^fThe values in parentheses are in L_{dn} .

Source: Chun and Rabchuk 1991.

TABLE E.41 Estimated Hourly Traffic Volumes and Traffic Noise Levels along the Highways Leading to the NPR Site for Baseline and NPR Construction and Operation Periods at Hanford^a

Period	Route Name	Off-Peak/ Peak ^c	Traffic Volume (vehicles/hr)				Noise Levels in L _{eq} (1h) (dBA) ^b				
			Base-line	NPR Technology			Base-line	NPR Technology			
				HWR	LWR	MHTGR		HWR	LWR	MHTGR	Maximum Increase ^d
Construction	SR24	Off peak Peak	91	91	91	91	62.0	62.0	62.0	62.0	0
			91	343	319	208	62.0	64.6	64.4	63.4	2.6
	SR 240	Off peak Peak	571	574	573	579	70.2	70.4	70.3	70.6	0.4
			571	2,839	2,619	1,626	70.2	73.5	73.3	72.0	3.3
Operation	SR 24	Off peak Peak	91	91	91	91	62.0	62.0	62.0	62.0	0
			300	350	386	382	65.7	66.0	66.2	66.2	1.5
	SR 240	Off peak Peak	571	572	573	582	70.2	70.2	70.2	70.5	0.3
			2,239	2,689	3,009	2,975	74.1	74.5	74.7	74.7	0.6

^aThe baseline years for the peak NPR construction and operation periods are 1997 and 2000, respectively.

^bL_{eq} (1h) is the hourly L_{eq} in dBA and is predicted for a receptor located 15 m from the road edge.

^cThe peak hour traffic occurs in the hour during which the NPR work force commutes to the site. For the construction period, the peak occurs at 5-6 p.m. For the operation period, the peak occurs at 4-5 p.m.

^dThe maximum increase in noise level above the baseline among the three technologies.

Source: Chun and Rabchuk 1991.

TABLE E.42 Major Noise Sources during the Peak NPR Construction Period at INEL and Their Estimated Sound Power Levels and Maximum Noise Impacts

Source	Capacity (hp/unit)	Sound Power Level (dBA)	Number of Units	Noise Level in L_{eq} (1 h) at the Nearest Receptor Location (dBA) ^a	
				Outdoors	Indoors ^b
Backhoe	325	121.8	10	19.2	0.8
Bulldozer	460	123.8	4	17.2	-1.2
Miscellaneous equipment ^c	3-300 ^d	100-117 ^d	130	10.7	-7.7
Intrusive noise levels due to all sources				21.7	3.3
CNR model, predicted community response				No reaction	
IAP model Intrusion levels (dB) Predicted individual response					0 No annoyance
Ambient noise levels ^e				30.6 (36.5) ^f	12.2 (18.1)

^a L_{eq} (1 h) is the hourly L_{eq} and is predicted for a receptor located at the INEL boundary 12.8 km south of the NPR site.

^bIndoor intrusive noise levels are determined for a house in a cold climate with the windows open (Sutherland, Braden, and Colman 1973).

^cIncludes drills, trucks, front-end loaders, pile drivers, pavers, graders, concrete batch plant, concrete vibrators, vibratory conveyor, and mobile generators.

^dRange of values.

^eThe ambient noise level combining both the background and intrusive noise levels. The background noise level is 30.0 dBA at the INEL boundary (EG&G 1991).

^fThe values in parentheses are in L_{dn} .

Source: Chun and Rabchuk 1991.

TABLE E-43 Major Noise Sources during the NPR Operation Period at INEL and Their Estimated Sound Power Levels and Maximum Noise Impacts

Noise Level in L _{eq} (1h) at the Nearest Receptor Location (dBA) ^a											
Source	Capacity (hp/unit)	Sound Power Level (dBA)	Number of Units			Outdoors		Indoors ^b			
			HWR	LWR (WNP-1)	MHTGR	HWR	LWR	MHTGR	HWR	LWR	MHTGR
Air blast heat exchanger	18/fan	96.2	0	0	1,152	--	--	--	--	--	1.1
Mechanical-draft cooling tower fan	150/fan	105.5	57	57	0	14.2	14.2	--	--	-4.2	--
Diesel generator	150	102.1	6	0	0	18.1	--	--	--	--	--
Miscellaneous equipment ^d	50-9,100 ^c	95-110 ^c	18	19	20	-2.3	-1.2	11.2	-20.7	-19.6	-7.2

Intrusive noise levels due to all sources											
			19.6			14.3		20.1	1.2	-4.1	1.7

CNR model, predicted community response											
-----No reaction-----											
IAP model											
Intrusion levels (dB)											
Predicted individual response											
-----No annoyance-----											
Ambient noise levels ^e											
			30.4			30.1		30.4	12.0	11.7	12.0
			(36.5) ^f			(36.4)		(36.5)	(18.1)	(18.0)	(18.1)
									0	0	0

^a L_{eq} (1 h) is the hourly L_{eq} and is predicted for a receptor located at the INEL boundary 12.8 km south of the NPR site.

^{b7}Indoor intrusive noise levels are determined for a house in a cold climate with the windows open (Sutherland, Braden, and Colman 1973).

Range of values.

^dincludes diesel generators, water pumps, 60- or 100-MVA transformers, and air compressors.

*The ambient noise level combining both the background and intrusive noise levels. The background noise level is 30.0 dBA for the INEL boundary (EG&G 1991).

^fThe values in parentheses are in L_{dn} .

Source: Chun and Rabchuk 1991.

TABLE E.44 Estimated Hourly Traffic Volumes and Traffic Noise Levels along the Highways Leading to the NPR Site for Baseline and NPR Construction and Operation Periods at INEL^a

		Traffic Volume (vehicles/hr)				Noise Levels in L _{eq} (1h) (dBA) ^b					
Period	Route Name	Off-Peak/ Peak ^c	Base-line	NPR Technology			Base-line	NPR Technology			Maximum Increase ^d
				HWR	LWR	MHTGR		HWR	LWR	MHTGR	
Construction	US 20	Off peak Peak	171	173	172	173	65.2	65.6	65.4	65.6	0.4
			171	1,431	1,568	890	65.2	70.0	70.3	68.5	5.1
	US 26	Off peak Peak	112	113	113	114	63.3	63.5	63.5	63.8	0.5
			112	868	951	543	63.3	67.8	68.1	66.4	4.8
Operation	US 20/26	Off peak Peak	67	68	68	70	60.9	61.4	61.4	62.3	1.4
			67	571	626	354	60.9	65.9	66.2	64.4	5.3
	US 20	Off peak Peak	147	148	148	151	65.1	65.1	65.1	65.5	0.4
			277	527	627	495	67.7	68.6	68.9	68.5	1.2
	US 26	Off peak Peak	98	99	99	102	63.1	63.2	63.2	63.8	0.7
			184	334	394	315	65.9	66.7	67.0	66.6	1.1
	US 20/26	Off peak Peak	58	59	59	63	60.8	61.0	61.0	62.0	1.2
			109	209	249	196	63.6	64.5	64.8	64.4	1.2

^aThe baseline years for the peak NPR construction and operation periods are 1997 and 2000, respectively.^b L_{eq} (1h) is the hourly L_{eq} in dBA, and is predicted for a receptor located 15 m from the road edge.^cThe peak hour traffic occurs in the hour during which the NPR work force commutes to the site. For the construction period, the peak occurs at 5-6 p.m. For the operation period, the peak occurs at 4-5 p.m.^dThe maximum increase in noise level above the baseline among the three technologies.

Source: Chun and Rabchuk 1991.

TABLE E.45 Major Noise Sources during the Peak NPR Construction Period at SRS and Their Estimated Sound Power Levels and Maximum Noise Impacts

Source	Capacity (hp/unit)	Sound Power Level (dBA)	Number of Units	Noise Level in L_{eq} (1 h) at the Nearest Receptor Location (dBA) ^a	
				Outdoors	Indoors ^b
Crawler crane	700	124.8	4	-4.3	-15.5
Crawler crane	460	123.8	2	-8.3	-19.5
Bulldozer	460	123.8	2	-8.3	-19.5
Bulldozer	335	121.8	1	-13.3	-24.5
Truck or equivalent	270	114.0	130	-11.5	-22.7
Miscellaneous equipment ^d	50-270 ^c	100- 117 ^c	160	-10.3	-21.5
Intrusive noise levels due to all sources				-0.5	-11.7
CNR model, predicted community response				No reaction	
IAP model					
Intrusion levels (dB)				0	
Predicted individual response				No annoyance	
Ambient noise levels ^e				50.0 (57.0) ^f	38.8 (45.2)

^a L_{eq} (1 h) is the hourly L_{eq} and is predicted for a receptor located at the SRS boundary 11.2 km north from the NPR site. Negative dB values can occur because the values are sound pressure ratios in log scale.

^bIndoor intrusive noise levels are determined for a house in a warm climate with the windows open (Sutherland, Braden, and Colman 1973).

^cRange of values.

^dIncludes bulldozers, portable electric generators, pumps, backhoes, light trucks, scrapers, forklifts, air breathing machines, air compressors, paver, grader, front-end loaders, and tractors.

^eThe ambient noise level combining both the background and intrusive noise levels. The background is taken to be L_{eq} (1 h) = 50.0 dBA (NUS 1991).

^fThe values in parentheses are in L_{dn} .

Source: Chun and Rabchuk 1991.

TABLE E.46 Major Noise Sources during the NPR Operation Period at SRS and Their Estimated Sound Power Levels and Maximum Noise Impacts

Source	Capacity (hp/unit)	Sound Power Level (dBA)	Number of Units			Noise Level in L_{eq} (1 h) at the Nearest Receptor Location (dBA) ^a					
						Outdoors			Indoors ^b		
			HWR	LWR	MHTGR	HWR	LWR	MHTGR	HWR	LWR	MHTGR
Mechanical-draft cooling tower fan	255/fan	109.1	96	144	108	12.5	14.2	13	1.3	3	1.8
Diesel generator	150	102.1	6	0	0	1.1	--	--	-10.1	--	--
Miscellaneous equipment ^d	50-9,100 ^c	95-110 ^c	18	19	20	-16.7	-14.3	-0.9	-27.9	-25.5	-12.1

Intrusive noise levels due to all sources						12.8	14.2	13.2	1.6	3.0	2.0
CNR model, predicted community response						-----No reaction-----					
IAP model											
Intrusion levels (dB)											
Predicted individual response											
Ambient noise levels ^e						50.0 (57.0) ^f	50.0 (57.0)	50.0 (57.0)	38.8 (45.2)	38.8 (45.2)	38.8 (45.2) ^f

^a L_{eq} (1h) is the hourly L_{eq} and is predicted for a receptor located at the SRS boundary 11.2 km north of the NPR site.^bIndoor intrusive noise levels are determined for a house in a warm climate with the windows open (Sutherland, Braden, and Colman 1973).^cRange of values.^dIncludes diesel generators, water pumps, 60- or 100-MVA transformers, and air compressors.^eThe ambient noise level combining both the background and intrusive noise levels. The background level is taken to be L_{eq} (1 h) = 50 dBA (NUS 1991).^fThe values in parentheses are in L_{dn} .

Source: Chun and Rabchuk 1991.

TABLE E.47 Estimated Hourly Traffic Volumes and Traffic Noise Levels along the Highways Leading to the NPR Site for Baseline and NPR Construction and Operation Periods at SRS^a

Period	Route Name	Off-Peak/ Peak ^c	Traffic Volume (vehicles/hr)				Noise Levels in L _{eq} (1 h) (dBA) ^b				Maximum Increased ^d
			Base-line	NPR Technology			Base-line	NPR Technology			
				HWR	LWR	MHTGR		HWR	LWR	MHTGR	
Construction	SR125N	Off peak	434	436	436	440	66.5	66.8	66.8	67.2	0.7
		Peak	434	1,434	1,702	1,079	66.5	69.9	70.5	69.0	4.0
	SR 19	Off peak	404	404	404	406	63.9	63.9	63.9	64.3	0.4
		Peak	404	1,204	1,418	920	63.9	68.2	68.9	67.1	5.0
	SR 19 @100 ft	Off peak	404	404	404	406	61.1	61.1	61.1	61.6	0.5
		Peak	404	1,204	1,418	920	61.1	65.3	66.0	64.2	4.9
	SR 64	Off peak	66	66	66	66	57.6	57.6	57.6	57.6	0
		Peak	66	266	320	195	57.6	62.1	62.8	61.0	5.2
	US 1	Off peak	750	752	752	756	71.0	71.1	71.1	71.2	0.1
		Peak	750	750	750	750	71.0	71.0	71.0	71.0	0
Operation	SR 125N	Off peak	434	436	436	442	66.5	66.6	66.6	67.0	0.5
		Peak	2,527	2,777	2,802	2,734	72.4	72.7	72.8	72.7	0.4
	SR 19	Off peak	404	404	404	406	63.9	63.9	63.9	64.1	0.2
		Peak	2,078	2,278	2,298	2,244	70.5	70.9	70.9	70.8	0.4
	SR 19 @100 ft	Off peak	404	404	404	406	61.1	61.1	61.1	61.3	0.2
		Peak	2,078	2,278	2,298	2,244	67.5	67.9	68.0	67.9	0.5
	SR 64	Off peak	66	66	66	66	57.6	57.6	57.6	57.6	0
		Peak	286	336	341	327	62.9	63.4	63.5	63.3	0.6
	US 1	Off peak	750	752	752	758	71.0	71.0	71.0	71.1	0.1
		Peak	3,613	3,613	3,613	3,613	77.8	77.8	77.8	77.8	0

^aThe baseline years for the peak NPR construction and operation periods are 1997 and 2000, respectively.

^bL_{eq} (1 h) is the hourly L_{eq} in dBA, and is predicted for a receptor located 15 m from the road edge, except where otherwise noted.

^cThe peak hour traffic occurs in the hour during which the NPR workforce commutes to the site. For the construction period, the peak occurs at 5-6 p.m. For the operation period, the peak occurs at 4-5 p.m.

^dThe maximum increase in noise level above the baseline among the three technologies.

Source: Chun and Rabchuk 1991.

**TABLE E.48 Composite Noise Rating (CNR) Scale of
Community Complaint Reaction to Intrusive Noise**

CNR Category	Description of Complaint Activity
A	No complaints among community members or to noise source operators
B/C	A few spontaneous complaints, usually among residents, but not to noise source operators
D	Sporadic individual complaints to noise source operators or to authorities, irregular in number, time, and area of origin
E	Widespread individual complaints regularly received; single threat of legal action; initiation of group organization to register complaints with authorities
F/G	Several threats of legal action; organized, strong group appeals to authorities to stop noise
H+	Vigorous group and community action highly organized and supported communitywide; police authority, legal action, or other public authority is employed to oppose noise-source operation

Sources: Stevens, Rosenblith, and Bolt 1953; Parrack 1957; Bolt, Beranek, and Newman 1984.

TABLE E.49 Individual Annoyance Prediction (IAP) Scale Related to Intrusion Level^a

Intrusion Level ^b	Description of Annoyance
<21	Not at all annoyed
21 through 30	Slightly annoyed
31 through 39	Moderately annoyed
40 through 45	Very annoyed
>45	Extremely annoyed

^aA measure of the audibility of a noise source above the masking level of the background noise in any audible range frequency band.

^bThe intrusion level in this table is for a receptor with a neutral attitude toward the existence of the noise source, who is engaged in some type of ongoing activity.

Source: Fidell et al. 1988.

**TABLE E.50 Threshold
Limit Values for Con-
tinuous or Intermittent
Noise**

Duration (h/d)	Sound Level (dBA)
16	80
8	85
4	90
2	95
1	100
0.5	105
0.25	110
0.125	115 ^a

^aNo exposure to continuous or intermittent noise in excess of 115 dBA.

Source: ACGIH 1990.

**TABLE E.51 Threshold Limit Values for
Impulsive or Impact Noise**

Sound Level (dB) ^a	Permitted Numbers of Impulses per day (or impacts per day)
140	100
130	1,000
120	10,000

^aDecibels peak sound pressure level, referenced to 20 μ Pa.

Source: ACGIH 1990.

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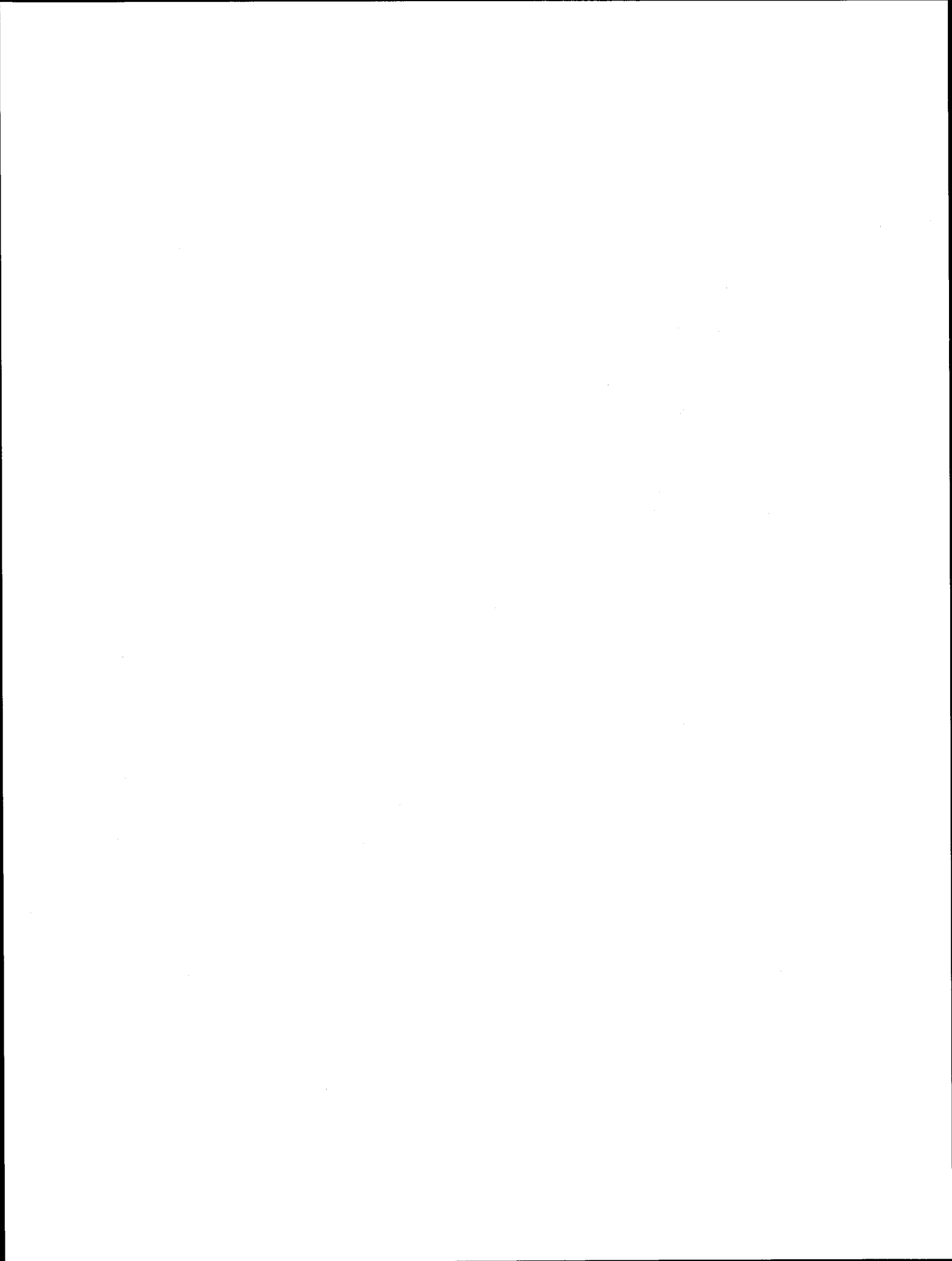
The following information was obtained from the records of the Department of the Interior, Bureau of Reclamation, and the Bureau of Land Management, and is being furnished for your information. The information is being furnished to you for your information and is not to be used for any other purpose.

APPENDIX F: WATER RESOURCES

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APPENDIX F: WATER RESOURCES

Appendix F provides detailed, site-specific information on such water resource topics as radioactive discharges; water quality of rivers, streams, and on-site ponds; water use; groundwater quality; and hydrogeological parameters.

Information on water resources is presented graphically in Figure F.1 for the Hanford Site, Figures F.2 and F.3 for INEL, and Figures F.4 through F.6 for SRS. Figure F.1 shows the locations of surface water intake and discharge points at the Hanford Site. For INEL, Figure F.2 shows the extent of the probably maximum flood combined with failure of MacKay Dam, and Figure F.3 shows the location of the Snake River Plain Aquifer. For SRS, Figure F.4 shows the extent of the Savannah River Drainage basin; Figure F.5 shows surface water quality sampling locations; and Figure F.6 shows the relationship of hydrologic and stratigraphic units at SRS.

Basic water resources data are presented in Tables F.1-F.9 for Hanford, Tables F.10-F.15 for INEL, and Tables F.16-F.24 for SRS. Information on water withdrawals and wastewater discharges for the three NPR technologies is presented in Tables F.25 and F.26 for Hanford, Tables F.27 through F.29 for INEL, and Tables F.30 through F.32 for SRS.

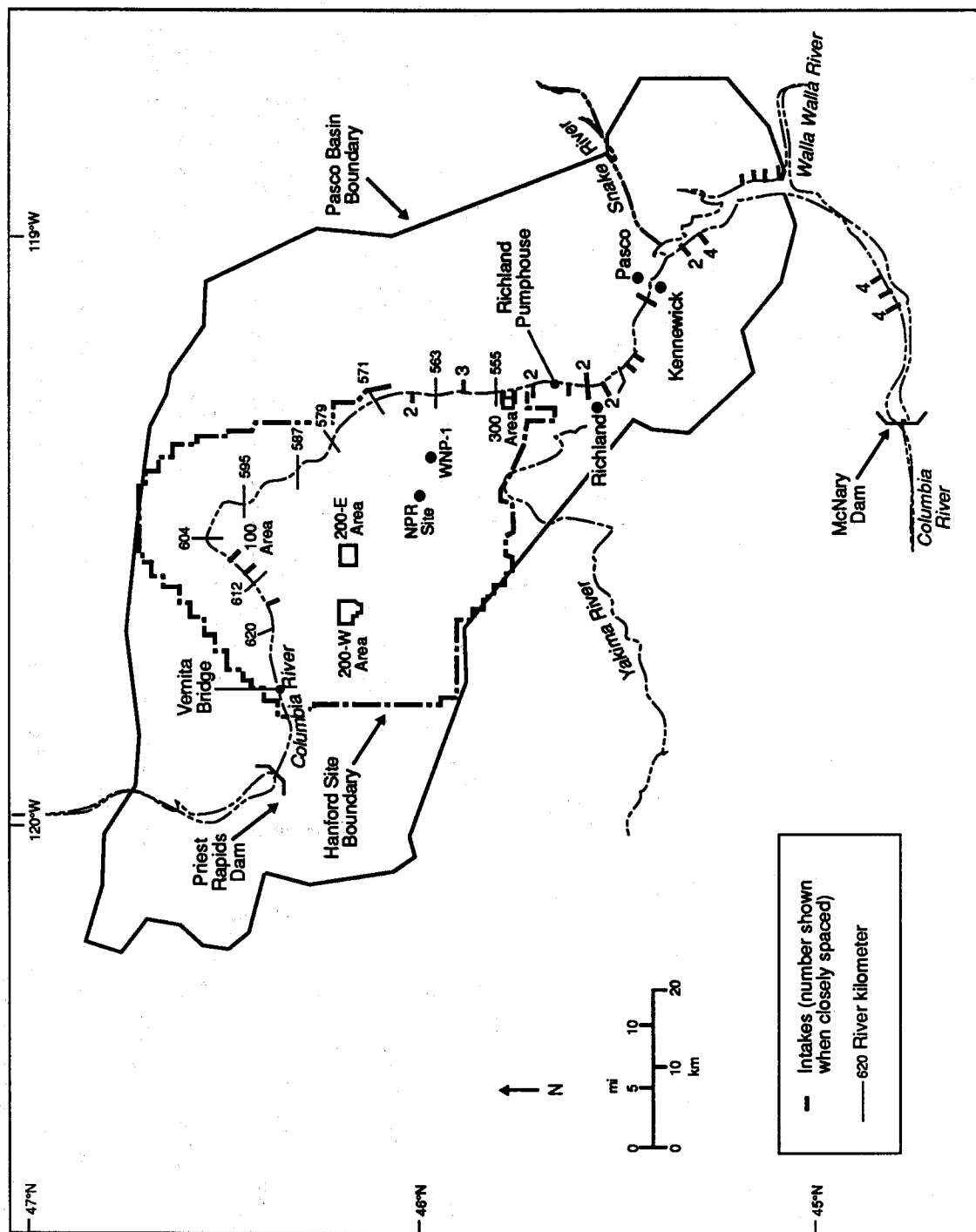


FIGURE F.1 Locations of Surface Water Intakes in the Hanford Site Vicinity (Source: Modified from Cushing 1990)

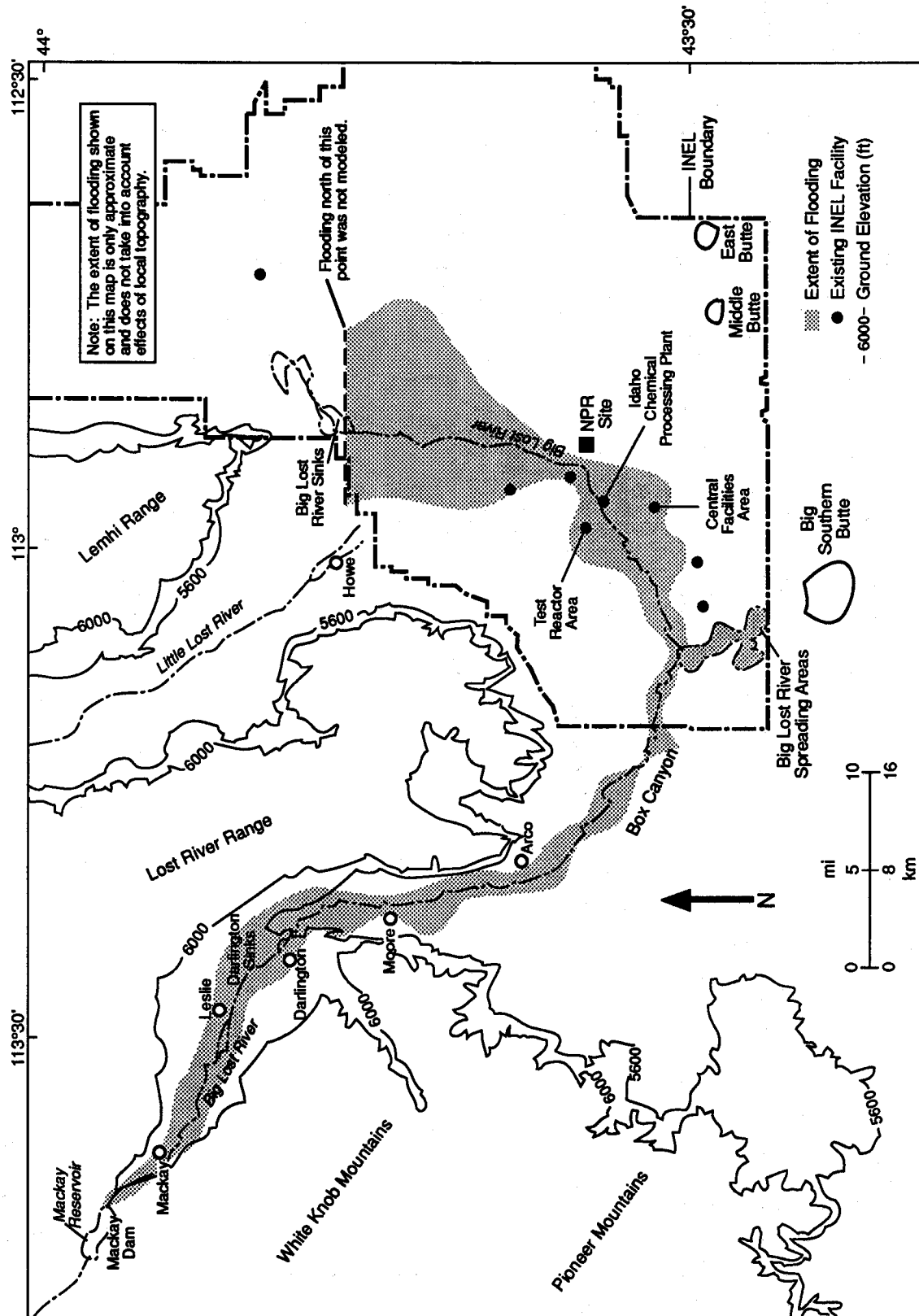


FIGURE F.2 Extent of Probable Maximum Flood Combined with Failure of MacKay Dam at INEL and Vicinity (Source: Modified from EG&G 1984)

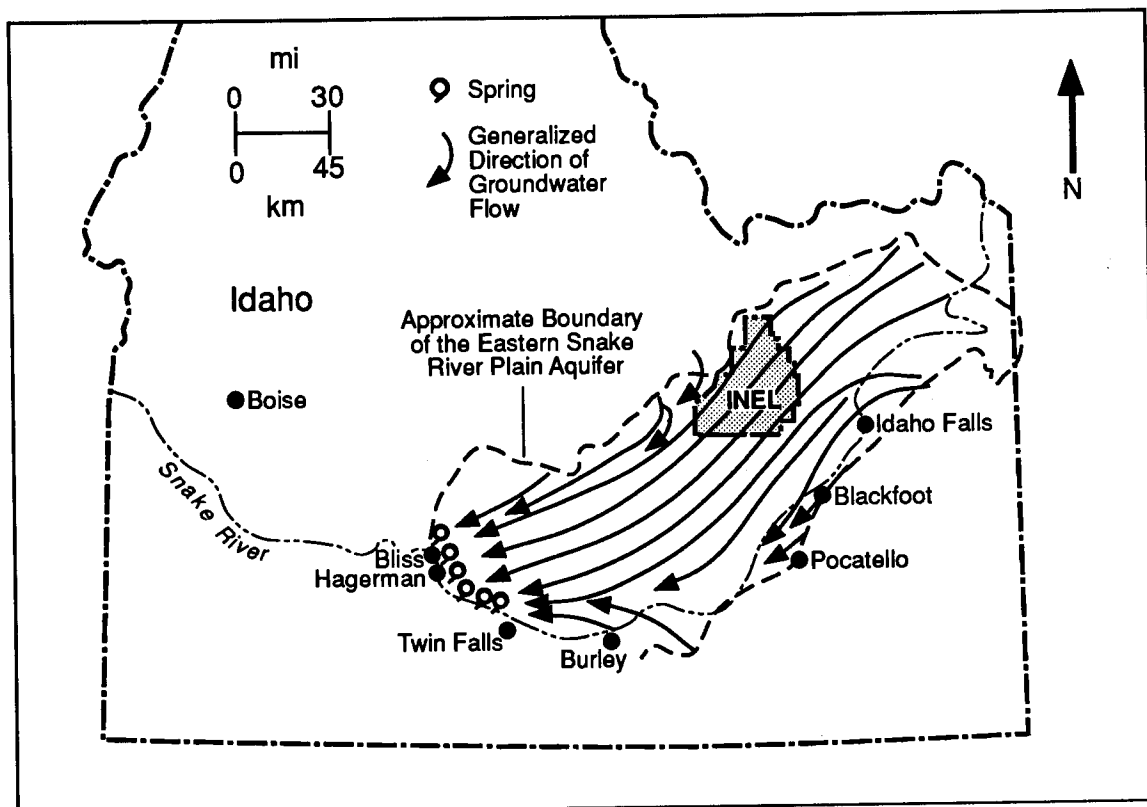


FIGURE F.3 Location of INEL Relative to the Snake River Plain Aquifer (Source: Modified from EG&G 1984)

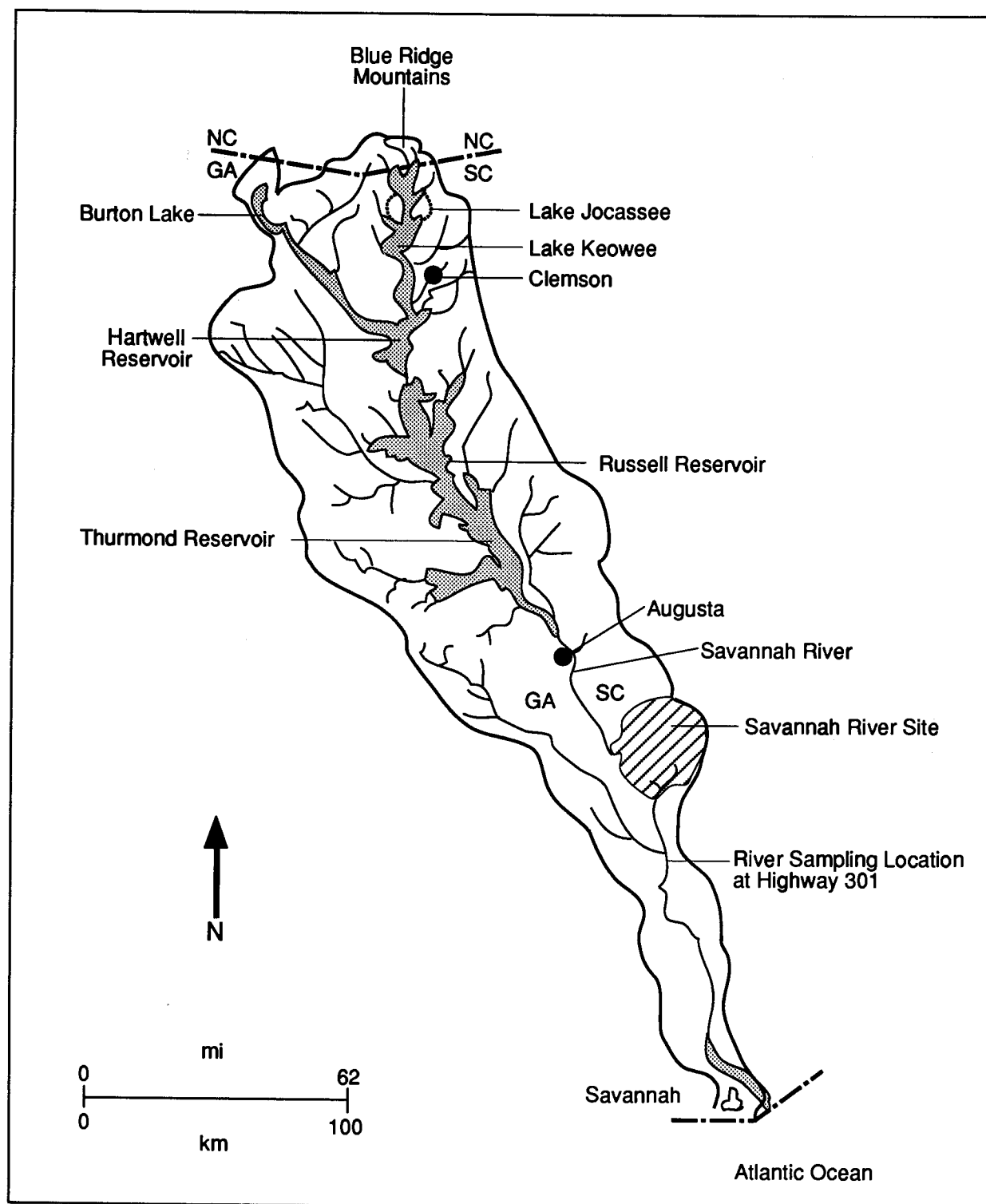


FIGURE F.4 Savannah River Drainage Basin and Reservoirs (Source: Modified from DOE 1990)

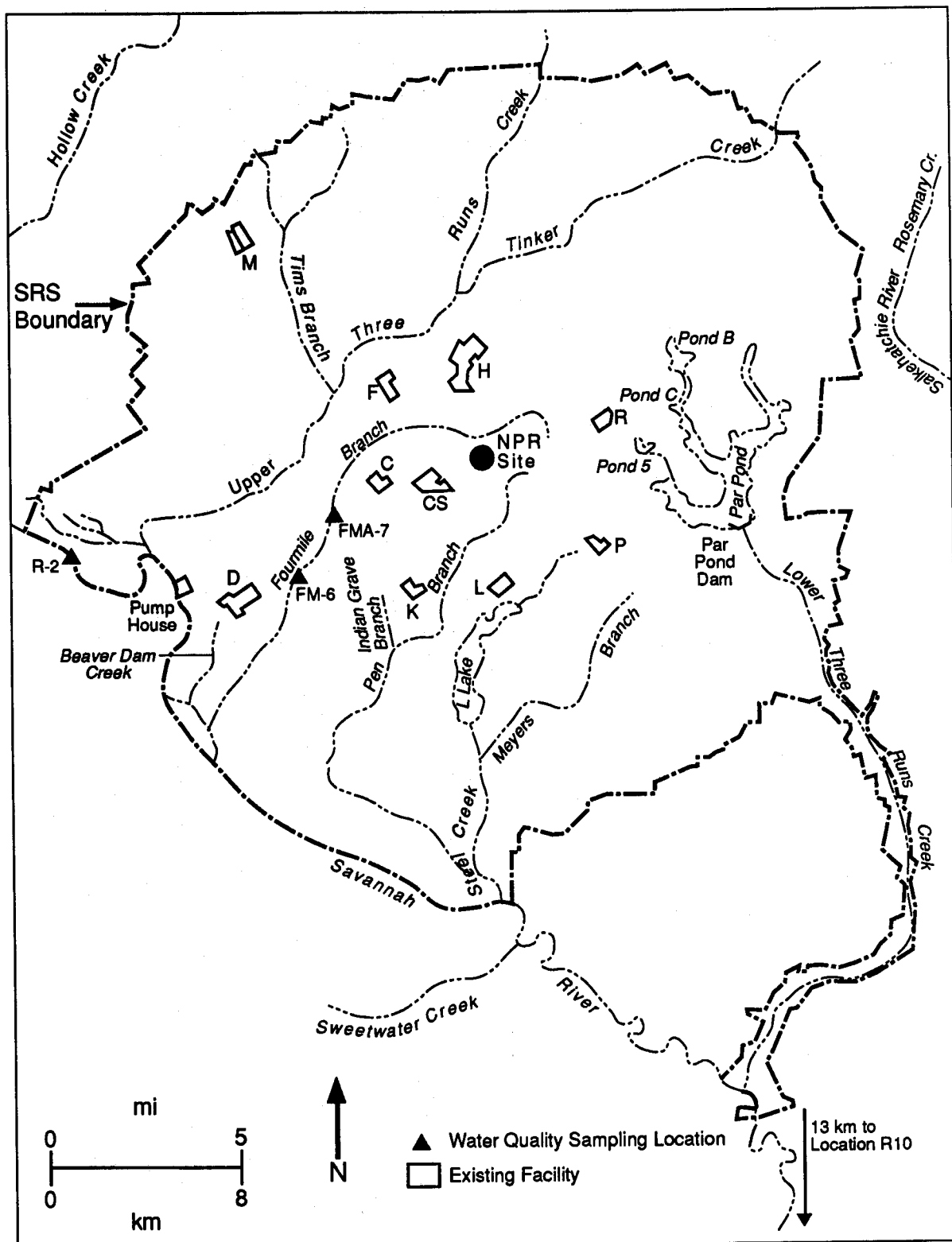


FIGURE F.5 Water Quality Sampling Locations for Surface Water for SRS (Source: Modified from WSRC 1989a)

Former Name	Term Used in the EIS	Stratigraphic Unit
Barnwell	Unsaturated Zone	Upland
Tan Clay	Water Table	Tobacco Road
McBean	Aquifer 4	Dry Branch
		Clinchfield
Green Clay		Santee
Congaree	Aquifer 3	Congaree
		Fishburne
		Williamsburg
Ellenton	Aquitard 2	Rhems
Upper Tuscaloosa Aquifer	Aquifer 2	Steel Creek Member, Peedee Formation
		Black Creek
Middle Tuscaloosa Confining Unit	Aquitard 1	
Lower Tuscaloosa Aquifer	Aquifer 1	Middendorf
Basal Confining Unit	Base Aquitard	Cape Fear

FIGURE F.6 Hydrologic and Stratigraphic Units at SRS (Source: Modified from WSRC 1989b; Looney et al. 1990)

TABLE F.1 Radioactive Constituents Reported for the Columbia River Water at Priest Rapids Dam, 1987

Radionuclide ^a	No. of Samples	Concentration (pCi/L) ^b		Drinking Water Standard ^d (pCi/L)
		Maximum	Minimum ^c	
Composite system				
Gross alpha	12	0.92 ± 0.46	0.19 ± 0.28	15
Gross beta	12	2.1 ± 1.4	0.19 ± 0.92	50
H-3 (tritium)	12	110 ± 10	50 ± 10	20,000
Sr-89	12	0.10 ± 0.08	bb	20
Sr-90	12	0.18 ± 0.04	0.10 ± 0.03	8
U-234	12	0.29 ± 0.05	0.17 ± 0.04	- ^e
U-235	12	0.028 ± 0.022	0.004 ± 0.006	-
U-238	12	0.37 ± 0.06	0.15 ± 0.04	-
U-total	12	0.57 ± 0.07	0.33 ± 0.05	-
Continuous system				
Co-60	24	0.0038 ± 0.009	bb	100
Particulate	24	0.0074 ± 0.008	bb	
Dissolved				
Nb-95	24	0.0043 ± 0.003	bb	300
Particulate	24	0.0071 ± 0.013	bb	
Dissolved				
Zr-95	24	0.0043 ± 0.0034	bb	200
Particulate	24	0.010 ± 0.021	bb	
Dissolved				
Ru-106	24	0.020 ± 0.065	bb	30
Particulate	24	0.034 ± 0.064	bb	
Dissolved				
I-129	4	0.000012 ± 0.000001	0.00004 ± 0.0000004	1
Dissolved				

TABLE F.1 (Cont'd)

Radionuclide ^a	No. of Samples	Concentration (pCi/L) ^b		Drinking Water Standard ^d (pCi/L)
		Maximum	Minimum ^c	
I-131				
Particulate	24	0.011 ± 0.007	bb	
Dissolved	24	0.039 ± 0.031	bb	3
Cs-134				
Particulate	24	0.0023 ± 0.0035	bb	
Dissolved	24	0.0052 ± 0.0074	bb	20,000
Cs-137				
Particulate	24	0.0026 ± 0.0018	bb	
Dissolved	24	0.0085 ± 0.010	bb	200
Ce-144				
Particulate	24	0.0081 ± 0.017	bb	
Dissolved	24	0.056 ± 0.071	bb	-
Pu-238				
Particulate	4	0.0000008 ± 0.0000020	bb	
Dissolved	4	0.00003 ± 0.00004	bb	-
Pu-239,240				
Particulate	4	0.000028 ± 0.000007	0.000004 ± 0.000002	
Dissolved	4	0.00014 ± 0.00007	0.00007 ± 0.00004	-

^aRadionuclides measured using the continuous system show the particulate and dissolved fractions separately. Other radionuclides are based on samples collected by the composite system.

^bMaximum and minimum values ±2 sigma counting error.

^cbb = below background concentration.

^dFrom state of Washington and EPA (PNL 1988).

^e(-) indicates no drinking water standard.

Source: PNL 1988.

TABLE F.2 Radioactive Constituents Reported for the Columbia River Water at the Richland Pumpouse, 1987

Radionuclide ^a	No. of Samples	Concentration (pCi/L) ^b		Drinking Water Standard ^d (pCi/L)
		Maximum	Minimum ^c	
Composite system				
Gross alpha	12	0.89 ± 0.43	0.05 ± 0.26	15
Gross beta	12	2.4 ± 1.4	0.21 ± 1.17	50
H-3 (tritium)	12	180 ± 10	70 ± 10	20,000
Sr-89	12	0.11 ± 0.07	bb	20
Sr-90	12	0.18 ± 0.04	0.10 ± 0.03	8
U-234	12	0.45 ± 0.06	0.14 ± 0.03	- ^e
U-235	12	0.037 ± 0.017	0.003 ± 0.011	-
U-238	12	0.36 ± 0.05	0.18 ± 0.04	-
U-total	12	0.84 ± 0.08	0.33 ± 0.05	-
Continuous system				
Co-60				100
Particulate	26	0.0051 ± 0.007	bb	
Dissolved	26	0.010 ± 0.013	bb	
Nb-95				300
Particulate	26	0.049 ± 0.005	bb	
Dissolved	26	0.011 ± 0.012	bb	
Zr-95				200
Particulate	26	0.0057 ± 0.010	bb	
Dissolved	26	0.0086 ± 0.017	bb	
Ru-106				30
Particulate	26	0.025 ± 0.068	bb	
Dissolved	26	0.063 ± 0.070	bb	
I-129				1
Dissolved	4	0.00013 ± 0.00001	0.000080 ± 0.000007	

TABLE F.2 (Cont'd)

Radionuclide ^a	No. of Samples	Concentration (pCi/L) ^b		Drinking Water Standard ^d (pCi/L)
		Maximum	Minimum ^c	
I-131				
Particulate	26	0.013 ± 0.0092	bb	3
Dissolved	26	0.030 ± 0.027	bb	
Cs-134				
Particulate	26	0.0034 ± 0.0039	bb	20,000
Dissolved	26	0.012 ± 0.0093	bb	
Cs-137				
Particulate	26	0.0038 ± 0.0072	bb	200
Dissolved	26	0.0085 ± 0.0064	bb	
Ce-144				
Particulate	26	0.0055 ± 0.015	bb	-
Dissolved	26	0.0055 ± 0.059	bb	
Pu-238				
Particulate	4	0.000004 ± 0.000004	0.0000007 ± 0.000005	-
Dissolved	4	0.00002 ± 0.00003	bb	
Pu-239,240				
Particulate	4	0.00006 ± 0.00001	0.000017 ± 0.000008	-
Dissolved	4	0.00010 ± 0.00005	0.00005 ± 0.00004	-

^aRadionuclides measured using the continuous system show the particulate and dissolved fractions separately. Other radionuclides are based on samples collected by the composite system.

^bMaximum and minimum values ± 2 sigma counting error.

^cbb = below background concentration.

^dFrom state of Washington and EPA (PNL 1988).

^e(-) indicates no drinking water standard.

Source: PNL 1988.

TABLE F.3 Nonradioactive Constituents in Columbia River Water, 1987

Analysis	Vernita Bridge (Upstream)			Richland Pumphouse (Downstream)				State Standard ^b	
	No. of Samples	Max.	Min.	Annual Average ^a	No. of Samples	Max.	Min.		Annual Average ^a
PNL Environmental Monitoring									
pH (pH units)	12	8.3	7.3	NA ^b	12	8.3	7.2	NA	6.5-8.5
Fecal coliform (#/100 mL)	12	64	2	5 ^c	12	240	2	22 ^c	100
Total coliform (#/100 mL)	12	2,400	2	110 ^c	12	240	2	49 ^c	NS ^b
Biological oxygen demand (mg/L)	12	8.3	0.4	2.48 ± 1.25	12	3.0	0.5	2.0 ± 0.5	NS
Nitrate (mg/L)	12	0.17	0.02	0.09 ± 0.03	12	0.77	0.05	0.2 ± 0.1	NS
USGS Sampling Program									
Temperature (°C) ^d	365	20.2	3.0	11.7	365	20.4	2.8	12.0	20 (maximum)
Dissolved oxygen (mg/L)	6	13.3	9.6	11.2 ± 1.4	4	13.6	9.5	11.3 ± 2.0	8 (minimum)
Turbidity (NTU) ^e	6	2.6	0.1	1.2 ± 0.8	4	10.0	0.7	3.8 ± 4.3	5 + background
pH (pH units)	6	8.4	7.9	NA	4	8.2	8.0	NA	6.5-8.5
Fecal coliform (#/100 mL)	6	7	<1	1.5 ^c	4	5	1	1.5 ^c	100
Suspended solids, 105°C (mg/L)	4	16	7	7.8 ± 6.2	4	11	<1	6.5 ± 5.8	NS
Dissolved solids, 180°C (mg/L)	6	92	70	77 ± 7	4	95	61	76 ± 14	NS
Specific conductance (µmho/cm)	6	161	127	138 ± 11	4	150	127	134 ± 11	NS
Hardness, as CaCO ₃ (mg/L)	6	76	59	67 ± 7	4	75	59	65 ± 7	NS
Phosphorus, total (mg/L)	6	0.03	0.01	0.02 ± 0.01	4	0.03	0.01	0.025 ± 0.01	NS
Chromium, dissolved (µg/L)	3	1	<1	<1	3	<10	<1	<7	NS
Nitrogen, Kjeldahl (mg/L)	6	0.7	<0.2	0.4 ± 0.1	4	0.8	<0.2	0.5 ± 0.25	NS
Total organic carbon (mg/L)	4	40	1.2	11.2 ± 19.2	4	97	1.4	35 ± 45	NS
Iron, dissolved (µg/L)	4	11	3	5.3 ± 3.9	4	14	4	8 ± 4.5	NS
Ammonia, dissolved (mg/L as N)	6	0.07	<0.01	0.03 ± 0.02	4	0.04	<0.01	0.02 ± 0.01	NS

^aAverage values ±2 standard error of the calculated mean.^bNA = not applicable; NS = no standard established.^cAnnual median.^dMaximum and minimum represent daily averages.^eNephelometric turbidity units.

Source: PNL 1988.

**TABLE F.4 Radioactive
Constituents in Discharges
to the Columbia River from
the Hanford Site, 1987**

Radionuclide	Release (Ci)
H-3	9.8×10^1
Na-24	ND ^a
P-32	1.5×10^{-3}
Cr-51	ND
Mn-54	2.5×10^{-3}
Fe-59	ND
Co-58	ND
Co-60	3.3×10^{-1}
Sr-89	8.3×10^{-1}
Sr-90	2.5
Zr-95	ND
Mo-99	ND
Ru-103	8.2×10^{-3}
Ru-106	3.8×10^{-2}
Sb-124	ND
Sb-125	4.2×10^{-2}
I-131	4.3×10^{-3}
I-133	ND
Xe-133	ND
Cs-137	8×10^{-2}
Ba-140	ND
Ce-141	ND
Ce-144	ND
Pu-238	4.4×10^{-6}
Pu-239, -240	5×10^{-4}

^aND = not detected.

Source: PNL 1988.

TABLE F.5 (Cont'd)

Sample Parameter	Value	NPDES Discharge Points ^a							N Springs
		003	004	005 ^b	005 ^c	006 ^d	006 ^e	007	009
Chromium (mg/L)	Max.	-	-	-	-	-	-	-	-
	Avg.	-	-	-	-	-	-	-	-
Chlorine (mg/L)	Max.	<0.04	-	<0.04	-	-	-	-	<0.04
	Avg.	<0.04	-	<0.04	-	-	-	-	<0.04

^a003 = 181-KE inlet screen backwash; 004 = 1908-K outfall; 005 = 182-N tank farm overflow (36" raw water release); 006 = 182-N drain system (42" raw water release); 007 = 181-N inlet screen backwash; 009 = 102" outfall (raw water release); N Springs = 100-N Riverbank Springs.

^bWithout low-lift pumps.

^cWith low-lift pumps.

^dWithout fog sprays.

^eWith fog sprays.

^fMeasurement not required.

^gTotal suspended solids.

Source: Cushing 1990.

TABLE F.6 Radionuclide Concentrations in On-site Ponds at Hanford, 1987

Location/ Radionuclide	No. of Samples	Concentration (pCi/L) ^a			Drinking Water Standard ^b (pCi/L)
		Maximum	Minimum	Average	
West Lake					
Gross alpha	4	267 ± 15	85 ± 8	186 ± 89	15
Gross beta	4	490 ± 87	95 ± 19	286 ± 195	50
H-3 (tritium)	4	570 ± 140	430 ± 140	500 ± 110	20,000
Sr-90	4	2.8 ± 0.3	1.8 ± 0.2	2.2 ± 0.5	8
Cs-137	4	1.6 ± 3.0	0.0 ± 1.0	0.9 ± 1.4	200
Gable Mountain Pond					
Gross alpha	3	0.8 ± 0.4	0.6 ± 0.4	0.7 ± 0.3	15
Gross beta	3	12.3 ± 2.5	4.9 ± 1.8	9.0 ± 5.2	50
H-3 (tritium)	3	230 ± 180	12 ± 220	160 ± 180	20,000
Sr-90	3	1.4 ± 0.1	0.5 ± 0.1	0.9 ± 0.6	8
Cs-137	3	50 ± 5	0.9 ± 0.9	21 ± 34	200
B Pond					
Gross alpha	4	0.9 ± 0.4	0.2 ± 0.3	0.4 ± 0.4	15
Gross beta	4	4.3 ± 1.7	2.3 ± 1.4	3.0 ± 1.2	50
H-3 (tritium)	4	160 ± 180	bb ^c	50 ± 140	20,000
Sr-90	4	0.4 ± 0.1	0.1 ± 0.1	0.3 ± 0.2	8
Cs-137	4	1.8 ± 1.9	0.3 ± 1.9	0.9 ± 1.2	200
FFTF Pond					
Gross alpha	3	1.3 ± 1.3	0.1 ± 0.3	0.5 ± 0.9	15
Gross beta	3	18 ± 3	13 ± 3	16 ± 4	50
H-3 (tritium)	3	9,510 ± 360	2,410 ± 220	5,560 ± 4,850	20,000
Cs-137	3	1.1 ± 2.0	bb	0.4 ± 1.4	200
Na-22	3	1.5 ± 1.5	0.5 ± 0.8	1.0 ± 0.9	- ^d

^aMaximum and minimum values ±2 sigma counting error.
Averages ±2 standard error of the calculated mean.

^bFrom State of Washington and EPA (PNL 1988).

^cbb = below background concentration.

^d(-) indicates no drinking water standard.

Source: PNL 1988.

TABLE F.7 Radionuclide Concentrations in On-site Ponds at Hanford, 1988^a

Location/ Radionuclide	No. of Samples	Concentration (pCi/L) ^a			Drinking Water Standard ^b (pCi/L)
		Maximum	Minimum	Average	
West Lake					
Gross alpha	3	259 ± 15	182 ± 13	226 ± 46	15
Gross beta	3	295 ± 47	234 ± 42	267 ± 36	50
H-3 (tritium)	3	650 ± 150	370 ± 170	480 ± 170	20,000
Sr-90	3	2.6 ± 0.2	2.1 ± 0.1	2.4 ± 0.3	8
Cs-137	3	2.3 ± 2.1	0.2± 1.6	0.8 ± 1.5	200
U-234	3	175 ± 4	106 ± 3	145 ± 41	- ^c
U-235	3	6.3 ± 0.8	4.0 ± 0.6	5.3 ± 1.3	-
U-238	3	165 ± 4	100 ± 3	137 ± 39	-
U-Total	3	346 ± 6	210 ± 4	287 ± 81	-
B Pond					
Gross alpha	4	0.7 ± 0.4	0.1 ± 0.3	0.4 ± 0.2	15
Gross beta	4	32 ± 4	0.9 ± 1.1	9.5 ± 14.9	50
H-3 (tritium)	4	169 ± 162	65 ± 162	78 ± 102	20,000
Sr-90	4	1.6 ± 0.1	0.4 ± 0.08	1.1 ± 0.6	8
Cs-137	4	1.4 ± 1.8	0.3 ± 2.1	0.7 ± 0.8	200
FFTF Pond					
Gross alpha	2	0.07 ± 0.25	bb ^d	-0.04 ± 0.23	15
Gross beta	2	21 ± 4	16 ± 3	19 ± 5	50
H-3 (tritium)	2	5,550 ± 280	5,200 ± 260	5,380 ± 350	20,000
Cs-137	2	1.0 ± 2.0	0.2 ± 0.6	0.4 ± 1.2	200
Na-22	2	0.2 ± 1.7	0.2 ± 2.0	0.01 ± 0.34	-

^aMaximum and minimum values ±2 sigma counting error.
Averages ±2 standard error of the calculated mean.

^bFrom State of Washington and EPA (PNL 1989).

^c(-) indicates no drinking water standard.

^dbb = below background concentration.

Source: PNL 1989.

TABLE F.8 Nonradioactive Constituents in West Lake at Hanford

Parameter	1975-76	1976	1987	1989
pH (pH units)	9.7-10.0	9.6	9.5-9.6	ND ^a
Alkalinity (mg/L)	9,009	ND	ND	ND
Hardness (mg/L)	122	ND	ND	ND
Specific conductance (µmho/cm)	229,700	14,550	230,000- 250,000	ND
Total suspended solids (mg/L)	ND	15,410	ND	ND
Calcium (mg/L)	ND	5	7.72	ND
Magnesium (mg/L)	ND	13	10.7	ND
Sodium (mg/L)	ND	2,000	4,120- 4,260	7,520
Potassium (mg/L)	ND	400	209-219	325
Carbon (mg/L)				
Carbonate	ND	5,860	ND	ND
Total carbon	ND	ND	1,881	ND
Total organic carbon	ND	600	113	3,800
Sulfate (mg/L)	ND	1,375	2,410- 3,060	ND
Sulfide (mg/L)	ND	ND	1.58	2,000
Chloride (mg/L)	ND	2,080	1,230- 1,640	ND
Fluoride (mg/L)	ND	158	7,699- 9,981	<5.0
Nitrate (mg/L)	ND	0.25	1.7-1.8	0.367
Ammonium (mg/L)	2.61	ND	ND	ND
Phosphorus (mg/L)	2,160	ND	6,600	9,790
Silicate (mg/L as Si)	0.23	ND	1	ND

^aND = no data.

Source: Wallace 1990.

**TABLE F.9 Surface Water Uses of Columbia River Water
at the Hanford Site and Vicinity**

User	Location of Intake		Annual Withdrawal (10 ⁶ m ³)	Type of Use ^a
	River Kilometer	Bank		
Hanford Site				
200 Area	620	Right	b	I
100-K Area	615	Right	2.54	I
100-N Area	610	Right	568	I
200 Area	607	Right	b	I
300 Area	554	Right	2.33	I
Other Locations				
Washington Public Power Supply System	565	Right	80.4	I
Peter Kiewit Sons Co.	565	Right	0.89	I
L.L. Bailey	558	Left	1.84	A
H.D. Loyd	558	Left	0.88	A
Central Premix Cement	558	Left	1.79	I
Battelle Memorial Institute	552	Right	3.93	A
Washington State University	550	Right	1.56	A
City of Richland ^c	550	Right	0.60	M
City of Richland ^c	546	Right	27.7	M
City of Richland ^c	546	Right	20.8	M
City of Richland ^c	546	Right	27.7	M
City of Richland ^c	546	Right	83.1	M
E.C. Watts	544	Right	0.28	A
H.S. Petty	544	Right	0.43	A
N.H. & M.E. Ketchersid	544	Left	1.48	A
G.C. Walkley	544	Left	2.07	A
R.T. Justesen	541	Right	2.27	A
Central Premix Concrete	541	Right	0.98	I
Benton County	534	Right	0.89	A
City of Richland ^c	538	Right	1.79	A
City of Kennewick ^c	528	Right	49.7	M
City of Pasco ^c	528	Left	31.3	M
F.J. Henokel	521	Right	0.001	A
Allied Chemical	521	Right	3.17	I
Chevron Chemical	520	Right	3.37	I
Chevron Chemical	520	Right	35.7	I
Phillips Pacific Chemical	520	Right	73.2	I
Phillips Pacific Chemical	520	Right	17.9	I
Boise Cascade Corp.	512	Left	21.9	I
L.D. Hoyte	510	Left	161	A
D. Howe	509	Left	5.72	A
Crawford & Sons	489	Right	29.3	A
Barborosa Farms	489	Right	17.9	A
Crawford & Sons	489	Right	6.79	A

TABLE F.9 (Cont'd)

User	Location of Intake		Annual Withdrawal (10^6 m ³)	Type of Use ^a
	River Kilometer	Bank		
Rainier National Bank	489	Right	8.40	A
Anderson & Coffin	486	Right	216	A
Horse Heaven Farms	484	Right	73.2	A
Horse Heaven Farms	484	Right	491	A
Horse Heaven Farms	484	Right	259	A
Anderson & Coffin	484	Right	216	A
Total			2,577 ^d	

^aA = agricultural use; M = municipal use; I = industrial use.

^bAnnual water withdrawal for the Hanford Site 200 Area operations withdrawn at River Kilometers 620 and 607 reported as a total of 22.4×10^6 m³.

^cMunicipal populations served by Richland, Kennewick, and Pasco water supply systems are approximately 34,250, 33,200, and 17,300 persons, respectively.

^dIncluding withdrawals at river kilometers 620 and 607.

Source: Cushing 1990.

TABLE F.10 Water Chemical Parameters for the Big Lost River at Upper and Lower Sample Areas^a in the Vicinity of INEL, Spring and Autumn 1975^b

Chemical Parameter	Spring		Autumn	
	Upper	Lower	Upper	Lower
Turbidity (Jackson turbidity units)	27.00	72.00	34.00	63.00
Total solids (mg/L)	287.00	480.00	254.00	258.00
Nitrate (mg/L as N)	0.45	0.58	1.33	0.55
Orthophosphate (mg/L)	0.10	0.32	0.02	0.01
Hardness (mg/L, as CaCO ₃)	144.00	160.00	158.00	150.00
Calcium (mg/L)	42.00	45.00	41.00	34.00
Sulphate (mg/L)	25.00	21.00	15.20	17.00
Fluoride (mg/L)	0.15	0.15	0.10	0.10
Specific conductivity (µmhos/cm)	280.00	305.00	330.00	310.00
Alkalinity (mg/L)	140.00	152.00	164.00	152.00
Iron (mg/L)	0.19	0.43	0.05	0.48
Magnesium (mg/L)	50.00	230.00	10.00	20.00
Sodium (mg/L)	5.80	6.60	4.20	5.10
Potassium (mg/L)	1.50	1.50	0.80	1.00
Chloride (mg/L)	4.00	4.00	2.00	2.00
Suspended solids (mg/L)	112.00	312.00	89.00	120.00
Total Kjeldahl nitrogen (mg/L)	1.70	1.80	1.40	1.20
Total inorganic phosphate (mg/L)	0.52	0.37	0.08	0.20
Total phosphorus (mg/L)	0.18	0.11	0.03	0.07

^aUpper sample area is located above Arco, Idaho; lower sample area is located on INEL.

^bAppropriate Idaho water quality standards (if established) are as follows:
10 mg/L for nitrate (as N), 1.4 to 2.4 mg/L for fluoride, and 20 mg/L for sodium (suggested optimum) (IDHW 1988).

Source: EG&G 1984.

TABLE F.11 Quality of Surface Waters North, Northeast,
and Northwest of INEL, 1963^{a,b}

Parameter	Big Lost River	Little Lost River	Birch Creek
Silica (mg/L)	12	12	8.8
Calcium (mg/L)	48	39	39
Magnesium (mg/L)	11	15	14
Sodium (mg/L)	6.9	6.7	5.0
Potassium (mg/L)	1.4	1.2	1.0
Bicarbonate (mg/L)	192	177	164
Carbonate (mg/L)	0	0	0
Sulfate (mg/L)	18	16	25
Chloride (mg/L)	3.5	8.8	4.5
Fluoride (mg/L)	1.9	0.2	0.2
Nitrate (mg/L as N)	0.5	0.6	0.6
Specific conductivity (μ mhos/cm at 25°C)	333	323	309
pH (pH units)	7.7	7.7	8.0
Temperature (°C)	--	12.2	14.4

^aBig Lost River samples collected August 27, 1963, near Moore, Idaho; Little Lost River samples collected September 3, 1963, near Howe, Idaho; Birch Creek samples collected September 3, 1963, south of Blue Dome.

^bAppropriate Idaho water quality standards (if established) are as follows: 20 mg/L for sodium (suggested optimum), 1.4 to 2.4 mg/L for fluoride (applies only to Big Lost River and Birch Creek), 10 mg/L for nitrate as N (applies only to Big Lost River and Birch Creek), 6.5-9.0 for pH, and 13°C for temperature (IDHW 1988).

Source: Pittman, Jensen, and Fischer 1988.

**TABLE F.12 Hydrologic Properties
of the Snake River Plain Aquifer in
the Vicinity of the INEL Site**

Property	Value
Transmissivity (m ² /d)	3.7 × 10 ² to 2.2 × 10 ⁵
Storativity	0.01 to 0.06
Effective porosity	10%

Source: Pittman, Jensen, and Fischer
1988.

TABLE F.13 Statistical Summary of Groundwater Quality in the Snake River Group Basalt

Constituent	Number of Analyses	Min.	Max.	Median	Mean	Standard Deviation	Idaho Drinking Water MCL ^a
Specific conductance (µmho/cm)	810	35	2,840	402	443	194	NS ^b
pH (pH units)	668	6.3	12	7.9	--	--	6.5-8.5
Temperature (°C)	765	1.3	41	14	13	2.9	26.7
Dissolved solids (mg/L)	567	28	2,290	251	282	146	500
Nitrogen, NO ₂ + NO ₃ (mg/L as N)	214	0.1	13	1.1	1.3	1.1	10
Sulfate (mg/L as SO ₄)	690	0.2	498	29	35	34	250
Hardness (mg/L as CaCO ₃)	683	17	499	170	177	62	NS
Silica (mg/L as SiO ₂)	590	0.8	64	29	28	8.8	NS
Coliform, fecal (colony/100 mL)	100	<1 ^c	13	<1 ^c	--	--	NS
Coliform, total (colony/100 mL)	58	<1 ^c	180	<1 ^c	--	--	1
Iron, dissolved (µg/L as Fe)	284	0	1,400	--	57	124	300

^aMCL = maximum contaminant level (IDHW 1985).^bNS = no standard established.^cActual value is known to be less than the value shown.

Source: Yee and Souza 1987.

TABLE F.14 Comparison of Maximum Concentrations of Organic Chemical Contaminants in the Groundwater at INEL with EPA Maximum Contaminant Level for Drinking Water (all concentrations in µg/L)

Source	Date	Carbon-tetrachloride	Chloroform	1,1,1-Trichloroethane	Trichloroethylene	Tetrachloroethylene	1,1-dichloroethylene	Toluene
EPA Drinking Water Maximum Contaminant Level ^a		5	NS ^b	200	5	NS	7	NS
Radioactive Waste Management Complex Vicinity								
Well 88	6/24/88	4.1	ND ^c	ND	1.2	ND	ND	ND
Well 90	6/22/88	1.0	ND	ND	ND	ND	ND	ND
Well 92	4/6/88	1,400	940	250	1,100	120	2.6	ND
Well 120	6/27/88	2.1	ND	ND	ND	ND	ND	ND
Radioactive Waste Management Complex	12/28/88	1.5	ND	ND	ND	ND	ND	ND
Test Area North								
Well 24	1/14/88	ND	1.3	24	1,600	120	11	--
Rest Area	5/18/88	ND	ND	ND	ND	ND	ND	5.2

^aCode of Federal Regulation, 1989, Title 40, Part 141, National Primary Drinking Water Regulations, July 1.

^bNS = no limit specified.

^cND = no data.

Source: Hoff, Mitchell, and Moore 1989.

TABLE F.15 Maximum Concentrations of
Radionuclides in the Groundwater at INEL

Radionuclide	Maximum Concentration (pCi/L)	Well Number	Date
H-3 (tritium) ^a	80,600	65	10/14/87
Sr-90 ^b	54	47	10/26/87
Pu-239/240	0.05	43	10/5/87
Am-241	0.07	ARA-3	10/28/87
Cs-137	70	TAN-2	10/23/87

^aIdaho drinking water maximum contaminant level is 20,000 pCi/L for tritium and 8 pCi/L for Sr-90 (IDWH 1985).

^bIdaho drinking water maximum contaminant level is 8 pCi/L.

Source: Knobel and Mann 1988.

TABLE F.16 Water Quality of Savannah River Upstream of SRS, 1988^a

Parameter	Unit	No. of Analyses	Concentration				Water Quality Standard
			Maxi- mum	Mini- mum	Mean	±2 Standard Deviations	
Temperature	°C	12	28	9.0	18	±12	6.5-8.5 ^b
pH	pH	12	7.6	6.1			
Dissolved oxygen	mg/L	12	11	5.4	8.5	±4.0	
Alkalinity	mg/L	12	27	21	24	±4.3	
Hardness	mg/L	4	20	15	- ^c	-	1 ^d
Conductivity	µmho/cm	12	125	7.0	68	±99	
Turbidity	NTU	12	9.0	1.0	3.0	±4.6	
Suspended solids	mg/L	12	16	3.0	7.8	±7.8	
Volatile solids	mg/L	12	6.0	1.0	2.9	±2.6	500 ^b
Total dissolved solids	mg/L	12	94	23	73	±35	
Total solids	mg/L	12	94	33	80	±35	
Fixed residue	mg/L	12	10	2.0	5.2	±5.4	
Chemical oxygen demand	mg/L	12	20	4.0	12	±8.9	250 ^b 10 ^d 250 ^d
Chloride	mg/L	12	11	6.5	9.1	±2.8	
NO ₂ /NO ₃ (as N)	mg/L	12	0.82	0.21	0.31	±0.33	
Sulfate	mg/L	12	10	6.0	7.4	±2.2	
Phosphorus (as PO ₄)	mg/L	12	0.17	0.07	0.11	±0.07	1 ^b 0.01 ^d 0.05 ^b
Aluminum	mg/L	4	0.17	<0.05	-	-	
Nitrogen (as NH ₃)	mg/L	12	0.21	0.05	0.14	±0.09	
Calcium	mg/L	4	5.2	3.8	-	-	
Copper	mg/L	4	<0.01	<0.01	-	-	0.01 ^d 0.05 ^b
Cadmium	mg/L	4	<0.01	<0.01	-	-	
Magnesium	mg/L	4	1.7	1.3	-	-	
Manganese	mg/L	4	0.07	<0.01	-	-	

TABLE F. 16 (Cont'd)

Parameter	Unit	No. of Analyses	Concentration				Water Quality Standard
			Maxi- mum	Mini- mum	Mean	±2 Standard Deviations	
Mercury	µg/L	4	0.70	<0.20	-	-	2.0 ^d
Nickel	mg/L	4	<0.05	<0.05	-	-	-
Sodium	mg/L	4	17	11	-	-	-
Iron	mg/L	4	0.32	0.03	-	-	0.3 ^b
Lead	mg/L	4	<0.01	<0.01	-	-	-
Chromium	mg/L	4	<0.02	<0.02	-	-	0.05 ^d
Zinc	mg/L	4	<0.01	<0.01	-	-	5 ^b
Gross alpha	pCi/L	52	0.35	0	0.03	±0.22	15 ^d
Beta	pCi/L	52	3.3	0.19	1.8	±1.3	-
Tritium	pCi/L	52	810	0	270	±400	20,000 ^d
Strontium-89, -90	pCi/L	12	1.7	0	0.32	±0.96	-
Cobalt-60	pCi/L	52	23 ^c	0	0.45	±6.3	-
Cesium-137	pCi/L	52	0.068	0.001	0.014	±0.032	-
Iodine-131	pCi/L	52	0	0	0	-	-
Plutonium-238	pCi/L	4	0.00099	0	0.00021	-	-
Plutonium-239	pCi/L	4	0.013	0	0.0036	-	-

^aSamples taken at R-2 (Figure F.5).^bNational secondary drinking water regulations (40 CFR 143).^c(-) indicates insufficient data; mean not calculated.^dNational primary drinking water regulations (40 CFR 141).^eThis analysis result is considered anomalous.

Source: Davis, Martin, and Todd 1989.

TABLE F.17 Water Quality of Savannah River Downstream of SRS, 1988^a

Parameter	Unit	No. of Analyses	Concentration			Water Quality Standard
			Maximum	Minimum	±2 Standard Deviations	
Temperature	°C	12	26	9.0	18	±12
pH	pH	12	7.4	5.1		6.5-8.5 ^b
Dissolved oxygen	mg/L	12	11	5.0	7.8	±3.6
Alkalinity	mg/L	12	29	21	24	±4.9
Hardness	mg/L	4	21	16	- ^c	-
Conductivity	µmho/cm	12	122	2.0	62	±103
Turbidity	NTU	12	6.0	1.0	2.5	±2.9
Suspended solids	mg/L	12	11	1.0	7.5	±7.1
Volatile solids	mg/L	12	3.0	1.0	2.2	±1.9
Total dissolved solids	mg/L	12	77	51	67	±14
Total solids	mg/L	12	88	62	74	±17
Fixed residue	mg/L	12	9.0	<1.0	5.7	±5.6
Chemical oxygen demand	mg/L	12	18	3.0	8.6	±8.9
Chloride	mg/L	12	11	6.0	8.8	±2.9
NO ₂ /NO ₃ (as N)	mg/L	12	0.69	0.16	0.37	±0.33
Sulfate	mg/L	12	11	6.0	8.1	±3.2
Phosphorus (as PO ₄)	mg/L	12	0.13	0.05	0.09	±0.05
Aluminum	mg/L	4	0.20	0.07	-	-
Nitrogen (as NH ₃)	mg/L	12	0.13	0.02	0.07	±0.06
Calcium	mg/L	4	5.8	4.4	-	-
Copper	mg/L	4	<0.01	<0.01	-	-
Cadmium	mg/L	4	<0.01	<0.01	-	-
Magnesium	mg/L	4	1.6	1.3	-	-
Manganese	mg/L	4	0.07	<0.01	-	-
Mercury	µg/L	4	0.26	<0.20	-	-

^{1b}
0.01^d
0.05^b
2.0^d

TABLE F. 17 (Cont'd)

Parameter	Unit	No. of Analyses	Concentration				Water Quality Standard
			Maxi- mum	Mini- mum	Mean	±2 Standard Deviations	
Nickel	mg/L	4	<0.05	<0.05	-	-	-
Sodium	mg/L	4	17	11	-	-	-
Iron	mg/L	4	0.46	0.18	-	-	0.3 ^b
Lead	mg/L	4	0.06	<0.01	-	-	-
Chromium	mg/L	4	<0.02	<0.02	-	-	0.05 ^d
Zinc	mg/L	4	<0.01	<0.01	-	-	5 ^b
Gross alpha	pCi/L	52	0.49	0	0.04	±0.26	15 ^d
Beta	pCi/L	52	3.0	0.16	1.5	±1.1	-
Tritium	pCi/L	52	5,500	1,800	3,400	±2,200	20,000 ^d
Strontium-89, -90	pCi/L	12	4.1	0	0.72	±3.1	-
Cobalt-60	pCi/L	52	0	0	0	-	-
Cesium-137	pCi/L	52	0.12	0.035	0.065	±0.051	-
Iodine-131	pCi/L	52	0	0	0	-	-
Plutonium-238	pCi/L	4	0.00016	0	0.00001	-	-
Plutonium-239	pCi/L	4	0.016	0.00012	0.0042	-	-

^aSamples taken at R-10 at Highway 301 (Figures F.4 and F.5).^bNational secondary drinking water regulations (40 CFR 143).^c(-) indicates insufficient data; mean not calculated.^dNational primary drinking water regulations (40 CFR 141).

Source: Davis, Martin, and Todd 1989.

TABLE F.18 Water Quality of Fourmile Branch, 1988^a

Parameter	Unit	No. of Analyses	Concentration			Water Quality Standard
			Maxi- mum	Mini- mum	±2 Standard Deviations	
Temperature	°C	12	31	9.0	±13	6.5-8.5 ^b
pH	pH	12	7.7	3.1		
Dissolved oxygen	mg/L	12	10	6.5	±2.0	
Alkalinity	mg/L	12	17	2.0	±8.1	
Hardness	mg/L	4	12	10	- ^c	
Conductivity	µmho/cm	12	84	15	±53	
Turbidity	NTU	12	18	<1	±9.7	1 ^d
Suspended solids	mg/L	12	7.0	0.0	±4.5	
Volatile solids	mg/L	12	5.0	0.0	±3.0	
Total dissolved solids	mg/L	12	55	34	±15	500 ^b
Total solids	mg/L	12	62	34	±17	
Fixed residue	mg/L	12	6.0	0.0	±3.3	
Chemical oxygen demand	mg/L	12	16	1.0	±8.6	
Chloride	mg/L	12	3.7	2.1	±1.0	250 ^b
NO ₂ /NO ₃ (as N)	mg/L	12	2.7	0.18	±0.72	10 ^d
Sulfate	mg/L	12	8.0	4.0	±7.9	250 ^d
Phosphorus (as PO ₄)	mg/L	12	0.03	<0.02	±0.01	
Aluminum	mg/L	4	<0.05	<0.05	-	
Nitrogen (as NH ₃)	mg/L	12	0.06	0.0	±0.04	
Calcium	mg/L	4	3.6	3.1	-	1 ^b
Copper	mg/L	4	0.02	<0.01	-	0.01 ^d
Cadmium	mg/L	4	<0.01	<0.01	-	
Magnesium	mg/L	4	0.66	0.58	-	0.05 ^b
Manganese	mg/L	4	0.01	<0.01	-	2.0 ^d
Mercury	µg/L	4	<0.20	<0.20	-	

TABLE F. 18 (Cont'd)

Parameter	Unit	No. of Analyses	Concentration				Water Quality Standard
			Maxi-mum	Mini-mum	Mean	± 2 Standard Deviations	
Nickel	mg/L	4	<0.05	<0.05	-	-	-
Sodium	mg/L	4	10	7.5	-	-	-
Iron	mg/L	4	0.11	0.05	-	-	0.3 ^b
Lead	mg/L	4	<0.01	<0.01	-	-	-
Chromium	mg/L	4	<0.02	<0.02	-	-	0.050 ^d
Zinc	mg/L	4	0.02	<0.01	-	-	5 ^b
Gross alpha	pCi/L	26	1.3	0	0.27	±0.70	15 ^d
Beta	pCi/L	26	59	26	34	±14	-
Tritium	pCi/L	26	740,000	520,000	630,000	±150	20,000 ^d
Strontium-89, -90	pCi/L	12	15	2.9	10	±6.8	-
Uranium/plutonium	pCi/L	26	0.66	0	0.09	±0.44	-
Cobalt-60	pCi/L	12	0	0	0	-	-
Iodine-131	pCi/L	12	0	0	0	-	-
Cesium-137	pCi/L	12	3.7	0	0.31	±2.1	-

^aSamples taken at FM-6 (Figure F.5).^bNational secondary drinking water regulations (40 CFR 143).^c(-) indicates insufficient data; mean not calculated for <5 samples.^dNational primary drinking water regulations (40 CFR 141).

Source: Davis, Martin, and Todd 1989.

TABLE F.19 Waste Stream Data for C Reactor near the NPR Site at SRS

Waste Outfall	Effluent Contributors			Pollutants	
	Receiving Water	Operation	Average Flow (m ³ /d)	Parameter	Concen- tration
C-001	Fourmile Branch	Stormwater runoff Offices Reactor building cooling water	Varies 150 110	Biological oxygen demand (mg/L)	2.0
				Total organic carbon (mg/L)	1.01
				Total suspended solids (mg/L)	22.5
				Temperature (winter) (°C)	28.0
				Temperature (summer) (°C)	29.0
				pH	6.2-8.1
				Bromide (mg/L)	2.0
				Fecal coliform (#/100 mL)	240
				Oil and grease (mg/L)	5.6
				Phosphorus (mg/L)	0.05
				Sulfate (mg/L)	7.0
				Sulfite (mg/L)	4.0
				Iron (mg/L)	0.11
				Magnesium (mg/L)	0.63
				Phenols (µg/L)	54.0
C-003	Unnamed tributary of Fourmile Branch	Stormwater runoff Water plant waste Reactor building cooling water	Varies 380 380	Biological oxygen demand (mg/L)	2.0
				Total organic carbon (mg/L)	0.354
				Total suspended solids (mg/L)	37.2
				Temperature (winter) (°C)	26.0
				Temperature (summer) (°C)	28.0
				pH	6.45-8.15
				Chlorine (mg/L)	1.5
				Oil and grease (mg/L)	13.2
				Phosphorus (mg/L)	0.06
				Sulfate (mg/L)	8.0
				Sulfite (mg/L)	3.0
				Iron (mg/L)	0.42
				Magnesium (mg/L)	0.33
				Antimony	0.053

TABLE F.19 (Cont'd)

Waste Outfall	Effluent Contributors		Pollutants	Concen- tration	
	Receiving Water	Operation			Average Flow (m³/d)
C-004	Fourmile Branch	Overflow from 186 basin Sanitary plant Reactor building and nonprocess cooling water	570	Biological oxygen demand (mg/L)	2.0
				Total organic carbon (mg/L)	5.50
				Total suspended solids (mg/L)	16.0
				Temperature (winter) (°C)	68.0 ^a
				Temperature (summer) (°C)	71.0 ^a
				pH	6.1-8.4
				Bromide (mg/L)	3.0
				Fecal coliform (#/100 mL)	66
				Oil and grease (mg/L)	12.8
				Sulfate (mg/L)	6.0
				Sulfite (mg/L)	3.0
				Barium (mg/L)	0.27
				Magnesium (mg/L)	1.04
Tin	0.17				
C-004A	Fourmile Branch	Sanitary wastewater treatment plant	38	Biological oxygen demand (mg/L)	27.5
				Total organic carbon (mg/L)	12.4
				Total suspended solids (mg/L)	47.2
				Ammonia (mg/L)	1.0
				Temperature (summer) (°C)	17.0 ^a
				Bromide (mg/L)	7.0
				Chlorine (mg/L)	2.0
				Fecal coliform (#/100 mL)	100
				Fluoride (mg/L)	3.1
				Nitrate (mg/L)	30.0
				Nitrogen (mg/L)	2.1
				Oil and grease (mg/L)	2.0
				Phosphorus (mg/L)	5.5
				Sulfate (mg/L)	10.0
				Sulfite (mg/L)	15.0
				Aluminum (mg/L)	0.17
Iron (mg/L)	0.12				

TABLE F.19 (Cont'd)

Waste Outfall	Effluent Contributors		Pollutants	
	Receiving Water	Operation	Parameter	Concen- tration
C-004A (Cont'd)	Fourmile Branch	Sanitary wastewater treatment plant	Average Flow (m ³ /d)	38
			Magnesium (mg/L)	0.95
			Copper (mg/L)	0.05
			Lead (mg/L)	0.005
			Mercury (mg/L)	0.0006
			Bromoform (mg/L)	239.0
			Chlorodibromomethane (mg/L)	6.0

^aTemperatures are representative during reactor operations; however, there are no plans to operate C Reactor in the future.

Source: DOE 1988.

TABLE F.20 Waste Stream Data for H Area near NPR Site at SRS

Waste Outfall	Receiving Water	Effluent Contributors		Pollutants		Concen- tration
		Operation	Average Flow (m ³ /d)	Parameter		
H-008	Fourmile Branch	Stormwater runoff	Varies	Biological oxygen demand (mg/L)		22.0
		Ash sluice water	300	Total organic carbon (mg/L)		0.703
		Service tank overflow	110	Total suspended solids (mg/L)		40.0
		Domestic tank overflow	760	Temperature (winter) (°C)		28.6
		Waste management	1,500	Temperature (summer) (°C)		34.0
				pH		4.8-8.7
				Bromide (mg/L)		1.0
				Chlorine (mg/L)		1.6
				Oil and grease (mg/L)		8.8
				Sulfate (mg/L)		15.0
				Sulfite (mg/L)		12.0
				Aluminum (mg/L)		0.25
				Barium (mg/L)		0.16
				Iron (mg/L)		0.59
				Magnesium (mg/L)		0.39
H-008	Fourmile Branch			Manganese (mg/L)		0.109
				Lead (mg/L)		0.006
		Stormwater runoff	Varies	Total organic carbon (mg/L)		0.625
		Ash basin overflow	300	Total suspended solids (mg/L)		18.5
				Temperature (winter) (°C)		14.0
				pH		5.4-8.9
				Nitrate-nitrite		0.16
				Oil and grease (mg/L)		8.8
				Sulfate (mg/L)		27.0
				Sulfite (mg/L)		3.0
				Aluminum (mg/L)		0.12
				Barium (mg/L)		1.58
				Boron (mg/L)		0.11
				Iron (mg/L)		0.25

TABLE F.20 (Cont'd)

Waste Outfall	Receiving Water	Effluent Contributors		Pollutants				
		Operation	Average Flow (m ³ /d)	Parameter	Concen- tration			
H-008A (Cont'd)	Fourmile Branch	Ash basin overflow	300	Magnesium (mg/L)	1.33			
				Manganese (mg/L)	0.062			
				Zinc (mg/L)	0.07			
				Phenols (µg/L)	20.0			
H-012	Fourmile Branch	Stormwater runoff Sanitary treatment plant Canyon segregated cooling water Tritium facilities nonprocess cooling water Offices 706-H Waste management cooling tower and air compressor blowdown	Varies	Total organic carbon (mg/L)	1.24			
				Total suspended solids (mg/L)	39.8			
			1,900	Temperature (winter) (°C)	34.6			
				Temperature (summer) (°C)	38.0*			
			760	pH	6.24-8.20			
				Chlorine (mg/L)	0.6			
			190	Fecal coliform (#/100 mL)	4			
				Nitrate-nitrite (mg/L)	11.0			
			190	Oil and grease (mg/L)	11.84			
				Phosphorus	0.28			
			H-012A	Fourmile Branch	Sanitary wastewater treatment plant	150	Sulfate	29.5
							Sulfite	3.0
Aluminum	0.12							
Iron	0.26							
Magnesium (mg/L)	0.40							
Biochemical oxygen demand (mg/L)	57.7							
				Chemical oxygen demand (mg/L)	30.0			
				Total organic carbon (mg/L)	13.8			
				Total suspended solids (mg/L)	44.5			
				Ammonia (mg/L)	3.2			
				Temperature (summer) (°C)	21.0			
				pH	6.35-8.89			
				Chlorine (mg/L)	2.0			
				Fecal coliform (#/100 mL)	400			
				Fluoride (mg/L)	4.8			
				Nitrate-nitrite (mg/L)	26.0			

TABLE F.20 (Cont'd)

Waste Outfall	Receiving Water	Effluent Contributors		Pollutants	
		Operation	Average Flow (m ³ /d)	Parameter	Concen- tration
H-012A (Cont'd)	Fourmile Branch	Sanitary wastewater treatment plant	150	Nitrogen (mg/L)	8.4
				Phosphorus (mg/L)	3.9
				Sulfate (mg/L)	29.0
				Sulfite (mg/L)	8.0
				Aluminum (mg/L)	0.16
				Iron (mg/L)	0.14
				Magnesium (mg/L)	1.49
				Copper (mg/L)	0.05
				Lead (mg/L)	0.008
				Zinc (mg/L)	0.09
				Bromoform (mg/L)	89.0
				Chlorodibromomethane (mg/L)	18.0
				Biochemical oxygen demand (mg/L)	2.0
				Chemical oxygen demand (mg/L)	5.0
H-017	Tributary to Fourmile Branch	Stormwater runoff	Varies	Total organic carbon (mg/L)	2.0
				Total suspended solids (mg/L)	90.0
				Ammonia (mg/L)	1.0
				Temperature (winter) (°C)	15.0
				Temperature (summer) (°C)	30.0
				pH	5.0-7.0
				Magnesium (mg/L)	0.7
				Manganese	0.08
				Iron	3.8
				Sulfate	7.0
				Fluoride	1.0
				Oil and grease	15.0
				Biochemical oxygen demand (mg/L)	2.0
				Chemical oxygen demand (mg/L)	5.0
H-018	Tributary to Fourmile Branch	Cooling water retention basin	200	Total organic carbon (mg/L)	2.0
				Total suspended solids (mg/L)	30.0

TABLE F.20 (Cont'd)

Waste Outfall	Receiving Water	Effluent Contributors		Pollutants	
		Operation	Average Flow (m ³ /d)	Parameter	Concen- tration
H-018 (Cont'd)	Tributary to Fourmile Branch	Cooling water retention basin	200	Ammonia (mg/L)	1.0
				Temperature (winter (°C)	30.0
				Temperature (summer) (°C)	37.0
				pH	5.0-7.0
				Magnesium (mg/L)	0.4
				Manganese (mg/L)	0.01
				Iron (mg/L)	10.0
				Sulfate (mg/L)	14.0
				Fluoride (mg/L)	1.0
				Chlorine (mg/L)	0.5

^aTemperatures are representative during reactor operations.

Source: DOE 1988.

TABLE F.21 Hydraulic Characteristics of Hydrologic Units at SRS

Unit	Horizontal Hydraulic Conductivity (m/s)	Vertical Hydraulic Conductivity (m/s)	Effective Porosity	Storage Coefficient
Aquifer 4	1.4×10^{-5} to 3.5×10^{-5}	2×10^{-8} to 9×10^{-8}	0.25	0.25
Aquifer 3	2×10^{-4} to 3.9×10^{-4}	NA	0.25	0.25
Green Clay	NA	4×10^{-11}	NA	NA
Aquifers 1 and 2	4.7×10^{-4}	NA	0.20	4.5×10^{-4}

^aNA = data not available.

Sources: Scott et al. 1987; Root 1983; du Pont 1983.

TABLE F.22 SRS Groundwater Data, Radiological Constituents

Parameter	Number of Samples	Concentration (pCi/L)	
		Maximum	Minimum
Alpha ^a	1,539	360	<DL ^b
Nonvolatile beta	1,539	24,000	<DL
Tritium ^c	1,379	7×10^9	<DL
Cerium-144	42	180	<DL
Cesium-134	42	70	<DL
Cesium-137	42	20	<DL
Chromium-51	42	2,400	<DL
Cobalt-60	42	250	<DL
Ruthenium-103	42	100	<DL
Ruthenium-106	42	220	<DL
Antimony-125	42	10	<DL
Strontium-89, -90	31	140,000	<DL

^aThe national primary drinking water standard for gross alpha is 15 pCi/L (40 CFR 141).

^bDetection limits.

^cThe national primary drinking water standard for tritium is 20,000 pCi/L (40 CFR 141).

Source: DOE 1987b; Zeigler, Lawrimore, and Heath 1986.

TABLE F.23 SRS Groundwater Monitoring Data^a

Parameter	Standard or Criterion (S/C)	No. of Samples	No. of Samples Exceeding S/C	No. of Samples not Exceeding S/C	Maximum Value
pH-acid (units)	6.5 ^b	1,018	955	63	2.3
pH-alkaline (units)	8.5 ^b	1,018	23	995	12.5
Silver (mg/L)	0.05 ^c	672	0	672	d
Arsenic (mg/L)	0.05	654	0	654	d
Barium (mg/L)	1 ^c	597	4	593	2.3
Beryllium (mg/L)	0.011 ^e	568	29	539	0.31
Cadmium ^f (mg/L)	0.010 ^c	704	84	620	0.15
Chromium (mg/L)	0.050 ^c	874	51	823	6.3
Copper (mg/L)	1 ^b	527	0	527	d
Iron ^f (mg/L)	0.3 ^b	750	302	448	280
Mercury (mg/L)	0.002 ^c	817	54	763	3.1
Manganese (mg/L)	0.05 ^b	714	218	496	91
Lead ^f (mg/L)	0.05 ^e	781	115	666	4.9
Selenium (mg/L)	0.01 ^c	653	2	651	0.054
Zinc ^f (mg/L)	5 ^b	616	32	584	50
Chloride (mg/L)	250 ^b	835	0	835	d
Fluoride (mg/L)	1.4 ^c	680	1	679	2
Nitrate-N (mg/L)	10 ^c	734	127	607	370
Sulfate (mg/L)	400 ^g	752	1	751	765
Hydrogen sulfide (mg/L)	0.002 ^h	465	1	464	3.0
Cyanide (mg/L)	0.20 ^b	200	0	200	d
Phenol (mg/L)	3.5 ⁱ	631	0	631	d
TOH (mg/L)	0.0007 ^j	846	706	140	94
Endrin (mg/L)	0.0002 ^c	580	0	580	d
Lindane (mg/L)	0.004 ^c	580	1	579	0.01
Methoxychlor (mg/L)	0.1 ^c	580	0	580	d
Toxaphene (mg/L)	0.005 ^c	580	0	580	d
2,4-D (mg/L)	0.1 ^c	592	4	588	0.74
2,4,5-TP (mg/L)	0.01 ^e	592	0	592	d

TABLE F.23 (Cont'd)

Parameter	Standard or Criterion (S/C)	No. of Samples	No. of Samples Exceeding S/C	No. of Samples not Exceeding S/C	Maximum Value
1,1-Dichloroethane (mg/L)	4.05 ^k	24	0	24	d
1,1,1-Trichloroethane (mg/L)	0.2 ^k	359	4	355	0.26
Tetrachloromethane (mg/L)	0.005 ^k	150	6	144	0.14
1,1-Dichloroethylene (mg/L)	0.007 ^k	44	1	43	0.01
trans-1,2-Dichloroethylene (mg/L)	0.27 ⁱ	35	0	35	d
Trichloroethylene (mg/L)	0.005	417	187	230	161
Tetrachloroethylene (mg/L)	0.0007 ^m	411	146	265	269
Gross alpha (pCi/L)	15 ^c	769	104	665	11,500
Gross beta (pCi/L)	0.2 ⁿ	704	476	228	21,000
Radium (pCi/L)	5	618	88	530	128

^aData compiled from 26 existing waste sites Environmental Information Documents, which reported analytical results for samples taken from 1982 to third quarter 1985.

^bNational secondary drinking water regulations (40 CFR 143).

^cNational primary drinking water regulations (40 CFR 141).

^dAll values reported below standard or criterion.

^eEPA 1976 (maximum concentrations for protection of freshwater aquatic life).

^fResults of metals analyses performed between 1982 and 1984 might be inaccurate because of problems with well construction and sampling protocol. Actual groundwater concentration levels of these metals were probably somewhat less.

850 FR 46958.

TABLE F.23 (Cont'd)

^hEPA 1976 (maximum concentration for protection of freshwater aquatic life; detection limit of analysis procedure was 3 mg/L).

ⁱEPA 1986a.

^jNo standard or criterion available, conservatively set at standard for tetrachloroethylene (see footnote k below).

^kEPA 1986b, 1987 (52 FR 25690).

^lDOE 1987b.

^m50 FR 48950 (detection limit of analysis procedure was 0.001 mg/L).

ⁿNo standard or criterion available; set for comparative purposes at the detection limit of 0.2 pCi/L.

Source: DOE 1987b.

TABLE F.24 Groundwater Use at SRS and Vicinity

Type of Use	Aquifers	Pumpage (million m ³ /yr)
Municipal	1 and 2	12.6
	3 and 4	6.6
Industrial/ Agricultural (includes SRS)	1 and 2	27.0
	3 and 4	2.2
Small communities	3 and 4	0.7
Total		49.1

Conversion factor: 1 m³/s = 31.54 million
m³/yr.

Source: DOE 1987a.

TABLE F.25 Estimated Water Requirements for NPR Construction at the Hanford Site

Reactor Type	Source	Water Requirement (m ³ /yr)			Construction Period (yr)
		Reactor	Support Facilities ^a	Total	
HWR	Columbia River Groundwater	0	18,000	18,000	8.5
		380,000 ^b	32,000 ^c	412,000	
LWR	Columbia River Groundwater	0	18,000	18,000	5.0
		380,000 ^b	1,900	381,900	
MHTGR	Columbia River Groundwater	0	20,000	20,000	10
		380,000 ^b	38,000	418,000	

^aData from Heacock (1990).

^bAnnual requirements based on usage of 1 cfs (890,000 m³/yr) (WHC 1989a, 1989b, 1990) for 10 hours per day, 250 days per year plus 1 cfs for dust control for 5 hours per day, 250 days per year.

^cIncluding 9,000 m³/yr for heavy-water processing facility (Roman 1991).

Conversion factor: 1 m³/s = 31.54 million m³/yr.

TABLE F.26 Potential Maximum Water Requirements for NPR Operations at the Hanford Site^a

Reactor Type	Source	Water Requirements (m ³ /yr)						
		Reactor		Support Facilities ^a			Total	
		Cooling System	Other	Tritium Mode	Plutonium Mode	Tritium Mode	Plutonium Mode	
HWR	Columbia River	43,000,000 ^b	122,000 ^c	42,000	303,000	43,164,000	43,425,000	
	Groundwater	0	0	1,130,000	1,120,000	1,130,000	1,120,000	
LWR	Columbia River	44,900,000 ^b	200,000 ^c	144,000	372,000	45,244,000	45,472,000	
	Groundwater	0	0	20,000	0	20,000	0	
MHTGR	Columbia River	35,300,000 ^b	924,000 ^c	42,000	416,000	36,266,000	36,640,000	
	Groundwater	0	0	24,000	180,000	24,000	180,000	

^aData from Roman (1991).

^bFrom Appendix A, Table A.1; Appendix B, Table B.1; and Appendix C, Table C.2.

^cIncluding process water and potable water. Data from Roman (1991).

Conversion factor: 1 m³/s = 31.54 million m³/yr.

TABLE F.27 Annual Cooling Tower Blowdown Rates and Sizes of Evaporation Ponds Required at INEL

Reactor Type	Blowdown Rate ^b (m ³ /yr)	Pond Surface Area ^a (ha)		
		Unheated	Heated	Heated and Spray
HWR	8,400,000	1,300	820	420
LWR	8,700,000	1,300	860	430
MHTGR	6,700,000	1,000	680	340

^aModified from Dinneen (1989) by adding 30% for berm and conservatism in determining total land area disturbed.

^bData from Appendix A, Table A.1; Appendix B, Table B.1; and Appendix C, Table C.2.

Conversion factor: 1 m³/s = 31.54 million m³/yr.

TABLE F.28 Estimated Water Requirements for NPR Construction at INEL

Reactor Type	Source	Water Requirement (m ³ /yr)			Construction Period (yr)
		Reactor	Support Facilities	Total	
HWR	Groundwater	250,000	110,000	360,000	10
LWR	Groundwater	250,000	40,000	290,000	11
MHTGR	Groundwater	48,000	36,000	84,000	11

Source: Arnett, Martineau, and Lehto 1990.

Conversion factor: 1 m³/s = 31.54 million m³/yr.

TABLE F.29 Potential Maximum Water Requirements for NPR Operations at INEL

Reactor Type	Source	Water Requirements (m ³ /yr)					
		Reactor		Support Facilities ^a		Total	
		Cooling System ^b	Other ^c	Tritium Mode	Plutonium Mode	Tritium Mode	Plutonium Mode
HWR	Groundwater	42,600,000	547,000	1,280,000	1,430,000	44,427,000	44,577,000
LWR	Groundwater	45,000,000	200,000	160,000	502,000	45,360,000	45,702,000
MHTGR	Groundwater	34,600,000	524,000	152,000	5,340,000	35,276,000	40,464,000

^aFrom Roman (1991).

^bFrom Appendix A, Table A.1; Appendix B, Table B.1; and Appendix C, Table C.2.

^cFrom Roman (1991); including process and potable water.

Conversion factor: 1 m³/s = 31.54 million m³/yr.

TABLE F.30 Estimated Water Requirements for
NPR Construction at SRS

Reactor Type	Source	Water Requirement (m ³ /yr)		
		Reactor	Support Facilities	Total
HWR	Groundwater or Savannah River	380,000 ^a	9,000 ^b	389,000
LWR	Groundwater or Savannah River	380,000 ^a	61,000 ^c	441,000
MHTGR	Groundwater or Savannah River	380,000 ^a	44,000 ^c	424,000

^aAnnual requirement based on usage of 1 cfs (890,000 m³/yr) (WSRC 1991a-c) for 10 h/day, 250 days/yr plus 1 cfs for dust control for 5 h/day, 250 days/yr.

^bRoman 1991 (for heavy water processing facility only; other facilities already exist).

^cRoman 1991.

Conversion factor: 1 m³/s = 31.54 million m³/yr.

TABLE F.31 Potential Maximum Water Requirements for NPR Operations at SRS

Reactor Type	Source	Water Requirements (m ³ /yr)					
		Reactor		Support Facilities ^a		Total	
		Cooling System ^b	Other ^c	Tritium Mode	Plutonium Mode	Tritium Mode	Plutonium Mode
HWR	Savannah River Groundwater	90,400,000 0	0 548,000	0 3,470,000	0 2,230,000	90,400,000 4,018,000	90,400,000 2,778,000
LWR	Savannah River Groundwater	100,800,000 ^a 0	0 200,000	0 92,600	0 298,000	100,800,000 292,600	100,800,000 498,000
MHTGR	Savannah River Groundwater	74,300,000 ^a 0	0 524,000	0 230,000	0 5,340,000	74,300,000 754,000	74,300,000 5,864,000

^aFrom Roman (1991).

^bIncluding water for blowdown temperature control; data from Appendix A, Table A.1; Appendix B, Table B.1; Appendix C, Table C.2.

^cFrom Roman (1991), including process and potable water.

Conversion factor: 1 m³/s = 31.54 million m³/yr.

**TABLE F.32 Consumptive Water Use for Cooling
Tower Operations at SRS (10^6 m³/yr)**

Reactor Type	Cooling Tower Makeup Withdrawal	Cooling Tower Blowdown ^a	Consumptive Use
HWR	55.3	21.8	33.5
LWR	62.6	24.5	38.1
MHTGR	46.0	18.0	28.0

^aBased on a concentration factor of 2.5.

Sources: Appendix A, Table A.1; Appendix B, Table B.1;
and Appendix C, Table C.2.

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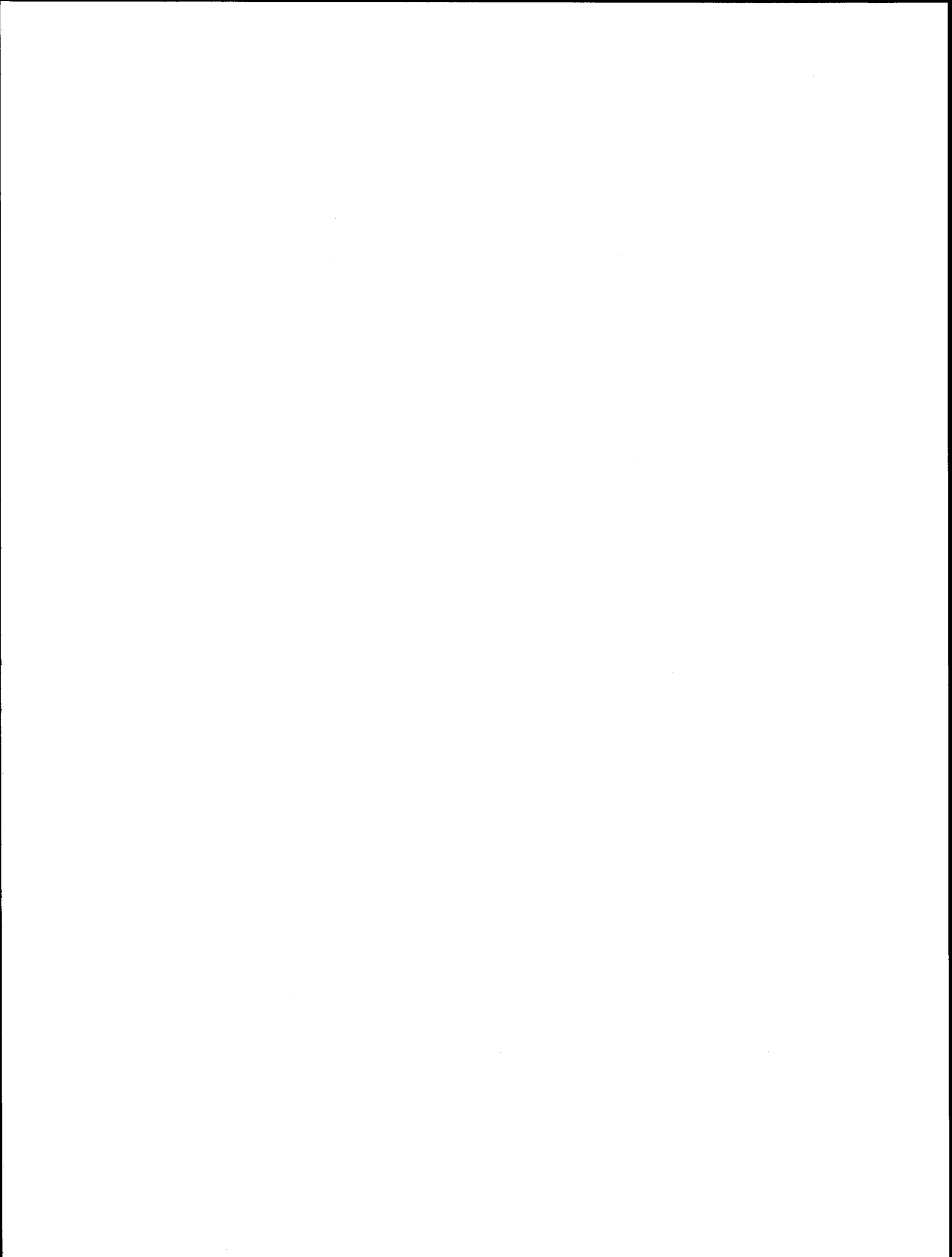
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APPENDIX G:
BIOTIC RESOURCES



APPENDIX G: BIOTIC RESOURCES

G.1 HANFORD SITE

The Hanford Site has been characterized as shrub-steppe (Daubenmire 1970). The primary productivity and the number of vascular plant species of the area are relatively low compared with other semiarid regions in North America. This condition is attributed to the low annual precipitation (16 cm), the low water-holding capacity of the sandy soil, and the occasionally very cold winters. In the early 1800s, the dominant vegetation in the area was big sagebrush, with an understory of perennial bunchgrasses. Today, cheatgrass, an introduced species, is the dominant plant on areas that were cultivated 40 years ago, as well as on previously grazed areas at elevations of less than 240 m (Rickard and Rogers 1983). Wildfires are common in the area. The most recent extensive fire, which occurred in 1984, significantly reduced the shrub cover in affected areas. Most of Hanford was treeless before land settlement; however, for several decades before 1943, trees were planted and irrigated on most of the farms to provide windbreaks and shade.

More than 240 species of plants have been identified on the Hanford Site (Cushing 1990). Some of the common plants are listed in Table G.1. Ten major plant communities within the shrub-steppe habitat of Hanford are (1) sagebrush/bluebunch wheatgrass, (2) sagebrush/cheatgrass-Sandberg's bluegrass, (3) sagebrush-bitterbrush/cheatgrass, (4) greasewood/cheatgrass-saltgrass, (5) winterfat/Sandberg's bluegrass, (6) thyme buckwheat/Sandberg's bluegrass, (7) cheatgrass-tumble mustard, (8) willow, (9) spiny hopsage, and (10) sand dunes (Cushing 1989).

Fifteen species of amphibians and reptiles have been recorded at Hanford (Cushing 1989) (Table G.1). The side-blotched lizard is the most abundant reptile and can be found throughout Hanford. The most common snakes are the gopher snake, western yellow-bellied racer, and Pacific rattlesnake. Striped whipsnakes and desert night snakes also occur on Hanford, but only rarely. Toads and frogs occur near the permanent water bodies and along the Columbia River.

More than 125 species of birds have been identified at Hanford (Rogers and Rickard 1977; Cushing 1989). The horned lark and western meadowlark are the most abundant nesting birds in the shrub-steppe. Table G.1 lists some of the bird species at Hanford. Hanford supports populations of chukar and gray partridge, primarily in the Rattlesnake Hills along the southern boundary of the Hanford Site. Wastewater ponds at Hanford are important habitats for songbirds, shorebirds, ducks, and geese (Fitzner and Price 1973; Fitzner and Rickard 1975; Fitzner and Schreckhise 1979). The American coot is an abundant nesting water bird on these ponds. The most important resident waterfowl species is the Canada goose. The nesting habitat of the goose is confined to the islands of the free-flowing reach of the Columbia River (Hanson and Eberhardt 1971). Forster's tern, ring-billed gulls, and California gulls also nest on these islands. The Columbia River also serves as a major resting area for migrant waterfowl. In addition, several species of hawks and owls use Hanford for foraging and nesting (Fitzner et al. 1980).

TABLE G.1 Partial List of Plant and Animal Species Found at Hanford

Common Name	Scientific Name
Plants	
Cattail	<i>Typha latifolia</i>
Thickspike wheatgrass	<i>Agropyron dasystachyum</i>
Crested wheatgrass	<i>Agropyron desertorum</i>
Cheatgrass	<i>Bromus tectorum</i>
Six-weeks fescue	<i>Festuca octoflora</i>
Pacific fescue	<i>Festuca pacifica</i>
Indian ricegrass	<i>Oryzopsis hymenoides</i>
Sandberg's bluegrass	<i>Poa sandbergii</i>
Sand dropseed	<i>Sporobolus cryptandrus</i>
Needle-and-thread grass	<i>Stipa comata</i>
Black cottonwood	<i>Populus trichocarpa</i>
Peachleaf willow	<i>Salix amygdaloides</i>
Sedge	<i>Carex</i> spp.
Wiregrass	<i>Eleocharis</i> spp.
Reed	<i>Scirpus</i> spp.
Wild onion	<i>Allium</i> spp.
Cluster lily	<i>Brodiaea douglasii</i>
Yellow bell	<i>Fritillaria pudica</i>
Comandra	<i>Comandra umbellata</i>
Snowy buckwheat	<i>Eriogonum niveum</i>
Water persicaria	<i>Polygonum amphibium</i>
Sand dock	<i>Rumex venosus</i>
Spiny hopsage	<i>Grayia spinosa</i>
Jagged chickweed	<i>Holosteum umbellatum</i>
Purple mustard	<i>Chorispora tenella</i>
Tansy mustard	<i>Descurainia pinnata</i>
Spring draba	<i>Draba yerna</i>
Western wallflower	<i>Erysimum asperum</i>
Watercress	<i>Rorippa nasturtium-aquatica</i>
Tumbling mustard	<i>Sisymbrium altissimum</i>
Thelypody	<i>Thelypodium lancinatum</i>
Bitterbrush	<i>Purshia tridentata</i>
Lupine	<i>Lupinus</i> spp.
Scurf pea	<i>Psoralea lanciniata</i>
Filaree	<i>Erodium cicutarium</i>
Desert mallow	<i>Sphaeralcea munroana</i>
Blazing star	<i>Mentzelia albicaulis</i>
Pale evening primrose	<i>Oenothera pallida</i>
Turpentine cymopterus	<i>Cymopterus terebinthinus</i>
Long-leaved phlox	<i>Phlox longifolia</i>
Polemonium	<i>Polemonium micranthum</i>
Phacelia	<i>Phacelia linearis</i>
Gray cryptantha	<i>Cryptantha leucopheea</i>
Beard's tongue	<i>Penstemon acuminatus</i>
Speedwell	<i>Veronica anagallis-aquatica</i>
Yarrow	<i>Achillea millefolium</i>

TABLE G.1 (Cont'd)

Common Name	Scientific Name
Plants (Cont'd)	
Ragweed	<i>Ambrosia acanthicarpa</i>
Absinthe	<i>Artemisia absinthium</i>
Big sagebrush	<i>Artemisia tridentata</i>
Hoary aster	<i>Aster canescens</i>
Balsamroot	<i>Balsamorhiza careyana</i>
Russian knapweed	<i>Centaurea repens</i>
False yarrow	<i>Chaenactis douglasii</i>
Golden aster	<i>Chrysopsis oregana</i>
Green rabbitbrush	<i>Chrysothamnus viscidiflorus</i>
Grey rabbitbrush	<i>Chrysothamnus nauseous</i>
Tickseed	<i>Coreopsis atkinsonia</i>
Hawk's-beard	<i>Crepis atrabarba</i>
Gaillardia	<i>Gaillardia aristata</i>
Gumweed	<i>Grindelia</i> spp.
Cusick's sandflower	<i>Helianthus cusickii</i>
Prickly lettuce	<i>Lactuca serriola</i>
Russia thistle	<i>Salsoa kali</i>
Goldenrod	<i>Solidago occidentalis</i>
Yellow salsify	<i>Tragopogon dubius</i>
Fish	
White sturgeon	<i>Acipenser transmontanus</i>
Bridgelip sucker	<i>Catostomus columbianus</i>
Largescale sucker	<i>Catostomus macrocheilus</i>
Mountain sucker	<i>Catostomus platyrhynchus</i>
Pumpkinseed	<i>Lepomis gibbosus</i>
Bluegill	<i>Lepomis macrochirus</i>
Smallmouth bass	<i>Micropterus dolomieu</i>
Largemouth bass	<i>Micropterus salmoides</i>
White crappie	<i>Pomoxis annularis</i>
Black crappie	<i>Pomoxis nigromaculatus</i>
American shad	<i>Alosa sapidissima</i>
Mottled sculpin	<i>Cottus bairdi</i>
Piute sculpin	<i>Cottus beldingi</i>
Torrent sculpin	<i>Cottus rotheus</i>
Chisel mouth	<i>Acrocheilus alutaceus</i>
Carp	<i>Cyprinus carpio</i>
Peamouth	<i>Mylocheilus caurinus</i>
Northern squawfish	<i>Ptychocheilus oregonensis</i>
Longnose dace	<i>Rhinichthys cataractae</i>
Redside shiner	<i>Richardsonius balteatus</i>
Tench	<i>Tinca tinca</i>
Burbot	<i>Lota lota</i>
Black bullhead	<i>Ictalurus melas</i>
Yellow bullhead	<i>Ictalurus natalis</i>
Brown bullhead	<i>Ictalurus nebulosus</i>

TABLE G.1 (Cont'd)

Common Name	Scientific Name
Fish (Cont'd)	
Channel catfish	<i>Ictalurus dunctatus</i>
Yellow perch	<i>Perca flavescens</i>
Walleye	<i>Stizostedion vitreum</i>
Sand roller	<i>Percopsis transmontana</i>
Pacific lamprey	<i>Entosphenus tridentatus</i>
River lamprey	<i>Lampetra ayresi</i>
Lake whitefish	<i>Coregonus clupeaformis</i>
Coho salmon	<i>Oncorhynchus kisutch</i>
Sockeye salmon	<i>Oncorhynchus nerka</i>
Chinook salmon	<i>Oncorhynchus tshawytscha</i>
Cutthroat trout	<i>Oncorhynchus clarki</i>
Rainbow trout (steelhead)	<i>Oncorhynchus mykiss</i>
Dolly Varden	<i>Salvelinus malma</i>
Mountain whitefish	<i>Prosopium williamsoni</i>
Amphibians and Reptiles	
Great Basin spadefoot toad	<i>Scaphiopus intermontanus</i>
Western toad	<i>Bufo boreas</i>
Woodhouse toad	<i>Bufo woodhouseii</i>
Pacific treefrog	<i>Hyla regilla</i>
Bullfrog	<i>Rana catesbeiana</i>
Sagebrush lizard	<i>Sceloporus graciosus</i>
Side-blotched lizard	<i>Uta stansburiana</i>
Short-horned lizard	<i>Phrynosoma douglassii</i>
Striped whipsnake	<i>Masticophis taeniatus</i>
Western yellow-bellied racer	<i>Coluber constrictor</i>
Gopher snake	<i>Pituophis melanoleucus</i>
Western terrestrial garter snake	<i>Thamnophis elegans</i>
Common garter snake	<i>Thamnophis sirtalis</i>
Desert night snake	<i>Hypsiglena torquata</i>
Pacific rattlesnake	<i>Crotalus viridis</i>
Birds	
Great blue heron	<i>Ardea herodias</i>
Canada goose	<i>Branta canadensis</i>
Mallard	<i>Anas platyrhynchos</i>
Red-tailed hawk	<i>Buteo jamaicensis</i>
Swainson's hawk	<i>Buteo swainsoni</i>
Rough-legged hawk	<i>Buteo lagopus</i>
Ferruginous hawk	<i>Buteo regalis</i>
Sage grouse	<i>Centrocercus urophasianus</i>
California quail	<i>Callipepla californicus</i>
Ring-necked pheasant	<i>Phasianus colchicus</i>
Chukar partridge	<i>Alectoris chukar</i>
Gray partridge	<i>Perdix perdix</i>
American coot	<i>Fulica americana</i>

TABLE G.1 (Cont'd)

Common Name	Scientific Name
Birds (Cont'd)	
California gull	<i>Larus californicus</i>
Ring-billed gull	<i>Larus delawarensis</i>
Forster's tern	<i>Sterna forsteri</i>
Mourning dove	<i>Zenaidura macroura</i>
Horned lark	<i>Eremophila alpestris</i>
Black-billed magpie	<i>Pica pica</i>
Western meadowlark	<i>Sturnella neglecta</i>
Sage sparrow	<i>Amphispiza belli</i>
Mammals	
Merriam's shrew	<i>Sorex merriami</i>
Little brown bat	<i>Myotis lucifugus</i>
Silver-haired bat	<i>Lasionycteris noctivagans</i>
California brown bat	<i>Myotis californicus</i>
Big brown bat	<i>Eptesicus fuscus</i>
Hoary bat	<i>Lasiurus cinereus</i>
Raccoon	<i>Procyon lotor</i>
Mink	<i>Mustela vison</i>
Long-tailed weasel	<i>Mustela frenata</i>
Short-tailed weasel	<i>Mustela erminea</i>
Badger	<i>Taxidea taxus</i>
Striped skunk	<i>Mephitis mephitis</i>
Coyote	<i>Canis latrans</i>
Bobcat	<i>Lynx rufus</i>
Least chipmunk	<i>Eutamias minimus</i>
Yellow-bellied marmot	<i>Marmota flaviventris</i>
Townsend's ground squirrel	<i>Spermophilus townsendii</i>
Northern pocket gopher	<i>Thomomys talpoides</i>
Great Basin pocket mouse	<i>Perognathus parvus</i>
Beaver	<i>Castor canadensis</i>
Western harvest mouse	<i>Reithrodontomys megalotis</i>
Deer mouse	<i>Peromyscus maniculatus</i>
Northern grasshopper mouse	<i>Onychomys leucogaster</i>
Bushy-tailed woodrat	<i>Neotoma cinerea</i>
Muskrat	<i>Ondatra zibethicus</i>
House mouse	<i>Mus musculus</i>
Norway rat	<i>Rattus norvegicus</i>
Porcupine	<i>Erethizon dorsatum</i>
Black-tailed jackrabbit	<i>Lepus californicus</i>
White-tailed jackrabbit	<i>Lepus townsendi</i>
Nuttall's cottontail rabbit	<i>Sylvilagus nuttallii</i>
Mule deer	<i>Odocoileus hemionus</i>
White-tailed deer	<i>Odocoileus virginianus</i>
Elk	<i>Cervus elaphus</i>

Source: Cushing 1989.

About 30 species of mammals have been identified on the Hanford Site (Table G.1) (Cushing 1989, 1990). Most are rodents, of which the Great Basin pocket mouse, deer mouse, Townsend's ground squirrel, northern pocket gopher, western harvest mouse, house mouse, Norway rat, northern grasshopper mouse, and least chipmunk are most common. Seven species of bats also are present. Nuttall's cottontail rabbits are widely distributed throughout the dryland habitats of Hanford, and black-tailed jackrabbits occur at lower elevations. Muskrats and porcupines have been observed along the shorelines of streams, ponds, and ditches, and beavers are residents of the sloughs along the Columbia River. Raccoons, skunks, bobcats, mink, and badgers also are present at Hanford. The coyote is the principal mammalian predator on the site. Larger mammals occurring on-site are the mule deer and elk. The elk herd is located almost entirely within the Rattlesnake Hills. Mule deer are found mostly along the Columbia River and in the Rattlesnake Hills.

The Columbia River is the dominant aquatic ecosystem on Hanford; it supports a large and diverse community of plankton, benthic invertebrates, and fish (Cushing 1990). The Columbia River has been dammed both upstream and downstream from Hanford, and the section of river flowing past Hanford (Hanford Reach) is the last free-flowing section of the Columbia in the United States. Gray and Dauble (1977) list 44 species of fish in the Hanford Reach. Chinook salmon, sockeye salmon, coho salmon, and steelhead trout use the Hanford Reach as a migration route to and from upstream spawning areas and are of economic importance in the Columbia River system (Cushing 1990). Fall chinook salmon and steelhead trout also spawn in the Hanford Reach. Since 1962, the Hanford Reach spawning population has represented about 15% to 20% of the total fall chinook escapement to the river.

The American shad, another anadromous species, may also spawn in the Hanford Reach (Cushing 1990). The upstream range of the shad has been increasing since 1956, when fewer than 10 adult shad ascended McNary Dam, below Hanford. Since then, the number ascending this dam and Priest Rapids Dam, immediately upstream from Hanford, has risen to many thousands each year, and juveniles of this species have been collected in the Hanford Reach. Other fish in the Hanford Reach of importance to sportfishermen are the whitefish, white sturgeon, smallmouth bass, crappie, catfish, walleye, and perch (Cushing 1990). Large populations of nongame fish, including carp, shiners, suckers, and squawfish, also are present.

Table G.2 lists the plants and animals classified as threatened or endangered by the Federal government (DOI 1989) and by the state of Washington (Washington State National Heritage Program 1987). The following discussion applies to the NPR site; no threatened or endangered species or candidate species are found on the WNP-1 site. The following definitions apply to the various status categories pertinent to the Hanford Site:

- *Endangered*: Any species that is in danger of extinction throughout all or a significant portion of its range (Endangered Species Act of 1973).
- *Threatened*: Any species that is likely to become an endangered species within the foreseeable future throughout all or a significant portion of its range (Endangered Species Act of 1973).
- *Category 2*: Any species for which the U.S. Fish and Wildlife Service has information that suggests that proposing to list it as threatened or endangered is possibly appropriate, but for which substantial data on biological vulnerability and threat(s) are not currently known to support the

immediate preparation of rules (Federal Register 1985a). Category 2 species are considered candidates for listing.

No Federally listed plant species occur on the Hanford Site. Columbia yellow cress (Federal Category 2, state endangered) occurs in the riparian areas along the Hanford Reach of the Columbia River. The nearest known population of Columbia milk-vetch is approximately 30 km from the NPR site in the northwestern portion of Hanford near Vermita. Hoover's desert parsley has not been found on the Hanford Site.

Several species of birds listed as threatened or endangered by the Federal government and the state occur on or in the vicinity of the Hanford Site (Cushing 1989, 1990). The peregrine falcon is an occasional migrant through the Hanford area but does not nest there. The bald eagle is a regular winter resident and forages on dead salmon and waterfowl along the Columbia River; it does not nest on Hanford. Perching habitat for bald eagles exists at the historic Hanford

TABLE G.2 Federal and State Threatened, Endangered, and Candidate Species that Are Known to Occur or Could Occur on the Hanford Site

Common Name	Scientific Name	Federal Status ^a	State Status ^a
Molluscs			
Giant Columbia River limpet	<i>Fisherola</i> (= <i>Lanx</i>) <i>nuttallii</i> ^b	C2	NL
Giant Columbia River spire snail	<i>Fluminicola</i> (= <i>Lithoglyphus</i>) <i>columbiana</i> ^b	C2	NL
Plants			
Columbia yellow cress	<i>Rorippa columbiae</i> ^b	C2	E
Columbia milk-vetch	<i>Astragalus columbianus</i>	C2	T
Hoover's desert parsley	<i>Lomatium tuberosum</i>	C2	T
Birds			
Common loon	<i>Gavia immer</i> ^b	NL	C
White pelican	<i>Pelecanus erythrorhynchos</i> ^b	NL	E
Peregrine falcon	<i>Falco peregrinus</i>	E	E
Bald eagle	<i>Haliaeetus leucocephalus</i> ^b	T	T
Swainson's hawk	<i>Buteo swainsoni</i> ^{b,c}	C2	NL
Western sage grouse	<i>Centrocercus urophasianus phaios</i>	C2	NL
Ferruginous hawk	<i>Buteo regalis</i> ^c	C2	T
Sandhill crane	<i>Grus canadensis</i>	NL	E
Long-billed curlew	<i>Numenius americanus</i> ^c	C2	NL

^aStatus codes: C = candidate species; C2 = Category 2 species; E = endangered species; NL = not listed; T = threatened species.

^bOccurs along the Hanford Reach of the Columbia River.

^cOccurs on the NPR site.

Source: Cushing 1989.

Townsite on the Columbia River and is used by eagles during the winter. The WNP-1 site includes no bald eagle perching habitat.

Several Category 2 animal species also are present on Hanford. Ferruginous hawks nest on transmission towers on Hanford; several nests are located about 1 km from the NPR site. Both the ferruginous hawk and Swainson's hawk forage over the NPR site and vicinity. Habitat similar to these foraging sites are relatively common on Hanford. Although no Swainson's hawk nests occur in the vicinity of the NPR site, several active nests are located in the trees at the historic Hanford Townsite, where the proposed pumphouse and intake and discharge pipelines would be located. The long-billed curlew nests on the NPR site and in other grassland areas that are relatively abundant throughout the Hanford Site. A small population of the western sage grouse occurs on the slopes of the Rattlesnake Hills on Hanford. The giant Columbia River limpet and giant Columbia River spire snail are found within the Hanford Reach of the Columbia River, where intake and discharge facilities would be constructed. The limpet is relatively abundant in this reach of the river, while the snail is less common.

State-listed species of the Hanford Site that are not on the Federal list include the white pelican (endangered) and the sandhill crane (endangered) (Cushing 1989, 1990). Both species could occur in areas that would be affected by development of an NPR (Cushing 1990). The white pelican forages for fish in the shallow areas of the Columbia River. Several hundred white pelicans inhabited the Hanford Reach during the summer of 1989, and about ten pelicans wintered there. The sandhill crane migrates through the Hanford area and could be found on the NPR site or along the river. The common loon, a candidate for state listing, occurs on the Hanford Reach of the Columbia River during migration and in the winter (Cushing 1989).

G.2 IDAHO NATIONAL ENGINEERING LABORATORY

Investigators have reported 399 vascular plant species on INEL (Jeppson and Holte 1978; Cholewa and Henderson 1984). Table G.3 lists some of the more common species found. The 20 separate plant associations that have been identified at INEL can be grouped into 6 major plant communities: (1) sagebrush, (2) juniper, (3) crested wheatgrass, (4) Indian ricegrass, (5) agriculture, and (6) wetlands. The plant associations on the NPR site consist of big sagebrush/rabbitbrush/bottlebrushsquirreltail and big sagebrush/Indian ricegrass/needle-and-thread grass.

Sagebrush communities occupy the greater part of INEL and provide important habitat for a variety of animals. Juniper communities are generally associated with areas of increasing elevation in the northwestern and southeastern portions of INEL. Areas planted to crested wheatgrass are found throughout INEL and have existed for more than 25 years with little reinvasion by native species (Staley 1989). Communities dominated by Indian ricegrass are generally restricted to a relatively narrow region near the eastern border of INEL.

One amphibian and nine reptile species have been recorded at INEL. Published ranges indicate that five other reptile and five other amphibian species may be found there as well (Reynolds et al. 1986). The only amphibian that has been observed on INEL is the Great Basin spadefoot toad. Of the nine reptile species occurring there, the short-horned lizard, sagebrush lizard, gopher snake, and western rattlesnake occur commonly throughout INEL, as well as on the NPR site. A diverse insect population is associated with sagebrush communities and is an integral part of the rangeland ecosystem (Stafford 1984; Halford 1981).

TABLE G.3 Partial List of Plant and Animal Species Found at INEL

Common Name	Scientific Name
Plants	
Juniper	<i>Juniperus</i> spp.
Cattail	<i>Typha latifolia</i>
Thickspike wheatgrass	<i>Agropyron dasystachyum</i>
Crested wheatgrass	<i>Agropyron desertorum</i>
Bluebunch wheatgrass	<i>Agropyron spicatum</i>
Cheatgrass	<i>Bromus tectorum</i>
Great Basin wild rye	<i>Elymus cinereus</i>
Wild barley	<i>Hordeum jubatum</i>
Indian ricegrass	<i>Oryzopsis hymenoides</i>
Bluegrass	<i>Poa</i> spp.
Bottle brush squirreltail grass	<i>Sitanion hystrix</i>
Needle-and-thread grass	<i>Stipa comata</i>
Sedge	<i>Carex</i> spp.
Gilia	<i>Gilia polycladon</i>
Rush	<i>Juncus</i> spp.
Plains cottonwood	<i>Populus deltoides</i>
Willow	<i>Salix</i> spp.
Oxytheca	<i>Oxytheca dendroidea</i>
Shadscale saltbush	<i>Atriplex confertifolia</i>
Nuttall saltbush	<i>Atriplex nuttalli</i>
Winterfat	<i>Ceratoides lanata</i>
Milk-vetch	<i>Astragalus giviflorus</i>
Milk-vetch	<i>Astragalus kentrophyta</i>
Bladder pod	<i>Lesquerella kingii</i>
Phlox	<i>Phlox</i> spp.
Phacelia	<i>Phacelia inconspicua</i>
Speedwell	<i>Veronica</i> spp.
Big sagebrush	<i>Artemisia tridentata</i>
Low sagebrush	<i>Artemisia arbuscula</i>
Rabbitbrush	<i>Chrysothamnus viscidiflorus</i>
Thistle	<i>Cirsium</i> spp.
Hawk's-beard	<i>Crepis</i> spp.
Wild lettuce	<i>Lactuca serriola</i>
Dandelion	<i>Taraxacum officinale</i>
Gray horsebrush	<i>Tetradymia canescens</i>
Yellow salsify	<i>Tragopogon dubius</i>
Fish	
Kokanee salmon	<i>Oncorhynchus nerka</i>
Rainbow trout	<i>Oncorhynchus mykiss</i>
Brook trout	<i>Salvelinus fontinalis</i>
Mountain whitefish	<i>Prosopium williamsoni</i>
Speckled dace	<i>Rhinichthys osculus</i>
Shorthead sculpin	<i>Cottus confusus</i>

TABLE G.3 (Cont'd)

Common Name	Scientific Name
Amphibians and Reptiles	
Great Basin spadefoot toad	<i>Scaphiopus intermontanus</i>
Short-horned lizard	<i>Phrynosoma douglassi</i>
Sagebrush lizard	<i>Sceloporus graciosus</i>
Gopher snake	<i>Pituophis melanoleucus</i>
Western rattlesnake	<i>Crotalus viridus</i>
Birds	
Ferruginous hawk	<i>Buteo regalis</i>
Rough-legged hawk	<i>Buteo lagopus</i>
Swainson's hawk	<i>Buteo swainsoni</i>
Golden eagle	<i>Aquila chrysaetos</i>
Prairie falcon	<i>Falco mexicanus</i>
American kestrel	<i>Falco sparverius</i>
Gray partridge	<i>Perdix perdix</i>
Chukar	<i>Alectoris chukar</i>
Ring-necked pheasant	<i>Phasianus colchicus</i>
Blue grouse	<i>Dendragapus obscurus</i>
Sage grouse	<i>Centrocercus urophasianus</i>
American coot	<i>Fulica americana</i>
Long-billed curlew	<i>Numenius americanus</i>
Common snipe	<i>Gallinago gallinago</i>
Mourning dove	<i>Zenaida macroura</i>
Burrowing owl	<i>Athene cunicularia</i>
Long-eared owl	<i>Asio otus</i>
Horned lark	<i>Eremophila alpestris</i>
Barn swallow	<i>Hirundo rustica</i>
Black-billed magpie	<i>Pica pica</i>
Robin	<i>Turdus migratorius</i>
Sage thrasher	<i>Oreoscoptes montanus</i>
Western meadowlark	<i>Sturnella neglecta</i>
Brewer's sparrow	<i>Spizella breweri</i>
Sage sparrow	<i>Amphispiza belli</i>
Mammals	
White-tailed jackrabbit	<i>Lepus townsendii</i>
Black-tailed jackrabbit	<i>Lepus californicus</i>
Nuttall's cottontail	<i>Sylvilagus nuttallii</i>
Pygmy rabbit	<i>Sylvilagus idahoensis</i>
Yellow-bellied marmot	<i>Marmota flaviventris</i>
Townsend's ground squirrel	<i>Spermophilus townsendii</i>
Least chipmunk	<i>Eutamias minimus</i>
Great Basin pocket mouse	<i>Perognathus parvus</i>
Ord's kangaroo rat	<i>Dipodomys ordi</i>
Western harvest mouse	<i>Reithrodontomys megalotis</i>
Deer mouse	<i>Peromyscus maniculatus</i>

TABLE G.3 (Cont'd)

Common Name	Scientific Name
Mammals (Cont'd)	
Northern grasshopper mouse	<i>Onychomys leucogaster</i>
Bushy-tailed wood rat	<i>Neotoma cinerea</i>
Montane vole	<i>Microtus montanus</i>
Badger	<i>Taxidea taxus</i>
Coyote	<i>Canis latrans</i>
Bobcat	<i>Lynx rufus</i>
Long-tailed weasel	<i>Mustela frenata</i>
Mountain lion	<i>Felis concolor</i>
Spotted skunk	<i>Spilogale gracilis</i>
Elk	<i>Cervus elaphus</i>
Mule deer	<i>Odocoileus hemionus</i>
Pronghorn	<i>Antilocapra americana</i>

Source: Staley 1989.

Most of the 184 bird species observed on INEL (Reynolds et al. 1986) could be expected to occur on the NPR site. The most common resident game bird at INEL is the sage grouse. The INEL contains important wintering and breeding/nesting habitat for this species (Connelly and Ball 1983). Although many sage grouse migrate from INEL to off-site areas during the summer, some grouse spend the summer near INEL facilities and the Big Lost River. The ring-necked pheasant, gray partridge, chukar, blue grouse, and mourning dove are other game birds found on INEL.

Eighty-two species of songbirds have been recorded on INEL (Reynolds et al. 1986). The most common species include the horned lark, black-billed magpie, robin, sage thrasher, Brewer's sparrow, sage sparrow, and western meadowlark. These species occur throughout INEL. The sage sparrow, Brewer's sparrow, and sage thrasher are the most common nongame bird species breeding on INEL (Peterson and Best 1983). The INEL is used as a nesting and wintering area by several species of raptors (Craig 1979). Twenty-two species of raptors have been observed on INEL. Rough-legged hawks, American kestrels, prairie falcons, and golden eagles are the most abundant raptors on INEL during the nonbreeding season. The most common nesting raptors on INEL are American kestrels and long-eared owls (Craig 1979).

Thirty-seven species of mammals are found on INEL (Reynolds et al. 1986). Eighteen of these species are rodents, with the most common being Townsend's ground squirrel, least chipmunk, Great Basin pocket mouse, Ord's kangaroo rat, western harvest mouse, deer mouse, bushy-tailed wood rat, and montane vole. Three species of rabbits are common on INEL -- black-tailed jackrabbits, Nuttall's cottontails, and pygmy rabbits. Three of the six species of carnivores are common -- coyote, long-tailed weasel, and badger (Reynolds et al. 1986). The bobcat occurs throughout INEL but is generally uncommon; the mountain lion is considered rare. The spotted skunk is generally uncommon but can be found in basalt outcrops.

The INEL supports resident populations of mule deer, elk, and pronghorn. Mule deer are uncommon and are generally concentrated in the southern and central portions of INEL. Pronghorn are abundant throughout INEL (Arthur et al. [in press]; Reynolds and Rose 1978; Reynolds et al. 1986). During the winter, 4,500 to 6,000 pronghorn, or about 30% of Idaho's total pronghorn population, may be on INEL. Studies conducted during October and November 1983 indicated that pronghorn use of the NPR site was relatively low (Staley 1989).

Since 1983, the elk population has been increasing on INEL (Staley 1989). A study begun in the spring of 1989 identified at least two groups of elk, each with 40 or more animals. One group was located north of the Experimental Breeder Reactor II facility (17 km from the NPR site). The second group was located south of the Twin Buttes (13 km from the NPR site), near the southern boundary of INEL. No elk have been observed on the NPR site; however, elk have used areas near the NPR site in the winter, late fall, or early spring (Staley 1989).

Table G.4 lists plant and animal species classified by Federal or state agencies as threatened or endangered that may occur on INEL. (Status categories used by the Federal Government are defined in Section G.1.) No plant species listed as threatened or endangered by the Federal government occur on INEL, but two species of milk-vetch are candidates for listing (Category 2). Neither of these species occurs on the NPR site (Cholewa and Henderson 1983).

Idaho maintains a State Watch List of plant species that are considered rare and of special interest, even though their populations may not be in jeopardy. Three plants found on INEL are on that list. These are the oxytheca, pincushion cactus, and large flowered-gymosteris. Oxytheca has been found on the NPR site and in several other locations on INEL (Cholewa and Henderson 1983). Four other species (bladder pod, gilia, and two species of milk-vetch) found on INEL have been recommended for the Idaho State Watch List (Cholewa and Henderson 1984). None of these species is found on the NPR site.

Only two animal species listed as threatened or endangered by the Federal government occur on INEL. These species, the bald eagle and peregrine falcon (both endangered), are rare on INEL. Several bald eagles usually winter on or near INEL and have been observed in the western and northern portions of INEL (Reynolds et al 1986). The peregrine falcon has been observed infrequently in the northern portion of INEL (Reynolds et al. 1986). The bald eagle and peregrine falcon are not known to use the NPR site. Four species (three birds and one bat) that are candidates for listing by the Federal government (Category 2) regularly occur on INEL (Table G.4). Of these, only the ferruginous hawk and Swainson's hawk currently use the NPR site; both species occasionally forage over the site.

Eleven animal species that occur on INEL are of special concern to the Idaho Department of Fish and Game (Gleisner 1983) and the U. S. Bureau of Land Management (Table G.4). Five of these species (the long-billed curlew, Swainson's hawk, ferruginous hawk, burrowing owl, and bobcat) occur regularly on INEL. Of these five, all but the long-billed curlew currently use the NPR site.

TABLE G.4 Federal Threatened, Endangered, and Candidate Species and State Species of Special Concern that May Occur on INEL

Common Name	Scientific Name	Federal Status ^a	State Status ^a
Plants			
Painted milk-vetch	<i>Astragalus ceramicus</i>	C2	NL
Wooly pod milk-vetch	<i>Astragalus purshii</i>	C2	NL
Oxytheca	<i>Oxytheca dendroidea</i> ^b	NL	SC
Pincushion cactus	<i>Coryphantha missouriensis</i>	NL	SC
Large-flowered gymosteris	<i>Gymosteris nudicaulis</i>	NL	SC
Birds			
Peregrine falcon	<i>Falco peregrinus</i>	E	SC
Merlin	<i>Falco columbarius</i>	NL	SC
Gyr falcon	<i>Falco rusticolus</i>	NL	SC
Osprey	<i>Pandion haliaetus</i>	NL	SC
Bald eagle	<i>Haliaeetus leucocephalus</i>	E	SC
Swainson's hawk	<i>Buteo swainsoni</i> ^b	C2	SC
Ferruginous hawk	<i>Buteo regalis</i> ^b	C2	SC
Burrowing owl	<i>Athene cunicularia</i> ^b	NL	SC
White-faced ibis	<i>Plegadis chihi</i>	NL	SC
Long-billed curlew	<i>Numenius americanus</i>	C2	SC
Mammals			
Townsend's western big-eared bat	<i>Plecotus townsendi</i>	C2	NL
Bobcat	<i>Lynx rufus</i> ^b	NL	SC

^aStatus codes: C2 = Category 2 species; E = endangered species; NL = not listed; SC = species of special concern; T = threatened species.

^bKnown to occur on the NPR site.

Source: Staley 1989.

G.3 SAVANNAH RIVER SITE

More than 1,300 plant species have been collected on SRS (Wike et al. 1989), some of which are listed in Table G.5. Plant associations of SRS have been classified into six broad cover types: (1) nonforested areas, (2) mixed pine/hardwood forest, (3) pine forest stands, (4) upland hardwoods, (5) bottomland hardwood forest, and (6) swamp (Workman and McLeod 1990). Nonforested areas, such as old fields, typically occur along powerline rights-of-way and in forest openings, or are associated with developed reactor facilities. Old-field communities are early successional communities that form after land is cleared. Both natural and planted pine stands exist on SRS and are managed for timber. Natural stands also often support some hardwood tree species. Pine species on SRS include loblolly, slash, and longleaf pines.

TABLE G.5 Partial List of Plant and Animal Species Found at SRS

Common Name	Scientific Name
Plants	
Longleaf pine	<i>Pinus palustris</i>
Slash pine	<i>Pinus elliottii</i>
Loblolly pine	<i>Pinus taeda</i>
Bald cypress	<i>Taxodium distichum</i>
Cattail	<i>Typha latifolia</i>
Giant cattail	<i>Typha domingensis</i>
Triple-awned grass	<i>Aristida</i> spp.
Cutgrass	<i>Leersia</i> spp.
Sedge	<i>Carex</i> spp.
Duckweed	<i>Lemna perpusilla</i>
Greenbrier	<i>Smilax</i> spp.
Black willow	<i>Salix nigra</i>
Tag alder	<i>Alnus serrulata</i>
Pignut hickory	<i>Carya glabra</i>
Mockernut hickory	<i>Carya tomentosa</i>
River birch	<i>Betula nigra</i>
Red buckeye	<i>Aesculus pavia</i>
Southern red oak	<i>Quercus falcata</i>
Turkey oak	<i>Quercus laevis</i>
Post oak	<i>Quercus stellata</i>
White oak	<i>Quercus alba</i>
Water oak	<i>Quercus nigra</i>
Willow oak	<i>Quercus phellos</i>
Blackjack oak	<i>Quercus marilandica</i>
Bluejack oak	<i>Quercus incana</i>
Sweetgum	<i>Liquidambar styraciflua</i>
Tulip poplar	<i>Liriodendron tulipifera</i>
Poison ivy	<i>Rhus radicans</i>
Holly	<i>Ilex</i> spp.
Red maple	<i>Acer rubrum</i>
Water tupelo	<i>Nyssa aquatica</i>
Black gum	<i>Nyssa sylvatica</i>
Water purslane	<i>Ludwigia palustris</i>
Swamp primrose	<i>Ludwigia leptocarpa</i>
False nettle	<i>Boehmeria aquatica</i>
Dogwood	<i>Cornus</i> spp.
Waxmyrtle	<i>Myrica cerifera</i>
Sparkleberry	<i>Vaccinium stamineum</i>
Huckleberry	<i>Vaccinium</i> spp.
Blackberry	<i>Rubus</i> spp.
Honeysuckle	<i>Lonicera</i> spp.
Water ash	<i>Fraxinus caroliniana</i>

TABLE G.5 (Cont'd)

Common Name	Scientific Name
Amphibians and Reptiles	
American alligator	<i>Alligator mississippiensis</i>
Mole salamander	<i>Ambystoma talpoideum</i>
Southern dusky salamander	<i>Desmognathus auriculatus</i>
Tiger salamander	<i>Ambystoma tigrinum</i>
Spadefoot toad	<i>Scaphiopus holbrooki</i>
Southern toad	<i>Bufo terrestris</i>
Squirrel treefrog	<i>Hyla squirella</i>
Eastern narrow-mouthed toad	<i>Gastrophryne carolinensis</i>
Slimy salamander	<i>Plethodon glutinosus</i>
Bullfrog	<i>Rana catesbiana</i>
Florida cooter	<i>Chrysemys floridana</i>
Snapping turtle	<i>Chelydra serpentina</i>
Eastern mud turtle	<i>Kinosternon subrubrum</i>
Yellow-bellied turtle	<i>Chrysemys scripta</i>
Box turtle	<i>Terrapene carolina</i>
Green anole	<i>Anolis carolinensis</i>
Southern fence lizard	<i>Sceloporus undulatus</i>
Broad-headed skink	<i>Eumeces laticeps</i>
Five-lined skink	<i>Eumeces fasciatus</i>
Brown water snake	<i>Nerodia taxispilota</i>
Eastern garter snake	<i>Thamnophis sirtalis</i>
Southern black racer	<i>Coluber constrictor</i>
Black rat snake	<i>Elaphe guttata</i>
Eastern kingsnake	<i>Lampropeltis getulus</i>
Eastern cottonmouth	<i>Agkistrodon piscivorus</i>
Canebrake rattlesnake	<i>Crotalus horridus</i>
Fish	
Mosquitofish	<i>Gambusia affinis</i>
American eel	<i>Anguilla rostrata</i>
Gizzard shad	<i>Dorosoma cepedianum</i>
Threadfin shad	<i>Dorosoma petenense</i>
Bowfin	<i>Amia calva</i>
Spotted sucker	<i>Minytrema melanops</i>
Redbreast sunfish	<i>Lepomis auritus</i>
Bluegill	<i>Lepomis macrochirus</i>
Bluespotted sunfish	<i>Enneacanthus gloriosus</i>
Redbreast sunfish	<i>Lepomis auritus</i>
Redear sunfish	<i>Lepomis microlophus</i>
Spotted sunfish	<i>Lepomis punctatus</i>
Black crappie	<i>Pomoxis nigromaculatus</i>
Largemouth bass	<i>Micropterus salmoides</i>
Channel catfish	<i>Ictalurus punctatus</i>
White catfish	<i>Ictalurus catus</i>
Flat bullhead	<i>Ictalurus platycephalus</i>

TABLE G.5 (Cont'd)

Common Name	Scientific Name
Fish (Cont'd)	
Chain pickerel	<i>Esox niger</i>
Redfin pickerel	<i>Esox americanus</i>
Birds	
Pied-billed grebe	<i>Podilymbus podiceps</i>
Great blue heron	<i>Ardea herodias</i>
Little blue heron	<i>Florida caerulea</i>
Mallard	<i>Anas platyrhynchos</i>
Green-winged teal	<i>Anas carolinensis</i>
Blue-winged teal	<i>Anas discors</i>
Ring-necked duck	<i>Aythya collaris</i>
Wood duck	<i>Aix sponsa</i>
Hooded merganser	<i>Mergus cucullatus</i>
Turkey vulture	<i>Cathartes aura</i>
Black vulture	<i>Coragyps satratus</i>
Cooper's hawk	<i>Accipiter cooperii</i>
Red-tailed hawk	<i>Buteo jamaicensis</i>
Red-shouldered hawk	<i>Buteo lineatus</i>
Marsh hawk	<i>Circus cyaneus</i>
America kestrel	<i>Falco sparverius</i>
Wild turkey	<i>Meleagris gallopavo</i>
American coot	<i>Fulica americana</i>
Bobwhite quail	<i>Colinus virginianus</i>
Mourning dove	<i>Zenaidura macroura</i>
Barred owl	<i>Strix varia</i>
Short-eared owl	<i>Asio flammeus</i>
Chuck-will's widow	<i>Caprimulgus carolinensis</i>
Common nighthawk	<i>Chordeiles minor</i>
Pileated woodpecker	<i>Hylatomus pileatus</i>
Red-bellied woodpecker	<i>Centurus carolinus</i>
Downy woodpecker	<i>Picoides pubescens</i>
Eastern kingbird	<i>Tyrannus tyrannus</i>
Acadian flycatcher	<i>Empidonax virescens</i>
Rough-winged swallow	<i>Stegidopteryx ruficollis</i>
Blue jay	<i>Cyanocitta cristata</i>
Common crow	<i>Corvus brachyrhynchos</i>
Carolina chickadee	<i>Parus carolinensis</i>
Carolina wren	<i>Thryothorus ludovicianus</i>
Mockingbird	<i>Mimus polyglottus</i>
Eastern bluebird	<i>Sialia sialis</i>
Blue-gray gnatcatcher	<i>Polioptila caerulea</i>
Parula warbler	<i>Parula americana</i>
Pine warbler	<i>Dendroica pinus</i>
Ovenbird	<i>Seiurus aurocapillus</i>
Yellowthroat	<i>Geothlypis trichas</i>

TABLE G.5 (Cont'd)

Common Name	Scientific Name
Birds (Cont'd)	
House sparrow	<i>Passer domesticus</i>
Eastern meadowlark	<i>Sturnella magna</i>
Red-winged blackbird	<i>Agelaius phoeniceus</i>
Common grackle	<i>Quiscalus quiscula</i>
Cardinal	<i>Cardinalis cardinalis</i>
Indigo bunting	<i>Passerina cyanea</i>
Savannah sparrow	<i>Passerculus sandwichensis</i>
White-throated sparrow	<i>Zonotrichia albicollis</i>
Song sparrow	<i>Melospiza melodia</i>
Mammals	
Opossum	<i>Didelphis virginianus</i>
Short-tailed shrew	<i>Blarina brevicauda</i>
Least shrew	<i>Cryptotis parva</i>
Eastern mole	<i>Scalopus aquaticus</i>
Red bat	<i>Lasiurus borealis</i>
Silver-haired bat	<i>Lasionycteris noctivagans</i>
Eastern pipistrelle	<i>Pipistrellus subflavus</i>
Swamp rabbit	<i>Sylvilagus aquaticus</i>
Eastern cottontail	<i>Sylvilagus floridanus</i>
Eastern wood rat	<i>Neotoma floridana</i>
Cotton mouse	<i>Peromyscus gossypinus</i>
Old-field mouse	<i>Peromyscus polionotus</i>
Eastern harvest mouse	<i>Reithrodontomys humulis</i>
Cotton rat	<i>Sigmodon hispidus</i>
Beaver	<i>Castor canadensis</i>
Eastern gray squirrel	<i>Sciurus carolinensis</i>
Gray fox	<i>Urocyon cinereoargenteus</i>
Bobcat	<i>Lynx rufus</i>
River otter	<i>Lutra canadensis</i>
Striped skunk	<i>Mephitis mephitis</i>
Raccoon	<i>Procyon lotor</i>
White-tailed deer	<i>Odocoileus virginianus</i>
Feral hog	<i>Sus scrofa</i>

Sources: Batson, Angerman, and Jones 1985; Dukes 1984; DOE 1987; Wike et al. 1989.

Upland hardwood communities are quite variable on SRS. Scrub oak communities typically occur on the drier, sandier areas. Longleaf pine, turkey oak, bluejack oak, blackjack oak, and post oak dominate these upland forest communities. Understories of triple-awned grass and huckleberry are typical. Oak-hickory upland forest communities occur on more fertile, dry uplands. Characteristic species are white oak, post oak, southern red oak, mockernut hickory, pignut hickory, and loblolly pine, with an understory of sparkleberry, holly, greenbriar, and poison ivy.

Plant community composition of bottomland hardwood forests on SRS is even more variable. This community forms on moist soils along small streams or on old floodplains. Characteristic species include tulip poplar, river birch, sweet gum, willow oak, water oak, and loblolly pine. The understory can include dogwood, honeysuckle, holly, and red buckeye. The Savannah River floodplain supports an extensive swamp forest that covers approximately 3,020 ha of SRS; a natural levee separates the swamp from the river (DOE 1990). Currently, the swamp forest consists of second-growth bald cypress, water tupelo, gum, and many other hardwood tree species (Sharitz, Irwin, and Christy 1974).

About 72% of the NPR site is occupied by a pine plantation of predominantly immature slash pine and loblolly pine ranging in age from newly planted saplings to immature trees. Other vegetation cover types include old field (10%), upland deciduous forest (7%), grassland (6%), and bottomland hardwood forest (5%). Rainbow Bay, adjacent to the NPR site, is a 2.2-ha emergent wetland. Rainbow Bay is surrounded by slash pine and loblolly pine plantations of various ages, with sweetgum, black gum, water oak, blackberry, and waxmyrtle around the edge (Semlitsch 1986).

A partial list of animal species known to occur on SRS is presented in Table G.5. Gibbons and Patterson (1978) provide a comprehensive description of the amphibians and reptiles at SRS, including taxonomy, distribution, and ecological information. Of the amphibian and reptile species found on SRS, 26 are frogs and toads, 17 are salamanders, 31 are snakes, 9 are lizards, 10 are turtles, and 1 is the American alligator (Dukes 1984).

As many as 213 species of resident and migratory birds occur on SRS (Wike et al. 1989), representing a diverse assemblage of waterfowl, raptors, upland gamebirds, and songbirds. Common species are listed in Table G.5. An estimated 10,000 to 17,000 ducks and coots spend the winter on SRS; most congregate on Par Pond and some of the other large ponds on SRS. Upland game birds include the bobwhite, mourning dove, and wild turkey. Many species of songbirds are abundant on SRS, including the eastern kingbird, common crow, mockingbird, blue-gray gnatcatcher, cardinal, Savannah sparrow, and song sparrow (Wike et al. 1989).

Forty-eight species of mammals either have been observed or could occur on SRS (Wike et al. 1989). Seventeen of these are rodents; common species are the old-field mouse, eastern harvest mouse, eastern wood rat, cotton rat, and eastern gray squirrel. Seven species of bats and three species of rabbits occur in various habitats. Thirteen species of carnivores occur on SRS, with the gray fox, bobcat, river otter, raccoon, and striped skunk being fairly common. White-tailed deer are abundant and, along with feral hogs, are hunted annually during controlled public hunts.

Six major types of aquatic habitats occur on SRS -- small ponds, Carolina bays, reservoirs, streams, the Savannah River, and the floodplain swamp associated with the river (Wike et al. 1989). The streams on SRS, ranging from undisturbed blackwater streams to streams

that have been severely degraded by heated water (thermal effluents) released during SRS reactor operations, provide a diversity of aquatic habitats. As a whole, SRS supports a diverse fish fauna typical of other southeastern coastal plain rivers and streams (Bennett and McFarlane 1983); the most common of these species are listed in Table G.5. Fish occurring at SRS include a variety of game species (e.g., largemouth bass, channel catfish, and black crappie), as well as a number of nongame species (e.g., bowfin, spotted sucker, and threadfin shad). Many fish species use the Savannah River Swamp as a spawning and rearing ground. This use is highest when water levels are high.

Portions of the swamp and streams that have received thermal effluent and high flow volumes from reactor operations exhibit a degraded vegetative community because of the die-off of trees and shrubs exposed to high water temperatures (Halverson et al. 1987). Most of these areas are dominated by tag alder and waxmyrtle in the riparian zone and by cutgrass, sedges, false nettle, and water purslane in marshy shallow areas. Thermal effluents also affect aquatic fauna. In general, thermally perturbed sites exhibit reduced diversity and lower populations of macroinvertebrates and fishes (DOE 1987; Halverson et al. 1987).

Nineteen species of plants and animals on SRS are listed by the Federal government as threatened, endangered, or candidates for listing (Category 2) (Table G.6). An additional 46 species are listed by the state as threatened, endangered, rare, or of special concern. Although none of these species has been found on the NPR site at SRS, 21 species are known to occur in areas adjacent to the NPR site or in areas where NPR cooling water would be discharged or withdrawn. Sixteen species could be found on the NPR site or immediate vicinity based on the presence of apparently suitable habitat there. Several plant species that have not been found on SRS have been identified by the FWS as possibly occurring there. These include relict trillium, piedmont bishop-weed, and stream-bank spider-lily.

Most of the Federally and state-listed plant species on SRS are found in various wetlands (Knox and Sharitz 1990). Of these, Florida false loosestrife and green-fringed orchid are known to occur in Rainbow Bay adjacent to the NPR site, and cypress stump sedge occurs in the Savannah River Swamp, where cooling water would be discharged. The remaining plant species are found in drier upland areas of SRS.

A majority of the Federally listed or Category 2 animal species on SRS are restricted to the streams of SRS or the Savannah River Swamp (Wike et al. 1989). These include the American sandburrowing mayfly (Upper Three Runs Creek), brother spike mussel (Savannah River), bighead redhorse (Upper and Lower Three Runs Creeks), shortnose sturgeon (Savannah River), American alligator (throughout), and wood stork (Savannah River Swamp). The shortnose sturgeon is known to spawn upstream of SRS. Shortnose sturgeon larvae have been collected near SRS pumphouses on the river and in the cooling water intake canals. The American alligator is common in Par Pond, Beaver Dam Creek, and the Savannah River Swamp. The wood stork forages within SRS but nests approximately 45 km south of the site. As mitigation associated with the restart of L Reactor, DOE developed and currently maintains alternate foraging habitat for the wood stork at a nearby nature preserve (Coulter 1987).

The remaining Federally listed or Category 2 species on SRS are found in a variety of habitats and locations (Wike et al. 1989). Thirty-six bald eagles have been observed at several locations on SRS, particularly in the vicinity of Par Pond and L Lake. Bald eagles have recently nested below Par Pond dam in the Lower Three Runs Creek drainage (Wike et al. 1989).

TABLE G.6 Federal and State Threatened, Endangered, and Candidate Species that Are Known to Occur or Could Occur on SRS

Common Name	Scientific Name	Federal Status ^a	State Status ^a
Plants			
Milk-pea	<i>Astragalus villosus</i>	NL	R
Yellow wild indigo	<i>Baptisia lanceolata</i>	NL	R
Cypress stump sedge	<i>Carex decomposita</i>	C2	R
Few-fruited sedge	<i>Carex oligocarpa</i>	NL	R
Pink tickseed	<i>Coreopsis rosea</i>	NL	T
Elliott's croton	<i>Croton elliotii</i>	C2	R
Smooth purple cone-flower	<i>Echinacea laevigata</i>	C2	T
Little bur-head	<i>Echinodorus tenellus</i>	NL	T
Gaura	<i>Gaura biennis</i>	NL	R
Piedmont bishop-weed	<i>Ptilimnium nodosum</i>	E	E
Bog spice bush	<i>Lindera subcoriacea</i>	C2	NL
Swamp lobelia	<i>Lobelia boykinii</i>	C2	NL
Florida false loosestrife	<i>Ludwigia spathulata</i>	NL	R
Loose water-milfoil	<i>Myriophyllum laxum</i>	C2	T
Nestronia	<i>Nestronia umbellula</i>	C2	NL
Sandhill lily	<i>Nolina georgiana</i>	NL	R
Nailwort	<i>Paronychia americana</i>	NL	R
Green-fringed orchid	<i>Platanthera lacera</i>	NL	T
Pondweed	<i>Potamogeton foliosus</i>	NL	R
Awed meadow-beauty	<i>Phexia aristosa</i>	C2	R
Stream bank spider-lily	<i>Hymenocallis coronaria</i>	C2	T
Oconee azalea	<i>Rhododendron flammeum</i>	NL	T
Beak-rush	<i>Rhynchospora inundata</i>	NL	R
Beak-rush	<i>Phynchospora tracyi</i>	NL	R
Yellow cress	<i>Rorippa sessiliflora</i>	NL	R
Quill-leaved swamp potato	<i>Sagittaria isoetiformis</i>	NL	T
Trecocarpus	<i>Trecocarpus aethusae</i>	NL	R
Relict trillium	<i>Trillium reliq-num</i>	E	E
Invertebrates			
American sandburrowing mayfly	<i>Dolania americana</i>	C2	NL
Savannah shore mussel	<i>Carunculina pulla</i>	NL	T
Brother spike mussel	<i>Moxostoma anisurum</i>	E	NL
Fishes			
Shortnose sturgeon	<i>Acipenser brevirostrum</i>	E	E
Atlantic sturgeon	<i>Acipenser aestivalis</i>	NL	T
Bighead redhorse	<i>Moxostoma anisurum</i>	C2	NL
Blueback herring	<i>Alosa aestivalis</i>	NL	SC
Hickory shad	<i>Alosa mediocris</i>	NL	SC
American shad	<i>Alosa sapidissima</i>	NL	SC
Amphibians			
Eastern tiger salamander	<i>Ambystoma tigrinum tigrinum</i>	NL	SC

TABLE G.6 (Cont'd)

Common Name	Scientific Name	Federal Status ^a	State Status ^a
Amphibians (Cont'd)			
Upland chorus frog	<i>Pseudacris triseriata feriaru</i>	NL	SC
Bird-voiced treefrog	<i>Hyla avivoca ogechienis</i>	NL	SC
Pig frog	<i>Rana grylio</i>	NL	SC
Pickerel frog	<i>Rana palustris</i>	NL	SC
Carolina crawfish frog	<i>Rana areolata capito</i>	C2	NL
Reptiles			
American alligator	<i>Alligator mississippiensis</i>	T	NL
Spotted turtle	<i>Clemmys guttata</i>	NL	SC
Eastern coral snake	<i>Micrurus fulvius fulvius</i>	NL	SC
Green water snake	<i>Nerodia cyclopion</i>	NL	SC
Eastern slender glass lizard	<i>Ophisaurus attenuatus longicaudus</i>	NL	SC
Pine snake	<i>Pituophis melanoleucus melanoleucus</i>	NL	SC
Carolina swamp snake	<i>Seminatrix pygaea</i>	NL	SC
Smooth earth snake	<i>Virginia valeriae</i>	NL	SC
Birds			
Canvasback	<i>Aythya valisneria</i>	NL	SC
Wood stork	<i>Mycteria americana</i>	E	E
Mississippi kite	<i>Ictinia mississippiensis</i>	NL	SC
Cooper's hawk	<i>Accipiter cooperi</i>	NL	T
Bald eagle	<i>Haliaeetus leucocephalus</i>	E	E
Golden eagle	<i>Aquila chrysaetos</i>	NL	E
American osprey	<i>Pandion haliaetus</i>	NL	T
Peregrine falcon	<i>Falco peregrinus</i>	E	E
Great-horned owl	<i>Bubo virginianus</i>	NL	SC
Red-cockaded woodpecker	<i>Picoides borealis</i>	E	E
Hairy woodpecker	<i>Picoides villosus</i>	NL	SC
Red-headed woodpecker	<i>Melanerpes erythrocephalus</i>	NL	SC
Loggerhead shrike	<i>Lanius ludovicianus</i>	NL	SC
Mammals			
Star-nose mole	<i>Condylura cristata parva</i>	NL	SC
Bobcat	<i>Lynx rufus</i>	NL	SC
River otter	<i>Lutra canadensis</i>	NL	SC
Rafinesque's big-eared bat	<i>Plecotus rafinesquii</i>	C2	NL

^aStatus codes: C2 = Category 2 species; E = endangered species; NL = not listed; R = rare species; SC = species of special concern; T = threatened species.

Sources: Modified from McCort and Wein 1988; Wike et al. 1989; Knox and Sharitz 1990.

Another active bald eagle nest is located on a beaver pond on Pen Branch. This area would receive runoff from the NPR site. Thirty-five red-cockaded woodpecker colony sites have been found on SRS in mature pine forests. Currently, there are 25 red-cockaded woodpeckers on SRS in 7 colony sites (Allen 1990). All colonies are located more than 8 km from the NPR site, and suitable habitat for this species is not present on the NPR site. The Carolina crawfish frog inhabits the burrow systems of gopher tortoises and crawfish, but because of its secretive nature, little is known of this frog's status on the site (Gibbons and Patterson 1978). Rafinesque's big-eared bat has not been studied on SRS, but two specimens were collected in the past and are in the mammal collection at the University of Georgia (Wike et al. 1989).

Ten of the state-listed animal species that occur on SRS have been found in areas where cooling water would be discharged or withdrawn. These include the Atlantic sturgeon, blueback herring, hickory shad, American shad, eastern tiger salamander, pig frog, pickerel frog, eastern slender glass lizard, bobcat, and river otter. All but the glass lizard and bobcat utilize aquatic habitats within these areas. Six state-listed species have been documented at Rainbow Bay adjacent to the NPR site -- eastern tiger salamander, eastern slender glass lizard, smooth earth snake, Cooper's hawk, great-horned owl, and star-nosed mole.

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APPENDIX H:
RADIOLOGICAL IMPACTS OF NORMAL OPERATIONS

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APPENDIX H:

RADIOLOGICAL IMPACTS OF NORMAL OPERATIONS

H.1 INTRODUCTION

Detailed information on potential radiological impacts of normal (nonaccident) operations of new production reactors and their support facilities are presented in this appendix; this information is intended to support the impact assessments summarized in Sections 5.2.6.1, 5.3.6.1, and 5.4.6.1 of the main text. Section H.2 provides background information on the nature of radiation and its biological effects and explains the measures of radiological impacts and the types of individuals and populations for which radiological impacts are measured. Section H.3 describes the methodology used in projecting the impacts of normal operations, with emphasis on the GENII computer code used in this assessment process. Sections H.4 and H.5 present source terms of radioactive releases and detailed results of the impact assessments for the three reactor technologies (HWR, LWR, and MHTGR) and their support facilities at the three alternative sites.

H.2 BACKGROUND

H.2.1 Nature of Radiation and Its Biological Effects

All matter is composed of submicroscopic atoms. An atom, in turn, contains a relatively small nucleus encircled by a number of electrons. The nucleus contains protons, which carry positive charge, and neutrons, which carry no charge. Electrons carry a negative charge. The number of protons determine the chemical element. For example, a carbon nucleus has 6 protons, a uranium nucleus 92 protons. The number of protons and the number of neutrons determine the nuclide. Nuclides of an element (same number of protons) are called isotopes, e.g., uranium-235 and uranium-238. Some nuclides are stable but most are not. Of the approximately 1,700 known nuclides, about 280 are stable.

Over a length of time that varies by nuclide, the structure of unstable nuclides changes, or decays. Unstable nuclides are termed radionuclides because they emit radiation when they decay. They transform spontaneously into either another nuclide or a more stable form of the same nuclide. The decay rate of a radionuclide is its activity. In this document, the unit of activity is the curie (Ci), 3.7×10^{10} disintegrations per second, or approximately the activity of one gram of the radionuclide radium-226.

Radiation is a particle or electromagnetic wave from the decay of a radionuclide that causes ionization when traveling through matter. Matter is said to be ionized when the negatively charged electrons are separated from the positively charged nucleus. When the matter is living tissue, the ionization can damage the cells composing the tissue. The energy imparted by radiation per unit mass of irradiated matter is called the absorbed dose. The unit of absorbed dose used in this EIS is the rad; 1 rad is equal to the amount of radiation that leads to the deposition of 0.01 joule of energy per 1 kg of the absorbing material.

Some forms of radiation are more effective than others in causing biological harm. Dose equivalent is the term used for dose that takes into account both the absorbed dose and the ability, or effectiveness, of different forms of radiation to cause biological harm. Dose equivalent is equal to absorbed dose multiplied by a factor, the quality factor, that takes into account the "biological effectiveness" (degree of harm) of a particular radiation. For electromagnetic radiation (gamma rays and X rays) and beta particles, the quality factor is set at unity (1); for alpha particles, the quality factor is 20. (A beta particle is the electron formed when a neutron decays into a proton; an alpha particle is a doubly charged helium-4 nucleus.) The physical measure (average energy loss per unit track length) associated with the quality factor is called the linear energy transfer (LET). Electromagnetic radiation and beta particles are low LET radiation. Alpha particles, as well as as protons and neutrons, are high LET radiation.

Other than chemically inert noble gases, radionuclides that are inhaled or ingested can interact chemically with body tissues or organs so that the radionuclides may reside in the body for years. While these radionuclides reside in the body, they continue to decay and emit radiation so that internal tissues and organs are exposed over an extended period of time. The quantity that takes this effect into account for an organ or tissue is called the committed dose equivalent (CDE), the time-integrated dose. In the calculations made for this EIS, dose equivalent is accumulated over 50 years (i.e., a 50-year dose commitment).

The various organs of the body have different susceptibilities to harm from radiation. The quantity that takes these different susceptibilities into account to provide a broad indicator of the risk to the health of an individual from radiation is called the committed effective dose equivalent. It is obtained by multiplying the CDE in each major organ or tissue by a weighting factor associated with the risk susceptibility of the tissue or organ and then summing. It is possible that the CDE to an organ is larger than the committed effective dose equivalent if that organ has a small weighting factor. The concept of committed dose applies only for internal pathways. For other pathways (external pathways), there is no long-term residence of radionuclides in the body and the appropriate measure of dose is called the effective dose equivalent (EDE). For convenience, the sum of the committed effective dose equivalent from internal pathways and the effective dose equivalent from external pathways is called the committed effective dose equivalent (CEDE) in this EIS (note that in DOE Order 5400.5, this quantity is called the effective dose equivalent).

The unit used in this EIS for CDE, EDE, and CEDE to an individual is the rem. The corresponding unit for the collective dose to a population (the sum of the doses to members of the population, or the product of the number of exposed individuals and their average dose) is the person-rem. The rem is defined as the dose of a particular type of radiation required to produce the same biological effects as absorption of 0.01 joules of energy from X rays or gamma rays in 1 kilogram of tissue. Thus, 1 rem of one type of radiation is presumed to have the same biological effects as 1 rem of any other kind of radiation. This standard allows comparison of the biological effects of radionuclides that emit different types of radiation.

H.2.2 Health Effects

If the whole body is exposed to a very high dose of radiation, death may occur immediately or within a matter of weeks. The dose that is lethal to about 50% of the exposed population within 60 days of exposure is about 500 rem (Abrahamson et al. 1989). If a limited area of the body is exposed briefly to a very high dose, death may not occur but there may be

other early (sometimes called "acute") effects; for example, doses to the gonads (i.e., testes or ovaries) might cause sterility. Short-term health effects are usually not observed below an acute dose of about 25 rem. However, changes in blood cells have been detected at doses as low as 5 rem (NCRP 1971). Estimated doses to the general population from normal operations of reactors and support facilities are in the range of fractions of millirem (1 millirem = 1/1000 rem) per year of operations and fall well below the level that would produce acute effects.

Doses of radiation that are too weak to cause early effects may have consequences later in life. These are known as long-term health effects. Long-term health effects can be broken down into latent somatic effects and latent genetic effects. The somatic effect of greatest importance is the possible development of cancer 5 to 30 years after exposure. Although the basic processes by which radiation induces cancer may not be fully understood, studies of the survivors of the atomic bombings in Japan, of patients who have been exposed to radiation, of uranium workers, and of workers in the radium-luminizing industry in the 1930s have established that the incidence of cancer is greater in groups who were exposed to high doses of radiation in earlier years than in groups who were not exposed. Latent genetic effects include mutations in the genetic material of exposed persons that affect later generations.

The data that established a link between cancer and radiation were data for persons who received high doses, and no equivalent statistical link has been established between cancer and low doses of radiation. However, a conservative assumption is that the probability of a late effect is proportional to dose (linear dose-risk relationship); then, half the dose would result in half the number of persons developing the effect, a tenth the dose in a tenth the number of persons developing the effect, and so on. Also, a linear dose-risk relationship would enhance the meaningfulness of the collective CEDE as a measure of the effect of radiation on a population because it would not matter how the exposure was distributed among the individuals.

Radiation from releases to the environment can reach individuals via six major pathways that cause radiological exposure (external and internal) and thus health risks: (1) internal doses from breathing contaminated air (inhalation); (2) internal doses from eating contaminated food or drinking contaminated water (ingestion); (3) external doses from surrounding contaminated air (air immersion); (4) external doses from radionuclides deposited on ground surfaces (ground surface or ground shine); (5) external doses from immersion in water (water immersion); and (6) external doses from a radioactive cloud overhead (cloud shine).

These environmental pathways for the contamination and exposure of individuals and the public in general are mutually dependent and interconnected by an array of subpathways. For instance, radionuclides deposited on ground surfaces can be sources of external dose through ground shine and sources of internal dose through the ingestion pathway. Internal dose through ingestion can occur directly as a result of ingestion of contaminated vegetation (through uptake of radionuclides from the ground) or indirectly as a result of ingestion of animal products from animals that have grazed on contaminated vegetation. Also, water bodies contaminated by liquid releases can be sources of external dose through direct contact with the water (immersion) and sources of internal dose, either directly through ingestion of contaminated water or indirectly through ingestion of contaminated aquatic foods or ingestion of food crops that have become contaminated as a result of uptake of radionuclides in contaminated water used for irrigation.

H.2.3 Radiological Impacts

Radiological impacts (potential doses) resulting from releases of radiation from nuclear facilities during normal operations were estimated for certain off-site populations, maximally exposed off-site individuals, and in-plant workers. For normal operations, the maximally exposed off-site individual is defined as the off-site individual who, because of location and habits, would receive a dose that bounds the estimated dose from releases at a specific facility. For each alternative site (Hanford Site, INEL, or SRS), the maximally exposed off-site individual is assumed to reside for 1 year on or near the site boundary at a habitable location (e.g., not on the Columbia River) that would receive the highest average annual radiological exposure. This individual is also assumed to consume locally grown food. The radiological impacts to the maximally exposed off-site individual are evaluated to determine whether operation of NPR facilities would conform to limits imposed by DOE Orders and by regulations issued under authority of the Clean Air Act and Safe Drinking Water Act. The estimated exposure for the maximally exposed off-site individual can also be used as a measure for determining whether protective action guides (PAGs) for the ingestion pathway or for exposure to a radioactive airborne plume might be exceeded in accidents. If projections from field monitoring and other data indicated that PAGs might be exceeded, off-site emergency response authorities might recommend actions to protect the health and safety of the public.

Radiological impacts to the maximally exposed off-site individual are presented as committed effective dose equivalent (CEDE), committed dose equivalent (CDE) to the thyroid and the maximally exposed organ, and lifetime fatal cancer risk. The lifetime fatal cancer risk is taken to be CEDE (in rem) multiplied by 4×10^{-4} (EPA 1989). Doses to the thyroid are of interest because of the affinity of the thyroid for iodine and the associated risk of thyroid cancer from releases containing radioiodine.

The population of interest for normal operations is the projected off-site population in the year 2020 within an 80-km radius of a specific facility from which radiation is assumed to be released. The year 2020 was chosen because it would be approximately the middle of the operating lifetime of the NPR. Radiological impacts to the off-site population are presented as the CEDE, the CDE to the thyroid and gonads, and the incremental cancer deaths and incremental genetic effects per year of normal operations. Incremental genetic effects are taken as the sum of the external effective dose equivalent (EDE) and collective CDE to the gonads (in person-rem) multiplied by 2.6×10^{-4} (EPA 1989). Genetic effects are the calculated number of severe physical or mental defects in the offspring of exposed persons.

H.3 METHODOLOGY FOR ESTIMATING THE RADIOLOGICAL IMPACTS OF NORMAL OPERATIONS

The radiological impacts of normal operations of reactors and support facilities were calculated using Version 1.473 of the GENII computer code (Napier et al. 1988; Rhoads 1990). Site-specific and technology-specific input data were used -- including meteorology, population, food production and consumption, and source terms. The GENII code was used for analysis of normal operations, nonsevere accidents, and transportation accidents. Section H.3.1 briefly describes GENII and outlines the approach used for normal operations. The approaches used for nonsevere accident and transportation accident analyses are discussed in Appendixes I and K, respectively.

H.3.1 GENII Computer Code

The GENII code comprises seven linked computer codes and their associated data files. These codes and their linkages are illustrated in Figure H.1. The codes are connected through data transfer files. The output of one code is stored in a file that can be used by the next code in the system. The computer programs are of three types: (1) user interfaces (i.e., interactive menu-driven programs to assist the user with scenario generation and data input), (2) internal and external dose factor generators, and (3) environmental dosimetry programs. The GENII code is described in three volumes of documentation (Napier et al. 1988). Each of the seven modules in the code is briefly described below.

H.3.1.1 APPRENTICE

The APPRENTICE code helps the user, through a series of interactive menus and questions, to prepare a text input file for the environmental dosimetry programs. In addition, APPRENTICE prepares a batch processing file to manage the file handling needed to control the operations of subsequent codes and to prepare an output report.

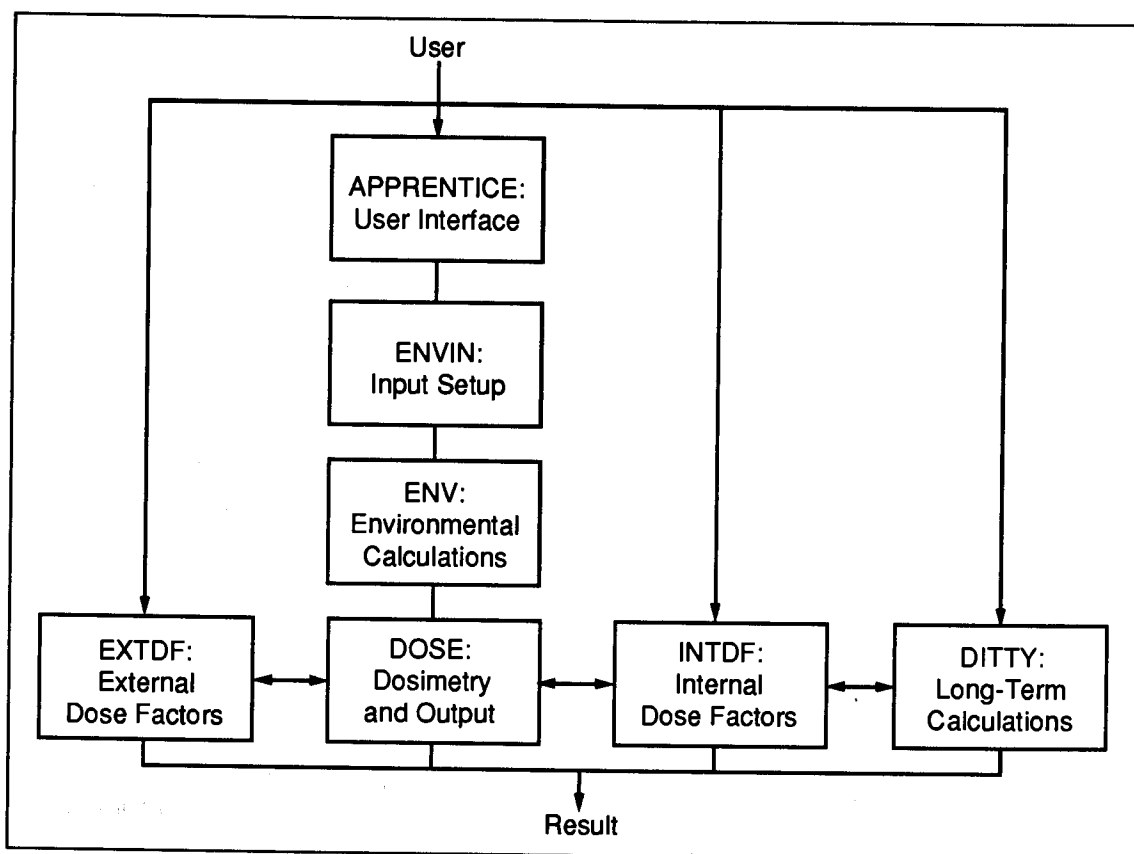


FIGURE H.1 User/Computer Program Interaction in the GENII Computer Code
(Source: Napier et al. 1988)

H.3.1.2 ENVIN

The ENVIN module of the GENII code controls the reading of the input files optionally prepared by APPRENTICE and organizes the input for optimal use in the environmental transport and exposure module, ENV. The ENVIN code interprets the basic input, reads the basic GENII data libraries and other optional input files, and organizes the input into sequential segments on the basis of radionuclide decay chains.

A standardized file that contains scenario, control, and inventory parameters is used as input to ENVIN. Radionuclide inventories can be entered as functions of releases to air or water, concentrations in basic environmental media (air, soil, or water), or concentrations in foods. If certain atmospheric dispersion options have been selected, this module can generate tables of atmospheric dispersion parameters that will be used in later calculations. If the finite plume air submersion option is requested in addition to the atmospheric dispersion calculations, preliminary energy-dependent finite plume dose factors also are prepared. The ENVIN module prepares the data transfer files that are used as input by the ENV module; ENVIN generates the first portion of the calculation documentation -- the run input parameters report.

H.3.1.3 ENV

The ENV module calculates the environmental transfer, uptake, and human exposure to radionuclides that result from the chosen scenario for the defined source term. The code reads the input files from ENVIN and then, for each radionuclide chain, sequentially performs the precalculations to establish the conditions at the start of the exposure scenario. Environmental concentrations of radionuclides are established at the beginning of the scenario by assuming decay of preexisting sources, considering biotic transport of existing subsurface contamination, and defining soil contamination from continuing atmospheric or irrigation depositions. Then, for each year of postulated exposure, the code estimates air, surface soil, deep soil, groundwater, and surface water concentrations of each radionuclide in the chain. Human exposures and intakes of each radionuclide are calculated for (1) pathways of external exposure from finite or infinite atmospheric plumes; (2) inhalation; (3) external exposure from contaminated soil, sediments, and water; (4) external exposure from special geometries; and (5) internal exposures from consumption of terrestrial foods, aquatic foods, drinking water, animal products, and inadvertent intake of soil. The intermediate information on annual media concentrations and intake rates are written to data transfer files. Although these may be accessed directly, they are usually used as input to the DOSE module of GENII.

H.3.1.4 DOSE

The DOSE module reads the annual intake and exposure rates defined by the ENV module and converts the data to radiation dose. External dose is calculated with precalculated factors from the EXTDF module or from a data file prepared outside of GENII. Internal dose is calculated with precalculated factors from the INTDF module.

H.3.1.5 EXTDF

The EXTDF module calculates the external dose-rate factors for submersion in an infinite cloud of radioactive materials, immersion in contaminated water, and direct exposure to plane

or slab sources of radionuclides. Because of what were considered to be significant differences between the dose rate factors generated by the EXTDF module and those listed in DOE report DOE/EH-0070, *External Dose Rate Factors for Calculation of Dose to the Public* (DOE 1988a), the factors given in the DOE report were used for this EIS.

H.3.1.6 INTDF

Using the ICRP-30 model (ICRP 1979-1982), the INTDF module calculates the internal (inhalation and ingestion) dose conversion factors of radionuclides for specific organs. The factors generated by INTDF were used for the calculations presented in this EIS. The differences between those internally generated factors and the factors published in DOE report DOE/EH-0071, *Internal Dose Conversion Factors for Calculation of Dose to the Public* (DOE 1988b), were generally within 1% of each other.

H.3.1.7 DITTY

The DITTY module of GENII was not used in the EIS program. It is designed to calculate long-term (10,000-year) population exposures from nuclear waste disposal sites.

H.3.1.8 Surface Water Transport Models

Surface water transport models available in the GENII code were used for estimating, at selected water use locations, the water concentrations resulting from releases of radionuclides to rivers or lakes. The mathematical models used are the same as those in the LADTAP II computer program (Streng, Peloquin, and Whelan 1986).

H.3.2 Data and Assumptions

H.3.2.1 Data Descriptions

Meteorological Data

The meteorological data used for the three sites (Hanford, INEL, and SRS) were in the form of joint frequency data (JFD) files. A JFD file is a table listing the fractions of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The JFD files were based on measurements over a 5-year period at various locations and at different heights at the three alternative sites. Average meteorological conditions (averaged over the 5-year period) were used. Meteorological data are presented in the report of Kier and Avci (1991).

Population Data

Population projections were based on U.S. Bureau of the Census statistics. Projections were determined for the year 2020 for areas within 80 km of the proposed facilities at each

alternative site. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances up to 80 km. The grid was centered on the facility from which the radionuclides were assumed to be released. A separate grid was established for each major NPR facility or for a group of nearby facilities at each site. Population data are presented in the report of Kier and Avci (1991).

Source Term Data

The source terms (quantities of radionuclides released to the environment over a given period) were estimated on the basis of generic or conceptual designs of NPR facilities and experience with similar facilities. The source terms used to generate the estimated impacts of normal operations are provided in Section H.4 for reactors and Section H.5 for support facilities.

Food Production and Consumption Data

Site-specific data were used for food production and consumption rates when available. Otherwise, generic data from NRC regulatory guides (see, for example, Regulatory Guide 1.109 [NRC 1977]) were used. People living within the 80-km assessment area were assumed to consume food grown only in that area.

H.3.2.2 Calculational Parameters and Assumptions

Doses were calculated with GENII for a hypothetical maximally exposed off-site individual (as defined in Section H.2.3) and for persons (population) living off-site within 80 km of a facility. To estimate the radiological impacts from normal operations of reactors and support facilities, additional assumptions and factors were considered in using GENII, as follows:

- Twenty years of prior deposition of radionuclides on ground surfaces was assumed. Twenty years was selected as an average time throughout the lifetime of the proposed facilities.
- The annual inhalation exposure time was 1 year.
- For the maximally exposed off-site individual, the annual exposure time to the plume and to soil contamination was 0.7 years (based on Regulatory Guide 1.109 [NRC 1977]).
- For the population, the annual exposure time to the plume and to soil contamination was 0.5 years (based on Regulatory Guide 1.109 [NRC 1977]).
- An infinite plume model (infinite plume in the horizontal and vertical directions above ground) was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, ingestion of food crops, and ingestion of animal products.

- For elevated releases, the actual stack height was used as the effective stack height. This neglects plume rise and makes the calculated doses conservative.
- The calculated doses were 50-year committed doses from 1 year of intake.
- Resuspension of particulates was not considered because prior calculations of dust loading in the atmosphere showed that this pathway was negligible compared to others.
- For population exposures, the average individual parameters and options in GENII were used. Average individual parameters assume that the individual breathes and consumes food at average rates (maximum individual parameters assume higher breathing and food consumption rates).

H.3.3 Health Effects Calculations

Doses calculated by GENII were used to estimate health effects according to risk models recommended by the EPA (1989). The incremental cancer fatalities in a population due to radiation exposure were estimated by multiplying the collective CEDE by 4×10^{-4} fatal cancers/person-rem. (In this EIS, the collective CEDE is actually the sum of the collective CEDE and the collective EDE [see Section H.2.1].) The risk of severe genetic effects in a population was estimated by multiplying the sum of the collective CDE to gonads and collective EDE by 2.6×10^{-4} genetic effects/person-rem. Severe genetic effects are defined as effects that would result in severe physical malformations or mental retardation in the offspring of individuals who are exposed.

Although health risk factors are statistical factors and therefore not strictly applicable to individuals, they have been used in the past to estimate the incremental risk to an individual from exposure to radiation. Therefore, the factor 4×10^{-4} per rem of individual CEDE has also been used in this EIS to calculate the individual's incremental cancer risk from exposure to radiation.

H.3.4 Mitigative Actions

The proposed facilities are being designed to limit releases during normal operations to as low as reasonably achievable (ALARA) levels. As described in Section 3.6, environmental monitoring programs are conducted on a continuing basis at Hanford, INEL and SRS, both on-site and off-site. These programs provide for monitoring of effluents that could contain radioactive and/or chemical contaminants in environmental media, including air, water, soil, foodstuffs, wildlife, and vegetation. During normal operations, it is not expected that any mitigative actions will be required. However, if there should be any unusual occurrences, these monitoring programs would alert the responsible authorities, and any necessary steps would be taken to understand and correct the situation. The most significant contributions to mitigating releases will be the incorporation of good design, adequate training, and attention to safety. New designs for NPR incorporate the latest in safety design and human operator interfaces. They would also include redundant systems to prevent releases to the environment.

H.3.5 Presentation of Results

Potential radiological impacts from normal operations are presented in terms of doses and health effects to the maximally exposed off-site individual, off-site population, and in-plant worker. Estimated doses and health effects were calculated for the reactor facility and all supporting facilities for the three reactor technologies. The maximally exposed off-site individual is defined as a person who lives on the site boundary and receives the highest dose of any individual on the site boundary in any direction. The off-site population is defined as persons living within 80 km of the facility under consideration. In general, the maximally exposed off-site individual is a different person for each facility. In this appendix, estimated doses and health effects are provided for each of these individuals separately. In the main text (Sections 5.2.6.1, 5.3.6.1, and 5.4.6.1), the doses are summed to obtain the dose to a hypothetical maximally exposed off-site individual. It is these summed doses to the hypothetical off-site individual that are compared to applicable regulations after adding in the baseline doses.

The NPR facilities may be quite widely spaced at a given alternative site so that the number of persons residing within an 80-km circle of one facility may be different than the number residing within an 80-km circle of another facility. In the main text, estimated population doses from individual facilities for a reactor technology have been summed to obtain the estimated overall population dose for that reactor technology. This is conservative because when the population circles are different, not every person receiving exposure from one facility would receive exposure from all other facilities.

Potential radiological impacts to in-plant workers were not calculated as part of this analysis. The impacts presented in the main text (Sections 5.2.6.2, 5.3.6.2, and 5.4.6.2) are based on (1) experience at operating facilities similar to the proposed NPR program facilities at Hanford, INEL, SRS, and elsewhere, (2) expected NPR facility staffing, and (3) ALARA considerations.

H.3.6 Uncertainties in Estimated Impacts

The sequence of analyses performed to generate the radiological impact estimates from normal operations include (1) estimation of source terms, (2) estimation of environmental transport and uptake of radionuclides, (3) calculation of radiation doses to exposed individuals, and (4) estimation of health effects. There are uncertainties associated with each of these steps. Uncertainties exist in the way the physical systems being analyzed are represented by the computational models, in the data required to exercise the models (due to measurement errors, sampling errors, or natural variability), and in the calculations themselves (e.g., roundoff errors by the computers).

In principle, one can estimate the uncertainty associated with each source and predict the remaining uncertainty in the results of each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final results. However, conducting such a full-scale quantitative uncertainty analysis is neither practical nor a standard practice for an EIS. Instead, the analysis is designed to ensure -- through judicious selection of release scenarios, models, and parameters -- that the results are bounding. This is accomplished by interjecting conservative assumptions into the calculations at each step. The models and parameters used in the calculations are selected in such a way that most intermediate results and, consequently, the final estimates of impacts are greater than what

would be expected. As a result, even though the range of uncertainty in a quantity might be large, the value calculated for the quantity is close to one of the extremes in the range of possible values, so that the chance of the quantity being greater than the calculated value is low (or the chance of the quantity being less than the calculated value if the criteria are such that the quantity has to be maximized). This has been the goal of the radiological assessment for normal operations in this EIS, i.e., to produce results that are conservative.

The degree of conservatism in the calculated results is closely related to the range of possible values the quantity can have. This range is determined by what can be expected to occur realistically. Thus, the only processes and events considered are those that are credible for the conditions under which the physical system being modeled operates. This consideration has also been employed for the analysis of normal operations in this EIS.

In summary, the radiological impact estimates presented in this EIS for normal operations were obtained by

- Using state-of-the-art computational tools,
- Considering the processes and events that can realistically be expected to occur in the proposed NPR facilities and the environment, and
- Making conservative assumptions when there is doubt about the exact nature of the processes and events taking place.

H.4 SOURCE TERMS AND RADIOLOGICAL IMPACTS OF NORMAL OPERATIONS OF REACTORS

H.4.1 Source Terms

Estimated source terms for atmospheric and liquid releases during normal operations of NPRs at Hanford are given in Tables H.1 and H.2, respectively. (For readability, all tables are presented in sequence at the end of the text in this appendix.) Estimated source terms for atmospheric releases during normal operations at INEL are given in Table H.3. At INEL, all radionuclides that would normally be released in liquid effluents at Hanford and SRS are assumed to be released to the atmosphere through heat exchangers or cooling towers. Estimated source terms for atmospheric and liquid releases during normal operations at SRS are given in Tables H.4 and H.5, respectively. The differences in liquid releases at Hanford and SRS are due mainly to differences in site temperature and differences in evaporative losses to the atmosphere from the cooling towers.

For each alternative site, the source terms for the LWR and MHTGR are calculated estimates for the tritium production mode because radiological impacts from normal operations of the LWR and MHTGR would be bounded by operations during that production mode. The source terms for the HWR are based on measurements of releases during both tritium production and plutonium production and represent a composite of releases during the two production modes. The HWR source term also includes an estimated atmospheric release of 2,000 Ci/yr of tritium from the heavy water processing facility.

H.4.2 Radiological Impacts

H.4.2.1 Hanford Site

Estimated radiological impacts to the maximally exposed off-site individual and the 80-km population from 1 year of normal operations of reactors at Hanford are given in Tables H.6 and H.7 for atmospheric releases and in Tables H.8 and H.9 for liquid releases. The impacts of atmospheric releases are larger than those of liquid releases for the HWR and LWR. For the MHTGR, liquid releases are estimated to have slightly larger impacts than atmospheric releases because the entire inventory of tritium in liquid effluents (5.6×10^3 Ci) is assumed to be released as a liquid. In all cases, the CEDE and organ CDE for the maximally exposed off-site individual are estimated to be less than 1 mrem (1×10^{-3} rem). Also, the estimated cancer deaths and genetic effects for the 80-km population are several orders of magnitude below unity so that none would be expected from routine lifetime operations of any NPR at Hanford.

H.4.2.2 Idaho National Engineering Laboratory

Estimated radiological impacts to the maximally exposed off-site individual and the 80-km population from 1 year of normal operations of reactors at INEL are given in Tables H.10 and H.11 for atmospheric releases. All radioactive effluents from the reactors are assumed to be released to the atmosphere by means of evaporation from cooling towers or heat exchangers near ground level. For the MHTGR, the radiological impacts are predominantly due to releases from heat exchangers. In all cases, the CEDE and organ CDE for the maximally exposed off-site individual are estimated to be less than 1 mrem, and the estimated cancer deaths and genetic effects for the 80-km population are several orders of magnitude below unity so that none would be expected from routine lifetime operations of any NPR at INEL.

H.4.2.3 Savannah River Site

Estimated radiological impacts to the maximally exposed off-site individual and the 80-km population from 1 year of normal operations of reactors at SRS are given in Tables H.12 and H.13 for atmospheric releases and in Tables H.14 and H.15 for liquid releases. For the MHTGR, 60% of the liquid tritium is assumed to be released to the atmosphere and 40% as a liquid. In all cases, the CEDE and organ CDE for the maximally exposed off-site individual are estimated to be less than 1 mrem, and the estimated cancer deaths and genetic effects for the 80-km population are several orders of magnitude below unity so that none would be expected from routine lifetime operations of any NPR at SRS.

H.5 SOURCE TERMS AND RADIOLOGICAL IMPACTS OF NORMAL OPERATIONS OF SUPPORT FACILITIES

H.5.1 Source Terms

Estimated source terms for 1 year of normal operations of HWR, LWR, and MHTGR support facilities are given in Tables H.16 through H.18 for Hanford, Tables H.19 through H.21

for INEL, and Tables H.22 through H.24 for SRS. Source terms are presented for releases from fuel processing, plutonium-target processing, and tritium-target processing facilities. With the exception of HWR support facilities at SRS, no liquid radioactive releases are expected.

Releases from fuel fabrication and plutonium-target fabrication facilities are estimated to have substantially smaller radiological impacts than releases from fuel processing, tritium-target processing, and plutonium-target processing facilities. Therefore, neither source terms nor radiological impacts for releases from fuel fabrication and plutonium-target fabrication facilities are presented.

H.5.2 Radiological Impacts

H.5.2.1 Hanford Site

Estimated radiological impacts to the maximally exposed off-site individual and the 80-km population from 1 year of normal operations at Hanford support facilities are given in Tables H.25 and H.26 for atmospheric releases. Liquid effluents from normal operations at Hanford support facilities would be processed and converted into solid waste; hence, no liquid releases of radionuclides are expected to reach the off-site environment. At Hanford, as well as at INEL and SRS, releases from the tritium-target processing facility (5,000 Ci/yr of tritium) are taken to be independent of reactor technology. For plutonium production, the radiological impacts from plutonium-target processing are included in fuel processing.

H.5.2.2 Idaho National Engineering Laboratory

Estimated radiological impacts to the maximally exposed off-site individual and the 80-km population from 1 year of normal operations at INEL support facilities are given in Tables H.27 and H.28 for atmospheric releases. At INEL, there would be no surface or subsurface water pathways for liquid effluents from support facilities to reach the off-site environment. For the LWR technology in plutonium production mode, the radiological impacts from plutonium-target processing are included in fuel processing.

H.5.2.3 Savannah River Site

Estimated radiological impacts to the maximally exposed off-site individual and the 80-km population from 1 year of normal operations at SRS support facilities are given in Tables H.29 and H.30 for atmospheric releases and Tables H.31 and H.32 for liquid releases. For the LWR technology in plutonium production mode, the radiological impacts from plutonium-target processing are included in fuel processing.

TABLE H.1 Estimated Atmospheric Releases of
Radionuclides from NPR Normal Operations at
Hanford

Radionuclide	Atmospheric Releases (Ci/yr)		
	HWR ^a	LWR	MHTGR
H-3	3.5×10^3	3.8×10^3	1.9×10^1
C-14	1.7×10^1	7.3	-
Ar-41	1.4×10^1	3.4×10^1	-
Cr-51	-	9.6×10^{-5}	-
Mn-54	-	5.6×10^{-5}	-
Fe-59	-	2.8×10^{-5}	-
Co-57	-	8.1×10^{-6}	-
Co-58	-	4.8×10^{-4}	-
Co-60	-	1.1×10^{-4}	-
Ni-59	-	5.4×10^{-8}	-
Kr-83m	-	6.6	-
Kr-85m	6.2×10^2	7.9×10^1	4.9
Kr-85	-	1.1×10^3	1.1×10^{-2}
Kr-87	4.3×10^2	3.0×10^1	1.0×10^1
Kr-88	6.5×10^2	1.1×10^2	1.5×10^1
Kr-89	-	-	5.6
Kr-90	-	-	2.3
Rb-86	-	-	1.7×10^{-3}
Sr-89	-	1.8×10^{-4}	1.2×10^{-3}
Sr-90	1.3×10^{-4}	1.9×10^{-5}	2.4×10^{-6}
Sr-91	-	-	3.1×10^{-6}
Y-90	-	1.6×10^{-5}	2.9×10^{-8}
Y-91	-	-	2.4×10^{-8}
Zr-95	-	1.0×10^{-5}	2.3×10^{-8}
Zr-97	-	-	2.0×10^{-7}
Nb-95m	-	6.8×10^{-8}	-
Nb-95	-	4.2×10^{-5}	4.2×10^{-8}
Mo-99	-	-	5.2×10^{-7}
Tc-99m	-	-	4.9×10^{-6}
Ru-103	-	1.7×10^{-5}	2.2×10^{-8}
Ru-105	-	9.0×10^{-6}	2.1×10^{-6}
Ru-106	-	1.0×10^{-7}	2.1×10^{-10}
Rh-103m	-	1.5×10^{-5}	-
Rh-105	-	8.3×10^{-6}	2.5×10^{-7}
Ag-110m	-	-	2.3×10^{-12}
Sb-125	-	6.1×10^{-7}	-
Sb-127	-	-	1.2×10^{-8}
Sb-129	-	-	1.5×10^{-5}
Te-127m	-	-	1.5×10^{-4}
Te-127	-	-	4.9×10^{-4}
Te-129m	-	-	1.3×10^{-3}
Te-129	-	-	1.8×10^{-1}
Te-131m	-	-	2.1×10^{-2}

TABLE H.1 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)		
	HWR ^a	LWR	MHTGR
Te-132	-	-	4.9×10^{-2}
I-129	4.4×10^{-11}	9.5×10^{-11}	-
I-131	1.1×10^{-2}	2.5×10^{-2}	6.8×10^{-2}
I-132	2.1×10^{-2}	-	8.3×10^{-1}
I-133	3.1×10^{-2}	9.0×10^{-2}	4.5×10^{-1}
I-134	2.7×10^{-2}	-	2.0
I-135	3.4×10^{-2}	-	7.3×10^{-1}
Xe-131m	-	2.8×10^3	-
Xe-133m	-	1.6×10^2	-
Xe-133	2.6×10^3	6.2×10^3	2.3
Xe-135m	-	6.0	-
Xe-135	9.5×10^2	1.3×10^3	4.8
Xe-137	-	1.0	-
Xe-138	-	7.0	-
Cs-134	-	1.3×10^{-5}	3.3×10^{-8}
Cs-136	-	1.8×10^{-5}	9.6×10^{-9}
Cs-137	-	2.3×10^{-5}	1.1×10^{-8}
Ba-140	-	4.3×10^{-6}	8.6×10^{-8}
La-140	-	-	1.0×10^{-6}
Ce-141	-	1.3×10^{-5}	4.4×10^{-8}
Ce-143	-	-	9.7×10^{-7}
Ce-144	-	-	2.0×10^{-9}
Pr-143	-	-	9.9×10^{-8}
Nd-147	-	-	5.0×10^{-8}
Pu-239	2.8×10^{-6}	-	-

^aIncludes a release of 2×10^3 Ci/yr of H-3 from the heavy water processing facility.

Sources: HWR - Cadek 1990; LWR - Reid 1990 for H-3, WHC 1990a-b for other radionuclides; MHTGR - Bowman 1990b for H-3, Bowman 1990a for other radionuclides.

TABLE H.2 Estimated Liquid Releases of
Radionuclides from NPR Normal Operations
at Hanford

Radionuclide	Liquid Releases (Ci/yr)		
	HWR	LWR	MHTGR
H-3	4.8×10^2	1.2×10^4	5.6×10^3
Na-24	7.5×10^{-2}	1.6×10^{-2}	-
P-32	5.9×10^{-6}	1.8×10^{-4}	-
S-35	2.2×10^{-5}	-	-
Cr-51	6.5×10^{-2}	6.4×10^{-3}	-
Mn-54	9.3×10^{-6}	4.7×10^{-3}	-
Fe-55	-	7.9×10^{-3}	-
Fe-59	8.7×10^{-5}	2.4×10^{-3}	-
Co-58	8.0×10^{-4}	1.0×10^{-2}	-
Co-60	8.9×10^{-4}	1.4×10^{-2}	-
Ni-63	-	1.7×10^{-3}	-
Zn-65	9.3×10^{-6}	2.7×10^{-4}	-
Rb-86	-	1.9×10^{-5}	-
Sr-89	1.3×10^{-5}	1.9×10^{-4}	-
Sr-90	1.9×10^{-1}	1.0×10^{-5}	-
Sr-91	2.0×10^{-3}	3.5×10^{-3}	-
Sr-92	7.8×10^{-4}	-	-
Y-91m	-	2.2×10^{-4}	-
Y-91	-	9.0×10^{-5}	-
Y-93	-	1.2×10^{-3}	-
Nb-95	8.6×10^{-6}	2.1×10^{-3}	-
Nb-97	-	3.5×10^{-4}	-
Zr-95	1.8×10^{-2}	1.3×10^{-3}	-
Zr-97	-	3.5×10^{-4}	-
Mo-99	-	3.1×10^{-3}	-
Tc-99m	-	2.3×10^{-3}	-
Ru-103	8.8×10^{-4}	4.4×10^{-3}	-
Ru-105	-	2.3×10^{-3}	-
Ru-106	-	7.9×10^{-3}	-
Rh-103m	-	4.1×10^{-3}	-
Rh-105	-	2.2×10^{-3}	-
Pd-109	-	2.5×10^{-4}	-
Ag-110m	-	1.9×10^{-3}	-
Sn-123	-	5.9×10^{-6}	-
Sb-124	1.1×10^{-3}	4.3×10^{-4}	-
Sb-125	1.1×10^{-3}	2.3×10^{-4}	-
Sb-127	-	3.2×10^{-4}	-
Te-125m	-	7.9×10^{-7}	-
Te-127m	-	2.3×10^{-5}	-
Te-127	-	2.7×10^{-4}	-
Te-129m	-	1.0×10^{-4}	-
Te-129	-	2.9×10^{-4}	-
Te-131m	-	6.3×10^{-4}	-
Te-132	-	9.0×10^{-4}	-

TABLE H.2 (Cont'd)

Radionuclide	Liquid Releases (Ci/yr)		
	HWR	LWR	MHTGR
I-129	7.5×10^{-12}	-	-
I-131	1.8×10^{-3}	1.2×10^{-2}	-
I-132	1.1×10^{-3}	8.1×10^{-3}	-
I-133	4.2×10^{-3}	3.1×10^{-2}	-
I-134	6.7×10^{-4}	5.5×10^{-4}	-
I-135	3.3×10^{-3}	3.2×10^{-2}	-
Cs-134	3.1×10^{-3}	5.5×10^{-3}	-
Cs-136	-	6.8×10^{-4}	-
Cs-137	6.1×10^{-2}	7.5×10^{-3}	-
Ba-140	3.0×10^{-3}	8.1×10^{-3}	-
La-140	3.2×10^{-6}	1.3×10^{-2}	-
Ce-141	8.7×10^{-3}	3.1×10^{-4}	-
Ce-143	2.2×10^{-3}	1.2×10^{-3}	-
Ce-144	6.7×10^{-3}	2.9×10^{-3}	-
Pr-143	-	2.0×10^{-5}	-
Pr-144	-	1.0×10^{-3}	-
Nd-147	-	3.7×10^{-3}	-
Pm-147	6.6×10^{-7}	2.7×10^{-4}	-
Pm-148m	-	4.5×10^{-5}	-
Pm-148	-	3.6×10^{-4}	-
W-187	-	1.0×10^{-3}	-
U-237	6.3×10^{-5}	-	-
Np-238	3.6×10^{-5}	-	-
Np-239	3.9×10^{-2}	1.0×10^{-3}	-
Pu-239	1.2×10^{-2}	-	-

Sources: HWR - Cadec 1990; LWR - Reid 1990 for H-3, WHC 1990a-b for other radionuclides; MHTGR - Bowman 1990b for H-3, Bowman 1990a for other radionuclides.

TABLE H.3 Estimated Atmospheric Releases of
Radionuclides from NPR Normal Operations at INEL

Radionuclide	Atmospheric Releases (Ci/yr)		
	HWR ^a	LWR ^a	MHTGR ^b
H-3	4.0×10^3	1.6×10^4	5.6×10^3
C-14	1.7×10^1	7.3	-
Na-24	7.4×10^{-2}	1.6×10^{-2}	-
P-32	5.8×10^{-6}	1.8×10^{-4}	-
S-35	2.2×10^{-5}	-	-
Ar-41	1.4×10^1	3.4×10^1	-
Cr-51	6.5×10^{-2}	6.5×10^{-3}	-
Mn-54	9.3×10^{-6}	4.8×10^{-3}	-
Fe-55	-	7.9×10^{-3}	-
Fe-59	8.6×10^{-5}	2.4×10^{-3}	-
Co-57	-	8.1×10^{-6}	-
Co-58	8.0×10^{-4}	1.0×10^{-2}	-
Co-60	8.9×10^{-4}	1.4×10^{-2}	-
Ni-59	-	5.4×10^{-8}	-
Ni-63	-	1.7×10^{-3}	-
Zn-65	9.3×10^{-6}	2.7×10^{-4}	-
Kr-83m	-	6.6	-
Kr-85m	6.2×10^2	7.9×10^1	4.9
Kr-85	-	1.1×10^3	1.1×10^{-2}
Kr-87	4.3×10^2	3.0×10^1	1.0×10^1
Kr-88	6.5×10^2	1.1×10^2	1.5×10^1
Kr-89	-	-	5.6
Kr-90	-	-	2.3
Rb-86	-	1.9×10^{-5}	1.7×10^{-3}
Sr-89	1.3×10^{-5}	3.7×10^{-4}	1.2×10^{-3}
Sr-90	1.9×10^{-1}	2.9×10^{-5}	2.4×10^{-6}
Sr-91	2.0×10^{-3}	3.5×10^{-3}	3.1×10^{-6}
Sr-92	7.7×10^{-4}	-	-
Y-90	-	1.6×10^{-5}	2.9×10^{-8}
Y-91	-	9.0×10^{-5}	2.4×10^{-8}
Y-91m	-	2.2×10^{-4}	-
Y-93	-	1.2×10^{-3}	-
Nb-95m	-	6.8×10^{-8}	-
Nb-95	8.6×10^{-6}	2.1×10^{-3}	4.2×10^{-8}
Nb-97	-	3.5×10^{-4}	-
Zr-95	1.8×10^{-2}	1.3×10^{-3}	2.3×10^{-8}
Zr-97	-	3.5×10^{-4}	2.0×10^{-7}
Mo-99	-	3.1×10^{-3}	5.2×10^{-7}
Tc-99m	-	2.3×10^{-3}	4.9×10^{-6}
Ru-103	8.8×10^{-4}	4.4×10^{-3}	2.2×10^{-8}
Ru-105	-	2.3×10^{-3}	2.1×10^{-6}
Ru-106	-	7.9×10^{-3}	2.1×10^{-10}
Rh-103m	-	4.1×10^{-3}	-
Rh-105	-	2.2×10^{-3}	2.5×10^{-7}

TABLE H.3 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)		
	HWR ^a	LWR ^a	MHTGR ^b
Pd-109	-	2.5×10^{-4}	-
Ag-110m	-	1.9×10^{-3}	2.3×10^{-12}
Sn-123	-	5.9×10^{-6}	-
Sb-124	1.1×10^{-3}	4.3×10^{-4}	-
Sb-125	1.1×10^{-3}	2.3×10^{-4}	-
Sb-127	-	3.2×10^{-4}	1.2×10^{-8}
Sb-129	-	-	1.5×10^{-5}
Te-125m	-	7.9×10^{-7}	-
Te-127m	-	2.3×10^{-5}	1.5×10^{-4}
Te-127	-	2.7×10^{-4}	4.9×10^{-4}
Te-129m	-	1.0×10^{-4}	1.3×10^{-3}
Te-129	-	2.9×10^{-4}	1.8×10^{-1}
Te-131m	-	6.3×10^{-4}	2.1×10^{-2}
Te-132	-	9.0×10^{-4}	4.9×10^{-2}
I-129	5.2×10^{-11}	9.5×10^{-11}	-
I-131	1.4×10^{-2}	3.7×10^{-2}	6.8×10^{-2}
I-132	2.2×10^{-2}	8.1×10^{-3}	8.3×10^{-1}
I-133	3.4×10^{-2}	1.2×10^{-1}	4.5×10^{-1}
I-134	2.8×10^{-2}	5.5×10^{-4}	2.0
I-135	3.7×10^{-2}	3.2×10^{-2}	7.3×10^{-1}
Xe-131m	-	2.8×10^3	-
Xe-133m	-	1.6×10^2	-
Xe-133	2.6×10^3	6.2×10^3	2.3
Xe-135m	-	6.0	-
Xe-135	9.5×10^2	1.3×10^3	4.8
Xe-137	-	1.0	-
Xe-138	-	7.0	-
Cs-134	3.1×10^{-3}	5.5×10^{-3}	3.3×10^{-8}
Cs-136	-	7.0×10^{-4}	9.6×10^{-9}
Cs-137	6.1×10^{-2}	7.5×10^{-3}	1.1×10^{-8}
Ba-140	3.0×10^{-3}	8.1×10^{-3}	8.6×10^{-8}
La-140	3.1×10^{-6}	1.3×10^{-2}	1.0×10^{-6}
Ce-141	8.7×10^{-3}	3.3×10^{-4}	4.4×10^{-8}
Ce-143	2.2×10^{-3}	1.3×10^{-3}	9.7×10^{-7}
Ce-144	6.7×10^{-3}	2.9×10^{-3}	2.0×10^{-9}
Pr-143	-	2.0×10^{-5}	9.9×10^{-8}
Pr-144	-	1.0×10^{-3}	-
Pm-147	6.5×10^{-7}	2.7×10^{-4}	-
Pm-148m	-	4.5×10^{-5}	-
Pm-148	-	3.6×10^{-4}	-
Nd-147	-	3.7×10^{-3}	5.0×10^{-8}
W-187	-	1.0×10^{-3}	-
U-237	6.3×10^{-5}	-	-

TABLE H.3 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)		
	HWR ^a	LWR ^a	MHTGR ^b
Np-238	3.6×10^{-5}	-	-
Np-239	3.9×10^{-2}	1.0×10^{-3}	-
Pu-239	1.0×10^{-2}	-	-

^aThe atmospheric releases for the HWR and LWR include liquid effluents that would be released to the atmosphere through cooling towers or heat exchangers; the HWR releases include a release of 2×10^3 Ci/yr of H-3 from the heavy water processing facility.

^bThe atmospheric H-3 release for the MHTGR is based on 100% of total H-3 release of 5.6×10^3 Ci/yr being to the atmosphere at INEL.

Sources: HWR - Cadek 1990; LWR - Reid 1990 for H-3, WHC 1990a-b for other radionuclides; MHTGR - Bowman 1990b for H-3, Bowman 1990a for other radionuclides.

TABLE H.4 Estimated Atmospheric Releases of
Radionuclides from NPR Normal Operations at SRS

Radionuclide	Atmospheric Releases (Ci/yr)		
	HWR ^a	LWR	MHTGR ^b
H-3	3.4×10^3	3.8×10^3	3.4×10^3
C-14	1.7×10^1	7.3	-
Ar-41	1.4×10^1	3.4×10^1	-
Cr-51	-	9.6×10^{-5}	-
Mn-54	-	5.6×10^{-5}	-
Fe-59	-	2.8×10^{-5}	-
Co-57	-	8.1×10^{-6}	-
Co-58	-	4.8×10^{-4}	-
Co-60	-	1.1×10^{-4}	-
Ni-59	-	5.4×10^{-8}	-
Kr-83m	-	6.6	-
Kr-85m	6.2×10^2	7.9×10^1	4.9
Kr-85	-	1.1×10^3	1.1×10^{-2}
Kr-87	4.3×10^2	3.0×10^1	1.0×10^1
Kr-88	6.5×10^2	1.1×10^2	1.5×10^1
Kr-89	-	-	5.6
Kr-90	-	-	2.3
Rb-86	-	-	1.7×10^{-3}
Sr-89	-	1.8×10^{-4}	1.2×10^{-3}
Sr-90	1.3×10^{-4}	1.9×10^{-5}	2.4×10^{-6}
Sr-91	-	-	3.1×10^{-6}
Y-90	-	1.6×10^{-5}	2.9×10^{-8}
Y-91	-	-	2.4×10^{-8}
Nb-95m	-	6.8×10^{-8}	-
Nb-95	-	4.2×10^{-5}	4.2×10^{-8}
Zr-95	-	1.0×10^{-5}	2.3×10^{-8}
Zr-97	-	-	2.0×10^{-7}
Mo-99	-	-	5.2×10^{-7}
Tc-99m	-	-	4.9×10^{-6}
Ru-103	-	1.7×10^{-5}	2.2×10^{-8}
Ru-105	-	9.0×10^{-6}	2.1×10^{-6}
Ru-106	-	1.0×10^{-7}	2.1×10^{-10}
Rh-103m	-	1.5×10^{-5}	-
Rh-105	-	8.3×10^{-6}	2.5×10^{-7}
Ag-110m	-	-	2.3×10^{-12}
Sb-125	-	6.1×10^{-7}	-
Sb-127	-	-	1.2×10^{-8}
Sb-129	-	-	1.5×10^{-5}
Te-127m	-	-	1.5×10^{-4}
Te-127	-	-	4.9×10^{-4}
Te-129m	-	-	1.3×10^{-3}
Te-129	-	-	1.8×10^{-1}
Te-131m	-	-	2.1×10^{-2}
Te-132	-	-	4.9×10^{-2}

TABLE H.4 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)		
	HWR ^a	LWR	MHTGR ^b
I-129	3.6×10^{-11}	9.5×10^{-11}	-
I-131	9.1×10^{-3}	2.5×10^{-2}	6.8×10^{-2}
I-132	2.1×10^{-2}	-	8.3×10^{-1}
I-133	2.8×10^{-2}	9.0×10^{-2}	4.5×10^{-1}
I-134	2.7×10^{-2}	-	2.0
I-135	3.3×10^{-2}	-	7.3×10^{-1}
Xe-131m	-	2.8×10^3	-
Xe-133m	-	1.6×10^2	-
Xe-133	2.6×10^3	6.2×10^3	2.3
Xe-135m	-	6.0	-
Xe-135	9.5×10^2	1.3×10^3	4.8
Xe-137	-	1.0	-
Xe-138	-	7.0	-
Cs-134	-	1.3×10^{-5}	3.3×10^{-8}
Cs-136	-	1.8×10^{-5}	9.6×10^{-9}
Cs-137	-	2.3×10^{-5}	1.1×10^{-8}
Ba-140	-	4.3×10^{-6}	8.6×10^{-8}
La-140	-	-	1.0×10^{-6}
Ce-141	-	1.3×10^{-5}	4.4×10^{-8}
Ce-143	-	-	9.7×10^{-7}
Ce-144	-	-	2.0×10^{-9}
Pr-143	-	-	9.9×10^{-8}
Nd-147	-	-	5.0×10^{-8}
Pu-239	2.8×10^{-6}	-	-

^aThe HWR releases include a release of 2×10^{-3} Ci/yr of H-3 from the heavy water processing facility.

^bThe atmospheric H-3 release for the MHTGR is based on 60% of total H-3 release of 5.6×10^3 Ci/yr being to the atmosphere at SRS.

Sources: HWR - Cadek 1990; LWR - Reid 1990 for H-3, WHC 1990a-b for other radionuclides; MHTGR - Bowman 1990b for H-3, Bowman 1990a for other radionuclides.

TABLE H.5 Estimated Liquid Releases of Radionuclides from NPR Normal Operations at SRS

Radionuclide	Liquid Releases (Ci/yr)		
	HWR	LWR	MHTGR
H-3	6.6×10^2	1.2×10^4	2.3×10^3
Na-24	1.6×10^{-1}	1.6×10^{-2}	-
P-32	1.6×10^{-5}	1.8×10^{-4}	-
S-35	6.1×10^{-5}	-	-
Cr-51	7.0×10^{-2}	6.4×10^{-3}	-
Mn-54	9.4×10^{-6}	4.7×10^{-3}	-
Fe-55	-	7.9×10^{-3}	-
Fe-59	9.2×10^{-5}	2.4×10^{-3}	-
Co-58	8.4×10^{-4}	1.0×10^{-2}	-
Co-60	9.0×10^{-4}	1.4×10^{-2}	-
Ni-63	-	1.7×10^{-3}	-
Zn-65	9.4×10^{-6}	2.7×10^{-4}	-
Rb-86	-	1.9×10^{-5}	-
Sr-89	6.0×10^{-5}	1.9×10^{-4}	-
Sr-90	1.9×10^{-1}	1.0×10^{-5}	-
Sr-91	4.6×10^{-3}	3.5×10^{-3}	-
Sr-92	2.0×10^{-3}	-	-
Y-91m	-	2.2×10^{-4}	-
Y-91	-	9.0×10^{-5}	-
Y-93	-	1.2×10^{-3}	-
Nb-95	9.2×10^{-6}	2.1×10^{-3}	-
Nb-97	-	3.5×10^{-4}	-
Zr-95	1.8×10^{-2}	1.3×10^{-3}	-
Zr-97	-	3.5×10^{-4}	-
Mo-99	-	3.1×10^{-3}	-
Tc-99m	-	2.3×10^{-3}	-
Ru-103	9.4×10^{-4}	4.4×10^{-3}	-
Ru-105	-	2.3×10^{-3}	-
Ru-106	-	7.9×10^{-3}	-
Rh-103m	-	4.1×10^{-3}	-
Rh-105	-	2.2×10^{-3}	-
Pd-109	-	2.5×10^{-4}	-
Ag-110m	-	1.9×10^{-3}	-
Sn-123	-	5.9×10^{-6}	-
Sb-124	1.1×10^{-3}	4.3×10^{-4}	-
Sb-125	1.1×10^{-3}	2.3×10^{-4}	-
Sb-127	-	3.2×10^{-4}	-
Te-125m	-	7.9×10^{-7}	-
Te-127m	-	2.3×10^{-5}	-
Te-127	-	2.7×10^{-4}	-
Te-129m	-	1.0×10^{-4}	-
Te-129	-	2.9×10^{-4}	-
Te-131m	-	6.3×10^{-4}	-
Te-132	-	9.0×10^{-4}	-
I-129	1.5×10^{-11}	-	-

TABLE H.5 (Cont'd)

Radionuclide	Liquid Releases (Ci/yr)		
	HWR	LWR	MHTGR
I-131	3.7×10^{-3}	1.2×10^{-2}	-
I-132	3.0×10^{-3}	8.1×10^{-3}	-
I-133	9.5×10^{-3}	3.1×10^{-2}	-
I-134	1.8×10^{-3}	5.5×10^{-4}	-
I-135	8.1×10^{-3}	3.2×10^{-2}	-
Cs-134	3.1×10^{-3}	5.5×10^{-3}	-
Cs-136	-	6.8×10^{-4}	-
Cs-137	6.2×10^{-2}	7.5×10^{-3}	-
Ba-140	3.5×10^{-3}	8.1×10^{-3}	-
La-140	5.5×10^{-6}	1.3×10^{-2}	-
Ce-141	9.3×10^{-3}	3.1×10^{-4}	-
Ce-143	4.0×10^{-3}	1.2×10^{-3}	-
Ce-144	6.8×10^{-3}	2.9×10^{-3}	-
Pr-143	-	2.0×10^{-5}	-
Pr-144	-	1.0×10^{-3}	-
Nd-147	-	3.7×10^{-3}	-
Pm-147	1.8×10^{-6}	2.7×10^{-4}	-
Pm-148m	-	4.5×10^{-5}	-
Pm-148	-	3.6×10^{-4}	-
W-187	-	1.0×10^{-3}	-
U-237	8.0×10^{-5}	-	-
Np-238	6.0×10^{-5}	-	-
Np-239	6.2×10^{-2}	1.0×10^{-3}	-
Pu-239	1.2×10^{-2}	-	-

Sources: HWR - Cadek 1990; LWR - Reid 1990 for H-3, WHC 1990a-b for other radionuclides; MHTGR - Bowman 1990b for H-3, Bowman 1990a for other radionuclides.

TABLE H.6 Estimated Maximally Exposed Off-site Individual Doses and Lifetime Fatal Cancer Risk from 1-Year Exposure to Atmospheric Releases from Reactors at Hanford during Normal Operations

Reactor	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR ^a	9.2×10^{-5}	Thyroid	9.3×10^{-5}	9.3×10^{-5}	4×10^{-8}
LWR ^b	2.8×10^{-4}	Thyroid	3.8×10^{-4}	3.8×10^{-4}	1×10^{-7}
MHTGR ^b	2.6×10^{-6}	Thyroid	6.1×10^{-5}	6.1×10^{-5}	1×10^{-9}

^aResults using a source term based on operations during both tritium production and plutonium production and representing a composite of releases during the two production modes.

^bResults for the tritium production mode, the bounding case.

TABLE H.7 Estimated Population Doses and Health Effects from 1-Year Exposure to Atmospheric Releases from Reactors at Hanford during Normal Operations

Reactor	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR ^a	4.2	3.5	3.9	2×10^{-3}	1×10^{-3}
LWR ^b	4.5	4.2	5.7	2×10^{-3}	1×10^{-3}
MHTGR ^b	1.1×10^{-1}	1.7×10^{-2}	2.5	4×10^{-5}	1×10^{-5}

^aResults using a source term based on operations during both tritium production and plutonium production and representing a composite of releases during the two production modes.

^bResults for the tritium production mode, the bounding case.

TABLE H.8 Estimated Maximally Exposed Off-site Individual Doses and Lifetime Fatal Cancer Risk from 1-Year Exposure to Liquid Releases from Reactors at Hanford during Normal Operations

Reactor	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^a	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR ^b	1.5×10^{-5}	Bone sur.	1.6×10^{-4}	3.8×10^{-6}	6×10^{-9}
LWR ^c	1.3×10^{-5}	LL. int.	1.6×10^{-5}	1.5×10^{-5}	5×10^{-9}
MHTGR ^c	5.7×10^{-6}	S. int.	6.7×10^{-6}	6.3×10^{-6}	2×10^{-9}

^aBone sur. = bone surface; LL. int. = lower large intestine; S. int. = small intestine.

^bResults using a source term based on operations during both tritium production and plutonium production and representing a composite of releases during the two production modes.

^cResults for the tritium production mode, the bounding case.

TABLE H.9 Estimated Population Doses and Health Effects from 1-Year Exposure to Liquid Releases from Reactors at Hanford during Normal Operations

Reactor	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR ^a	4.1×10^{-2}	2.2×10^{-2}	1.9×10^{-2}	2×10^{-5}	6×10^{-6}
LWR ^b	3.3×10^{-1}	3.6×10^{-1}	3.7×10^{-1}	1×10^{-4}	9×10^{-5}
MHTGR ^b	1.5×10^{-1}	1.7×10^{-1}	1.7×10^{-1}	6×10^{-5}	4×10^{-5}

^aResults using a source term based on operations during both tritium production and plutonium production and representing a composite of releases during the two production modes.

^bResults for the tritium production mode, the bounding case.

TABLE H.10 Estimated Maximally Exposed Off-site Individual Doses and Lifetime Fatal Cancer Risk from 1-Year Exposure to Atmospheric Releases from Reactors at INEL during Normal Operations

Reactor	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR ^a	1.2×10^{-4}	Bone sur. ^b	7.8×10^{-4}	4.3×10^{-5}	5×10^{-8}
LWR ^c	3.1×10^{-4}	Thyroid	3.6×10^{-4}	3.6×10^{-4}	1×10^{-7}
MHTGR ^c	1.3×10^{-4}	Thyroid	1.7×10^{-4}	1.7×10^{-4}	5×10^{-8}

^aResults using a source term based on operations during both tritium production and plutonium production and representing a composite of releases during the two production modes.

^bBone sur. = bone surface.

^cResults for the tritium production mode, the bounding case.

TABLE H.11 Estimated Population Doses and Health Effects from 1-Year Exposure to Atmospheric Releases from Reactors at INEL during Normal Operations

Reactor	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR ^a	1.7	6.7×10^{-1}	6.2×10^{-1}	7×10^{-4}	2×10^{-4}
LWR ^b	4.7	5.1	5.4	2×10^{-3}	1×10^{-3}
MHTGR ^b	2.0	2.2	2.4	8×10^{-4}	6×10^{-4}

^aResults using a source term based on operations during both tritium production and plutonium production and representing a composite of releases during the two production modes.

^bResults for the tritium production mode, the bounding case.

TABLE H.12 Estimated Maximally Exposed Off-site Individual Doses and Lifetime Fatal Cancer Risk from 1-Year Exposure to Atmospheric Releases from Reactors at SRS during Normal Operations

Reactor	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR ^a	3.1×10^{-5}	Thyroid	2.9×10^{-5}	2.9×10^{-5}	1×10^{-8}
LWR ^b	4.9×10^{-5}	Thyroid	7.5×10^{-5}	7.5×10^{-5}	2×10^{-8}
MHTGR ^b	1.1×10^{-5}	Thyroid	4.6×10^{-5}	4.6×10^{-5}	4×10^{-9}

^aResults using a source term based on operations during both tritium production and plutonium production and representing a composite of releases during the two production modes.

^bResults for the tritium production mode, the bounding case.

TABLE H.13 Estimated Population Doses and Health Effects from 1-Year Exposure to Atmospheric Releases from Reactors at SRS during Normal Operations

Reactor	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR ^a	2.1	1.5	1.7	8×10^{-4}	6×10^{-4}
LWR ^b	2.8	2.5	3.2	1×10^{-3}	8×10^{-4}
MHTGR ^b	6.9×10^{-1}	7.0×10^{-1}	1.7	3×10^{-4}	2×10^{-4}

^aResults using a source term based on operations during both tritium production and plutonium production and representing a composite of releases during the two production modes.

^bResults for the tritium production mode, the bounding case.

TABLE H.14 Estimated Maximally Exposed Off-site Individual Doses and Lifetime Fatal Cancer Risk from 1-Year Exposure to Liquid Releases from Reactors at SRS during Normal Operations

Reactor	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^a	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR ^a	8.2×10^{-5}	Bone sur.	8.4×10^{-4}	3.8×10^{-5}	3×10^{-8}
LWR ^b	3.8×10^{-5}	LL. int.	4.9×10^{-5}	4.5×10^{-5}	2×10^{-8}
MHTGR ^b	5.6×10^{-6}	S. int.	6.5×10^{-6}	6.2×10^{-6}	2×10^{-9}

^aBone sur. = bone surface; LL. int. = lower large intestine; S. int. = small intestine.

^bResults using a source term based on operations during both tritium production and plutonium production and representing a composite of releases during the two production modes.

^cResults for the tritium production mode, the bounding case.

TABLE H.15 Estimated Population Doses and Health Effects from 1-Year Exposure to Liquid Releases from Reactors at SRS during Normal Operations

Reactor	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR ^a	1.3	8.5×10^{-1}	8.1×10^{-1}	5×10^{-4}	2×10^{-4}
LWR ^b	11	13	13	4×10^{-3}	3×10^{-3}
MHTGR ^b	2.2	2.4	2.4	9×10^{-4}	6×10^{-4}

^aResults using a source term based on operations during both tritium production and plutonium production and representing a composite of releases during the two production modes.

^bResults for the tritium production mode, the bounding case.

TABLE H.16 Estimated Annual Atmospheric Releases from HWR Fuel Processing and Target Processing Facilities at Hanford

Radionuclide	Atmospheric Releases (Ci/yr)				
	Tritium Production			Plutonium Production	
	Fuel Processing	Uranium Processing	Target Processing	Fuel and Target Processing	Uranium Processing
H-3	6.2×10^2	2.2	5.0×10^3	5.5×10^2	1.9
Kr-85	2.7×10^5	-	-	2.3×10^5	-
Sr-89	4.4×10^{-8}	2.4×10^{-10}	-	6.1×10^{-8}	2.4×10^{-10}
Sr-90	1.2×10^{-7}	6.4×10^{-10}	-	9.8×10^{-8}	4.4×10^{-10}
Y-90	1.2×10^{-7}	6.4×10^{-10}	-	9.8×10^{-8}	4.4×10^{-10}
Y-91	9.7×10^{-8}	5.3×10^{-10}	-	1.3×10^{-7}	5.2×10^{-10}
Zr-95	1.6×10^{-7}	8.8×10^{-10}	-	2.1×10^{-7}	8.3×10^{-10}
Nb-95	3.3×10^{-7}	1.8×10^{-9}	-	1.3×10^{-7}	2.1×10^{-11}
Nb-95m	1.9×10^{-9}	1.1×10^{-11}	-	2.6×10^{-9}	1.0×10^{-11}
Tc-99	1.6×10^{-11}	8.8×10^{-14}	-	1.5×10^{-11}	6.4×10^{-14}
Ru-103	6.9×10^{-9}	3.8×10^{-11}	-	1.3×10^{-8}	3.8×10^{-11}
Ru-106	1.0×10^{-7}	5.6×10^{-10}	-	1.7×10^{-7}	4.1×10^{-10}
Rh-103m	6.9×10^{-9}	3.8×10^{-11}	-	1.3×10^{-8}	3.8×10^{-11}
Ag-110m	2.7×10^{-9}	1.5×10^{-11}	-	2.0×10^{-9}	5.8×10^{-12}
Sn-123	1.3×10^{-9}	6.8×10^{-12}	-	2.1×10^{-9}	5.9×10^{-12}
Sb-125	3.9×10^{-9}	2.1×10^{-11}	-	4.9×10^{-9}	1.5×10^{-11}
Te-125m	9.4×10^{-10}	5.1×10^{-12}	-	1.2×10^{-9}	3.6×10^{-12}
Te-127	1.8×10^{-9}	1.0×10^{-11}	-	2.8×10^{-9}	8.6×10^{-12}
Te-127m	1.9×10^{-9}	1.0×10^{-11}	-	2.9×10^{-9}	9.0×10^{-12}
Te-129	7.2×10^{-11}	3.9×10^{-13}	-	1.4×10^{-10}	4.1×10^{-13}
Te-129m	1.1×10^{-10}	6.1×10^{-13}	-	2.2×10^{-10}	6.4×10^{-13}
I-129	2.9×10^{-2}	-	-	3.4×10^{-2}	-
Cs-134	8.9×10^{-8}	4.8×10^{-10}	-	3.6×10^{-8}	1.8×10^{-10}
Cs-137	1.2×10^{-7}	6.4×10^{-10}	-	1.1×10^{-7}	4.5×10^{-10}
Ce-141	3.3×10^{-9}	1.8×10^{-11}	-	5.3×10^{-9}	1.8×10^{-11}
Ce-144	1.3×10^{-6}	7.0×10^{-9}	-	1.3×10^{-6}	5.5×10^{-9}
Pr-144	1.3×10^{-6}	7.0×10^{-9}	-	1.3×10^{-6}	5.5×10^{-9}
Pr-144m	1.6×10^{-8}	8.5×10^{-11}	-	1.5×10^{-8}	6.5×10^{-11}
Pm-147	1.7×10^{-7}	9.4×10^{-10}	-	2.4×10^{-7}	9.4×10^{-10}
Pm-148	1.5×10^{-10}	8.2×10^{-13}	-	2.1×10^{-10}	9.2×10^{-13}
Sm-151	3.6×10^{-10}	2.0×10^{-12}	-	8.4×10^{-10}	2.4×10^{-12}
Eu-154	4.4×10^{-9}	2.4×10^{-11}	-	2.0×10^{-9}	9.7×10^{-12}
Eu-155	2.4×10^{-9}	1.3×10^{-11}	-	2.8×10^{-9}	8.0×10^{-12}
U-235	7.8×10^{-20}	4.8×10^{-10}	-	2.2×10^{-13}	1.8×10^{-10}
U-238	5.1×10^{-21}	3.1×10^{-11}	-	7.1×10^{-12}	7.1×10^{-11}
Np-237	6.7×10^{-13}	3.6×10^{-15}	-	8.8×10^{-13}	2.8×10^{-9}
Pu-238	2.8×10^{-9}	1.5×10^{-11}	-	2.0×10^{-9}	6.6×10^{-6}
Pu-239	1.9×10^{-11}	1.0×10^{-13}	-	1.3×10^{-9}	1.8×10^{-13}
Pu-240	1.6×10^{-11}	8.6×10^{-14}	-	2.4×10^{-10}	7.9×10^{-14}
Pu-241	5.0×10^{-9}	2.7×10^{-11}	-	1.8×10^{-8}	1.5×10^{-11}

TABLE H.16 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)				
	Tritium Production			Plutonium Production	
	Fuel Processing	Uranium Processing	Target Processing	Fuel and Target Processing	Uranium Processing
Am-241	8.0×10^{-12}	4.4×10^{-14}	-	3.1×10^{-11}	2.6×10^{-14}
Am-243	-	-	-	1.3×10^{-16}	4.9×10^{-19}
Cm-242	-	-	-	1.8×10^{-11}	5.5×10^{-16}
Cm-244	-	-	-	2.2×10^{-14}	1.2×10^{-16}

Source: WHC 1989a.

TABLE H.17 Estimated Annual Atmospheric Releases from LWR Fuel Processing and Target Processing Facilities at Hanford

Radionuclide	Atmospheric Releases (Ci/yr)				
	Tritium Production			Plutonium Production	
	Fuel Processing	Uranium Processing	Target Processing	Fuel and Target Processing	Uranium Processing
H-3	6.1×10^2	2.5	5.0×10^3	3.7×10^2	1.5
C-14	7.6	-	-	1.9×10^1	-
Cr-51	1.8×10^{-8}	-	-	2.0×10^{-6}	-
Mn-54	1.6×10^{-6}	-	-	7.9×10^{-7}	-
Fe-55	1.2×10^{-5}	-	-	3.8×10^{-6}	-
Fe-59	8.5×10^{-9}	-	-	1.2×10^{-9}	-
Co-58	6.2×10^{-7}	-	-	3.3×10^{-6}	-
Co-60	5.3×10^{-6}	-	-	1.5×10^{-6}	-
Ni-59	6.2×10^{-9}	-	-	1.7×10^{-9}	-
Ni-63	7.9×10^{-7}	-	-	2.2×10^{-7}	-
Kr-85	6.6×10^5	-	-	3.5×10^5	-
Sr-89	1.8×10^{-6}	1.1×10^{-11}	-	3.5×10^{-6}	2.0×10^{-11}
Sr-90	5.7×10^{-6}	3.4×10^{-11}	-	3.2×10^{-7}	1.9×10^{-12}
Y-90	5.7×10^{-6}	3.4×10^{-11}	-	3.2×10^{-7}	1.9×10^{-11}
Y-91	4.3×10^{-6}	2.6×10^{-11}	-	5.5×10^{-6}	3.2×10^{-11}
Zr-95	9.4×10^{-6}	4.1×10^{-11}	-	1.1×10^{-5}	4.1×10^{-11}
Nb-95	2.0×10^{-5}	8.9×10^{-11}	-	2.0×10^{-5}	7.6×10^{-11}
Nb-95m	7.0×10^{-8}	3.0×10^{-13}	-	8.2×10^{-8}	3.0×10^{-13}
Tc-99	8.6×10^{-10}	5.1×10^{-15}	-	5.1×10^{-11}	3.0×10^{-16}
Ru-103	3.3×10^{-7}	2.0×10^{-12}	-	1.8×10^{-6}	1.0×10^{-11}
Ru-106	7.9×10^{-6}	4.7×10^{-11}	-	1.1×10^{-6}	6.5×10^{-12}
Rh-103m	3.0×10^{-7}	1.8×10^{-12}	-	1.6×10^{-6}	9.1×10^{-12}
Ag-110m	7.7×10^{-9}	4.6×10^{-14}	-	4.3×10^{-10}	2.5×10^{-15}
In-113m	2.6×10^{-7}	-	-	1.2×10^{-7}	-
Sn-113	2.6×10^{-7}	-	-	1.2×10^{-7}	-
Sn-119m	4.5×10^{-6}	2.6×10^{-14}	-	8.7×10^{-7}	7.1×10^{-15}
Sn-123	1.6×10^{-7}	2.9×10^{-13}	-	6.8×10^{-8}	1.3×10^{-13}
Cd-113m	1.2×10^{-9}	7.2×10^{-15}	-	1.5×10^{-10}	8.8×10^{-16}
Cd-115m	5.1×10^{-10}	3.0×10^{-15}	-	3.8×10^{-9}	2.2×10^{-14}
Sb-124	3.6×10^{-10}	2.2×10^{-15}	-	1.8×10^{-10}	1.0×10^{-15}
Sb-125	2.1×10^{-6}	2.6×10^{-12}	-	2.3×10^{-7}	3.4×10^{-13}
Te-125m	5.0×10^{-7}	6.4×10^{-13}	-	5.0×10^{-8}	7.3×10^{-14}
Te-127	1.3×10^{-7}	7.9×10^{-13}	-	6.2×10^{-8}	3.6×10^{-13}
Te-127m	1.4×10^{-7}	8.1×10^{-13}	-	6.3×10^{-8}	8.1×10^{-13}
Te-129	2.3×10^{-9}	1.4×10^{-14}	-	2.3×10^{-2}	1.4×10^{-13}
Te-129m	3.6×10^{-9}	2.1×10^{-14}	-	6.7×10^{-8}	2.2×10^{-13}
I-129	2.3×10^{-2}	-	-	2.6×10^{-2}	-
I-131	2.9×10^{-8}	-	-	1.5	-
Cs-134	2.1×10^{-6}	1.2×10^{-11}	-	3.1×10^{-8}	1.8×10^{-13}
Cs-137	6.2×10^{-6}	3.7×10^{-11}	-	3.6×10^{-7}	2.1×10^{-12}
Ba-140	-	-	-	6.4×10^{-9}	3.7×10^{-14}

TABLE H.17 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)				
	Tritium Production			Plutonium Production	
	Fuel Processing	Uranium Processing	Target Processing	Fuel and Target Processing	Uranium Processing
La-140	-	-	-	7.4×10^{-9}	4.3×10^{-14}
Ce-141	1.5×10^{-7}	8.6×10^{-13}	-	1.6×10^{-6}	9.1×10^{-12}
Ce-144	6.4×10^{-5}	3.8×10^{-10}	-	7.3×10^{-6}	4.2×10^{-11}
Pr-143	-	-	-	1.1×10^{-8}	6.4×10^{-14}
Pr-144	6.4×10^{-5}	3.8×10^{-10}	-	7.3×10^{-6}	4.2×10^{-11}
Pr-144m	7.7×10^{-7}	4.6×10^{-12}	-	8.8×10^{-8}	5.1×10^{-13}
Nd-147	-	-	-	5.4×10^{-10}	3.1×10^{-15}
Pm-147	1.5×10^{-5}	9.1×10^{-11}	-	1.3×10^{-6}	7.6×10^{-12}
Pm-148	4.4×10^{-10}	2.6×10^{-15}	-	4.4×10^{-10}	2.6×10^{-15}
Pm-148m	7.8×10^{-9}	4.6×10^{-14}	-	7.9×10^{-9}	4.6×10^{-14}
Sm-151	4.3×10^{-8}	2.6×10^{-13}	-	6.4×10^{-9}	3.7×10^{-14}
Eu-152	4.9×10^{-10}	2.9×10^{-15}	-	-	-
Eu-154	1.2×10^{-7}	7.1×10^{-13}	-	1.9×10^{-9}	1.1×10^{-14}
Eu-155	1.3×10^{-7}	7.9×10^{-13}	-	1.5×10^{-8}	8.9×10^{-14}
Eu-156	-	-	-	2.3×10^{-10}	1.3×10^{-15}
W-185	-	-	-	5.9×10^{-10}	-
U-234	4.8×10^{-10}	1.8×10^{-9}	-	8.1×10^{-11}	2.9×10^{-10}
U-235	1.4×10^{-11}	5.3×10^{-11}	-	2.7×10^{-12}	9.6×10^{-12}
U-236	2.7×10^{-11}	9.8×10^{-11}	-	2.5×10^{-12}	8.9×10^{-12}
U-238	2.6×10^{-11}	9.7×10^{-11}	-	2.9×10^{-11}	1.0×10^{-10}
Np-237	8.9×10^{-12}	5.3×10^{-17}	-	8.2×10^{-13}	4.7×10^{-18}
Pu-238	1.8×10^{-8}	1.1×10^{-13}	-	7.6×10^{-10}	4.4×10^{-15}
Pu-239	2.5×10^{-8}	1.5×10^{-13}	-	1.1×10^{-8}	6.2×10^{-14}
Pu-240	1.2×10^{-8}	7.4×10^{-14}	-	3.1×10^{-9}	1.8×10^{-14}
Pu-241	2.4×10^{-6}	1.4×10^{-11}	-	2.6×10^{-7}	1.5×10^{-12}
Pu-242	5.7×10^{-12}	3.4×10^{-17}	-	3.0×10^{-13}	1.7×10^{-18}
Am-241	4.6×10^{-9}	2.8×10^{-14}	-	2.3×10^{-10}	1.4×10^{-15}

Source: WHC 1990b.

TABLE H.18 Estimated Annual Atmospheric Releases from MHTGR Fuel Processing and Target Processing Facilities at Hanford

Radionuclide	Atmospheric Releases (Ci/yr)				
	Tritium Production			Plutonium Production	
	Fuel Processing	Uranium Processing	Target Processing	Fuel and Target Processing	Uranium Processing
H-3	4.5	-	5.0×10^3	4.9	-
C-14	3.2	-	-	1.6×10^2	-
Kr-85	3.4×10^5	-	-	4.2×10^5	-
Rb-86	3.7×10^{-11}	2.2×10^{-16}	-	8.1×10^{-10}	2.4×10^{-16}
Sr-89	1.2×10^{-5}	7.0×10^{-11}	-	1.8×10^{-4}	7.5×10^{-11}
Sr-90	1.7×10^{-4}	1.0×10^{-9}	-	1.4×10^{-4}	1.0×10^{-9}
Y-90	1.7×10^{-4}	1.0×10^{-9}	-	1.4×10^{-4}	1.0×10^{-9}
Y-91	2.9×10^{-5}	1.8×10^{-10}	-	2.9×10^{-4}	1.9×10^{-10}
Zr-95	5.1×10^{-5}	3.1×10^{-10}	-	3.7×10^{-4}	3.1×10^{-10}
Nb-95	1.1×10^{-4}	6.6×10^{-10}	-	7.1×10^{-4}	6.8×10^{-10}
Nb-95m	3.7×10^{-7}	2.3×10^{-12}	-	2.8×10^{-6}	2.3×10^{-12}
Tc-99	1.4×10^{-8}	8.8×10^{-14}	-	1.3×10^{-8}	8.8×10^{-14}
Ru-103	1.3×10^{-4}	1.8×10^{-11}	-	1.2×10^{-2}	2.2×10^{-11}
Ru-106	7.3×10^{-3}	9.9×10^{-10}	-	3.0×10^{-2}	1.0×10^{-9}
Rh-103m	2.7×10^{-6}	1.6×10^{-11}	-	1.2×10^{-4}	2.0×10^{-11}
Ag-110m	2.5×10^{-8}	1.5×10^{-13}	-	2.1×10^{-8}	1.5×10^{-13}
Sn-123	4.0×10^{-7}	2.4×10^{-12}	-	2.1×10^{-6}	2.5×10^{-12}
Sb-125	8.1×10^{-6}	4.9×10^{-11}	-	1.1×10^{-5}	4.9×10^{-11}
Te-125m	2.0×10^{-6}	1.2×10^{-11}	-	2.6×10^{-6}	1.3×10^{-11}
Te-127	1.2×10^{-6}	7.4×10^{-12}	-	6.3×10^{-6}	7.6×10^{-12}
Te-127m	1.3×10^{-6}	7.6×10^{-12}	-	7.4×10^{-6}	7.8×10^{-12}
Te-129	2.0×10^{-8}	1.2×10^{-13}	-	1.7×10^{-6}	1.7×10^{-13}
Te-129m	3.0×10^{-8}	1.8×10^{-13}	-	2.6×10^{-6}	2.6×10^{-13}
I-129	2.9×10^{-2}	-	-	3.4×10^{-2}	-
I-131	2.0×10^{-8}	-	-	7.9×10^{-1}	-
Xe-131m	4.7×10^{-4}	-	-	7.4×10^1	-
Cs-134	5.9×10^{-4}	3.6×10^{-9}	-	4.5×10^{-4}	3.6×10^{-9}
Cs-136	2.4×10^{-12}	1.4×10^{-17}	-	5.1×10^{-9}	1.6×10^{-16}
Cs-137	1.8×10^{-4}	1.1×10^{-9}	-	1.6×10^{-4}	1.1×10^{-9}
Ba-140	6.5×10^{-12}	4.0×10^{-17}	-	2.2×10^{-7}	6.4×10^{-15}
La-140	7.5×10^{-12}	4.5×10^{-17}	-	2.5×10^{-7}	7.4×10^{-15}
Ce-141	1.0×10^{-6}	6.3×10^{-12}	-	6.7×10^{-5}	8.2×10^{-12}
Ce-144	8.7×10^{-4}	5.3×10^{-9}	-	1.0×10^{-3}	5.3×10^{-9}
Pr-143	1.9×10^{-11}	1.2×10^{-16}	-	4.7×10^{-6}	1.1×10^{-14}
Pr-144	8.7×10^{-4}	5.3×10^{-9}	-	1.0×10^{-3}	5.3×10^{-9}
Pr-144m	1.0×10^{-5}	6.3×10^{-11}	-	1.2×10^{-5}	6.3×10^{-11}
Nd-147	5.6×10^{-14}	6.7×10^{-19}	-	1.8×10^{-8}	5.1×10^{-16}
Pm-148m	2.5×10^{-8}	1.5×10^{-13}	-	5.9×10^{-7}	1.7×10^{-13}
Sm-151	5.5×10^{-7}	3.3×10^{-12}	-	1.0×10^{-6}	3.4×10^{-12}

TABLE H.18 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)				
	Tritium Production			Plutonium Production	
	Fuel Processing	Uranium Processing	Target Processing	Fuel and Target Processing	Uranium Processing
Eu-154	2.1×10^{-5}	1.3×10^{-10}	-	1.6×10^{-5}	1.3×10^{-10}
Eu-155	1.4×10^{-5}	8.4×10^{-11}	-	1.3×10^{-5}	8.4×10^{-11}
U-234	5.2×10^{-9}	2.0×10^{-8}	-	3.9×10^{-9}	2.0×10^{-8}
U-235	1.6×10^{-10}	6.2×10^{-10}	-	1.2×10^{-10}	6.2×10^{-10}
U-236	2.2×10^{-10}	8.1×10^{-10}	-	1.6×10^{-10}	8.1×10^{-10}
U-238	1.1×10^{-12}	4.2×10^{-12}	-	8.3×10^{-13}	4.2×10^{-12}
Np-237	1.7×10^{-9}	1.0×10^{-14}	-	1.3×10^{-9}	1.0×10^{-14}
Pu-238	4.0×10^{-5}	2.4×10^{-10}	-	3.0×10^{-5}	2.4×10^{-10}
Pu-239	2.3×10^{-8}	1.4×10^{-13}	-	1.7×10^{-8}	1.4×10^{-13}
Pu-240	2.6×10^{-8}	1.6×10^{-13}	-	2.0×10^{-8}	1.6×10^{-13}
Pu-241	1.8×10^{-5}	1.1×10^{-10}	-	1.3×10^{-5}	1.1×10^{-10}
Am-241	3.1×10^{-8}	1.9×10^{-13}	-	2.3×10^{-8}	1.9×10^{-13}
Am-243	1.4×10^{-8}	8.4×10^{-14}	-	1.0×10^{-8}	8.4×10^{-14}
Cm-242	1.9×10^{-6}	1.1×10^{-11}	-	1.4×10^{-6}	1.1×10^{-11}
Cm-244	6.1×10^{-6}	3.7×10^{-11}	-	4.6×10^{-6}	3.7×10^{-11}

Source: WHC 1989b.

TABLE H.19 Estimated Annual Atmospheric Releases from HWR Fuel Processing and Target Processing Facilities at INEL

Radionuclide	Atmospheric Releases (Ci/yr)				
	Headend Fuel Processing	Fuel Processing	Tritium-Target Processing	Plutonium-Target Processing	Plutonium Oxide-to-Metal Conversion
H-3	1.1×10^3	1.0×10^2	5.0×10^3	2.1×10^2	-
Kr-85	2.7×10^5	2.7×10^3	-	4.3×10^4	-
Rb-86	3.2×10^{-13}	1.1×10^{-9}	-	-	-
Sr-89	2.6×10^{-6}	9.0×10^{-3}	-	9.4×10^{-7}	-
Sr-90	6.8×10^{-6}	2.4×10^{-2}	-	1.0×10^{-6}	-
Y-90	6.8×10^{-6}	2.4×10^{-2}	-	1.0×10^{-6}	-
Y-91	5.7×10^{-6}	2.0×10^{-2}	-	2.0×10^{-6}	-
Zr-95	9.4×10^{-6}	3.3×10^{-2}	-	3.5×10^{-6}	-
Nb-95	1.9×10^{-5}	6.7×10^{-2}	-	7.3×10^{-6}	-
Nb-95m	1.1×10^{-7}	3.9×10^{-4}	-	4.2×10^{-8}	-
Tc-99	9.4×10^{-10}	3.3×10^{-6}	-	1.9×10^{-10}	-
Ru-103	4.1×10^{-7}	1.4×10^{-3}	-	3.3×10^{-7}	-
Ru-106	6.0×10^{-6}	2.1×10^{-2}	-	5.6×10^{-6}	-
Rh-103m	4.1×10^{-7}	1.4×10^{-3}	-	3.3×10^{-7}	-
Rh-106	6.0×10^{-6}	2.1×10^{-2}	-	5.6×10^{-6}	-
Ag-110m	1.6×10^{-7}	5.4×10^{-4}	-	5.6×10^{-8}	-
Sn-123	7.3×10^{-8}	2.5×10^{-4}	-	5.6×10^{-8}	-
Sb-125	2.3×10^{-7}	7.9×10^{-4}	-	1.2×10^{-7}	-
Te-125m	5.5×10^{-8}	1.9×10^{-4}	-	2.8×10^{-8}	-
Te-127	1.1×10^{-7}	3.7×10^{-4}	-	7.0×10^{-8}	-
Te-127m	1.1×10^{-7}	3.8×10^{-4}	-	7.0×10^{-8}	-
Te-129	4.2×10^{-9}	1.5×10^{-5}	-	3.8×10^{-9}	-
Te-129m	6.5×10^{-9}	2.2×10^{-5}	-	5.9×10^{-9}	-
I-129	1.2×10^{-3}	3.7×10^{-3}	-	3.6×10^{-4}	-
Cs-134	5.2×10^{-6}	1.8×10^{-2}	-	1.4×10^{-7}	-
Cs-136	8.6×10^{-15}	3.0×10^{-11}	-	-	-
Cs-137	6.8×10^{-6}	2.4×10^{-2}	-	1.3×10^{-6}	-
Ba-137m	6.5×10^{-6}	2.2×10^{-2}	-	1.3×10^{-6}	-
Ba-140	1.2×10^{-12}	4.3×10^{-9}	-	1.1×10^{-12}	-
La-140	1.2×10^{-12}	4.3×10^{-9}	-	1.1×10^{-12}	-
Ce-141	1.9×10^{-7}	6.7×10^{-4}	-	1.1×10^{-7}	-
Ce-144	7.5×10^{-5}	2.6×10^{-1}	-	1.6×10^{-5}	-
Pr-143	3.9×10^{-12}	1.3×10^{-8}	-	3.3×10^{-12}	-
Pr-144	7.5×10^{-5}	2.6×10^{-1}	-	1.6×10^{-5}	-
Pr-144m	9.1×10^{-7}	3.1×10^{-3}	-	2.0×10^{-7}	-
Pm-147	1.0×10^{-5}	3.5×10^{-2}	-	3.8×10^{-6}	-
Pm-148m	8.8×10^{-9}	3.0×10^{-5}	-	2.3×10^{-9}	-
Sm-151	2.1×10^{-8}	7.4×10^{-5}	-	2.2×10^{-8}	-
Eu-154	2.6×10^{-7}	9.0×10^{-4}	-	1.2×10^{-8}	-
Eu-155	1.4×10^{-7}	4.9×10^{-4}	-	7.3×10^{-8}	-

TABLE H.19 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)				
	Headend Fuel Processing	Fuel Processing	Tritium- Target Processing	Plutonium- Target Processing	Plutonium Oxide-to- Metal Conversion
Np-237	3.9×10^{-11}	1.3×10^{-7}	-	2.7×10^{-12}	-
Pu-238	1.6×10^{-7}	5.6×10^{-4}	-	1.6×10^{-9}	1.3×10^{-10}
Pu-239	1.1×10^{-9}	3.8×10^{-6}	-	7.0×10^{-8}	3.8×10^{-10}
Pu-240	9.2×10^{-10}	3.2×10^{-6}	-	1.3×10^{-8}	3.3×10^{-10}
Pu-241	2.9×10^{-7}	1.0×10^{-3}	-	8.7×10^{-7}	1.3×10^{-8}
Pu-242	-	-	-	-	9.2×10^{-14}
Am-241	5.4×10^{-10}	1.9×10^{-6}	-	1.5×10^{-9}	2.7×10^{-10}
Am-243	-	2.0×10^{-11}	-	-	-
Cm-242	3.9×10^{-9}	1.3×10^{-5}	-	4.5×10^{-10}	-
Cm-244	2.8×10^{-12}	9.5×10^{-9}	-	-	-

Source: EG&G 1989c for plutonium oxide-to-metal conversion facility; EG&G 1989a for other facilities.

TABLE H.20 Estimated Annual Atmospheric Releases from LWR Fuel Processing and Target Processing Facilities at INEL

Radionuclide	Atmospheric Releases (Ci/yr)				
	Tritium Production ^a		Plutonium Production		
	Fuel Processing	Uranium Processing	Fuel and Target Processing	Uranium Processing	Plutonium Oxide-to-Metal Conversion
H-3	3.4×10^2	1.4	2.8×10^2	1.1	-
C-14	4.3	-	1.4×10^1	-	-
Cr-51	1.0×10^{-8}	-	1.5×10^{-6}	-	-
Mn-54	9.0×10^{-7}	-	5.9×10^{-7}	-	-
Fe-55	6.7×10^{-6}	-	2.9×10^{-6}	-	-
Fe-59	4.8×10^{-9}	-	9.0×10^{-8}	-	-
Co-58	3.5×10^{-7}	-	2.5×10^{-6}	-	-
Co-60	3.0×10^{-6}	-	1.1×10^{-6}	-	-
Ni-59	3.5×10^{-9}	-	1.3×10^{-9}	-	-
Ni-63	4.4×10^{-7}	-	1.7×10^{-7}	-	-
Kr-85	3.7×10^5	-	2.6×10^5	-	-
Sr-89	1.0×10^{-6}	6.2×10^{-12}	2.6×10^{-6}	1.5×10^{-11}	-
Sr-90	3.2×10^{-6}	1.9×10^{-11}	2.4×10^{-7}	1.4×10^{-12}	-
Y-90	3.2×10^{-6}	1.9×10^{-11}	2.4×10^{-7}	1.4×10^{-12}	-
Y-91	2.4×10^{-6}	1.5×10^{-11}	4.1×10^{-6}	2.4×10^{-11}	-
Zr-95	5.3×10^{-6}	2.3×10^{-11}	8.3×10^{-6}	3.1×10^{-11}	-
Nb-95	1.1×10^{-5}	5.0×10^{-11}	1.5×10^{-5}	5.7×10^{-11}	-
Nb-95m	3.9×10^{-8}	1.7×10^{-13}	6.2×10^{-8}	2.3×10^{-13}	-
Tc-99	4.8×10^{-10}	2.9×10^{-15}	3.8×10^{-11}	2.3×10^{-16}	-
Ru-103	1.8×10^{-7}	1.1×10^{-12}	1.3×10^{-6}	7.5×10^{-12}	-
Ru-106	4.4×10^{-6}	2.6×10^{-11}	8.3×10^{-7}	4.9×10^{-12}	-
Rh-103m	1.7×10^{-7}	1.0×10^{-12}	1.2×10^{-6}	6.8×10^{-12}	-
Rh-106	4.4×10^{-6}	2.6×10^{-11}	8.3×10^{-7}	4.9×10^{-12}	-
Ag-110m	4.3×10^{-9}	2.6×10^{-14}	3.2×10^{-10}	1.9×10^{-15}	-
Cd-113m	6.7×10^{-10}	4.0×10^{-15}	1.1×10^{-10}	6.6×10^{-16}	-
Cd-115m	2.9×10^{-10}	1.7×10^{-15}	2.9×10^{-9}	1.7×10^{-14}	-
In-113m	1.5×10^{-7}	-	9.0×10^{-8}	-	-
Sn-113	1.5×10^{-7}	-	9.0×10^{-8}	-	-
Sn-119m	2.5×10^{-6}	1.5×10^{-14}	6.5×10^{-7}	5.3×10^{-15}	-
Sn-123	9.0×10^{-8}	1.6×10^{-13}	5.1×10^{-8}	9.8×10^{-14}	-
Sb-124	2.0×10^{-10}	1.2×10^{-15}	1.3×10^{-10}	7.5×10^{-16}	-
Sb-125	1.2×10^{-6}	1.5×10^{-12}	1.7×10^{-7}	2.6×10^{-13}	-
Te-125m	2.8×10^{-7}	3.6×10^{-13}	3.7×10^{-8}	5.5×10^{-14}	-
Te-127m	7.8×10^{-8}	4.5×10^{-13}	4.7×10^{-8}	2.7×10^{-13}	-
Te-127	7.3×10^{-8}	4.4×10^{-13}	4.6×10^{-8}	2.7×10^{-13}	-
Te-129	1.3×10^{-9}	7.8×10^{-15}	1.7×10^{-2}	1.1×10^{-13}	-
Te-129m	2.0×10^{-9}	1.2×10^{-14}	5.0×10^{-8}	1.7×10^{-13}	-
I-129	1.3×10^{-2}	-	1.9×10^{-2}	-	-
I-131	1.6×10^{-8}	-	1.1	-	-

TABLE H.20 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)				
	Tritium Production ^a		Plutonium Production		
	Fuel Processing	Uranium Processing	Fuel and Target Processing	Uranium Processing	Plutonium Oxide-to-Metal Conversion
Cs-134	1.2×10^{-6}	6.7×10^{-12}	2.3×10^{-8}	1.4×10^{-13}	-
Cs-137	3.5×10^{-6}	2.1×10^{-11}	2.7×10^{-7}	1.6×10^{-12}	-
Ba-137m	3.3×10^{-6}	2.0×10^{-11}	2.5×10^{-7}	1.5×10^{-12}	-
Ba-140	-	-	4.8×10^{-9}	2.8×10^{-14}	-
La-140	-	-	5.6×10^{-9}	3.2×10^{-14}	-
Ce-141	8.4×10^{-8}	4.8×10^{-13}	1.2×10^{-6}	6.8×10^{-12}	-
Ce-144	3.6×10^{-5}	2.1×10^{-10}	5.5×10^{-6}	3.2×10^{-11}	-
Pr-143	-	-	8.2×10^{-9}	4.8×10^{-14}	-
Pr-144	3.6×10^{-5}	2.1×10^{-10}	5.5×10^{-6}	3.2×10^{-11}	-
Pr-144m	4.3×10^{-7}	2.6×10^{-12}	6.6×10^{-8}	3.8×10^{-13}	-
Nd-147	-	-	4.0×10^{-10}	2.3×10^{-15}	-
Pm-147	8.4×10^{-6}	5.1×10^{-11}	9.7×10^{-7}	5.7×10^{-12}	-
Pm-148	2.5×10^{-10}	1.5×10^{-15}	3.3×10^{-10}	2.0×10^{-15}	-
Pm-148m	4.4×10^{-9}	2.6×10^{-14}	5.9×10^{-9}	3.4×10^{-14}	-
Sm-151	2.4×10^{-8}	1.5×10^{-13}	4.8×10^{-9}	2.8×10^{-14}	-
Eu-152	2.7×10^{-10}	1.6×10^{-15}	-	-	-
Eu-154	6.7×10^{-8}	4.0×10^{-13}	1.4×10^{-9}	8.3×10^{-15}	-
Eu-155	7.3×10^{-8}	4.4×10^{-13}	1.1×10^{-8}	6.7×10^{-14}	-
Eu-156	-	-	1.7×10^{-10}	9.8×10^{-16}	-
Hf-175	-	-	7.3×10^{-10}	-	-
Hf-181	-	-	3.7×10^{-9}	-	-
W-185	-	-	4.4×10^{-10}	-	-
U-234	2.7×10^{-10}	1.0×10^{-9}	6.1×10^{-11}	2.2×10^{-10}	-
U-235	7.8×10^{-12}	3.0×10^{-11}	2.0×10^{-12}	7.2×10^{-12}	-
U-236	1.5×10^{-11}	5.5×10^{-11}	1.9×10^{-12}	6.7×10^{-12}	-
U-238	1.5×10^{-11}	5.4×10^{-11}	2.2×10^{-11}	7.5×10^{-11}	-
Np-237	5.0×10^{-12}	3.0×10^{-17}	6.2×10^{-13}	3.5×10^{-18}	-
Pu-238	1.0×10^{-8}	6.2×10^{-14}	5.7×10^{-10}	3.3×10^{-15}	1.3×10^{-10}
Pu-239	1.4×10^{-14}	8.4×10^{-14}	8.2×10^{-9}	4.7×10^{-14}	3.8×10^{-10}
Pu-240	6.7×10^{-9}	4.1×10^{-14}	2.3×10^{-9}	1.3×10^{-14}	3.3×10^{-10}
Pu-241	1.3×10^{-6}	7.8×10^{-12}	2.0×10^{-7}	1.1×10^{-12}	1.3×10^{-8}
Pu-242	3.2×10^{-12}	1.9×10^{-17}	2.3×10^{-13}	1.3×10^{-18}	9.2×10^{-14}
Am-241	2.6×10^{-9}	1.6×10^{-14}	1.7×10^{-10}	1.1×10^{-15}	2.7×10^{-10}

^aThe release from the tritium-target processing facilities is estimated to be 5×10^3 Ci/yr.

Sources: EG&G 1989b for LWR adaptation; EG&G 1989c for the plutonium oxide-to-metal conversion facility source terms, WHC 1990b for other facility source terms.

TABLE H.21 Estimated Annual Atmospheric Releases from MHTGR Fuel Processing and Target Processing Facilities at INEL

Radionuclide	Atmospheric Releases (Ci/yr)				
	Headend Fuel Processing	Fuel Processing	Tritium-Target Processing	Plutonium-Target Processing	Plutonium Oxide-to-Metal Conversion
H-3	4.5×10^2	-	5.0×10^3	4.2×10^1	-
C-14	2.4×10^{-1}	-	-	1.4×10^1	-
Kr-85	3.4×10^5	-	-	7.8×10^4	-
Rb-86	1.8×10^{-11}	6.2×10^{-9}	-	-	-
Sr-89	5.7×10^{-6}	2.0×10^{-3}	-	1.4×10^{-5}	-
Sr-90	8.2×10^{-5}	2.8×10^{-2}	-	1.8×10^{-5}	-
Y-90	8.2×10^{-5}	2.8×10^{-2}	-	1.8×10^{-5}	-
Y-91	1.4×10^{-5}	5.0×10^{-3}	-	3.1×10^{-5}	-
Zr-95	2.5×10^{-5}	8.5×10^{-3}	-	4.6×10^{-5}	-
Nb-95	5.4×10^{-5}	1.9×10^{-2}	-	9.9×10^{-5}	-
Nb-95m	1.8×10^{-7}	6.3×10^{-5}	-	3.4×10^{-7}	-
Tc-99	7.1×10^{-9}	2.5×10^{-6}	-	2.7×10^{-9}	-
Ru-103	1.3×10^{-4}	5.1×10^{-4}	-	4.7×10^{-4}	-
Ru-106	7.2×10^{-3}	2.8×10^{-2}	-	1.7×10^{-2}	-
Rh-103m	1.3×10^{-6}	4.6×10^{-4}	-	4.7×10^{-6}	-
Rh-106	8.0×10^{-5}	2.8×10^{-2}	-	1.8×10^{-4}	-
Ag-110m	1.2×10^{-8}	4.3×10^{-6}	-	1.7×10^{-9}	-
Sn-123	2.0×10^{-7}	6.8×10^{-5}	-	6.8×10^{-7}	-
Sb-125	4.0×10^{-6}	1.4×10^{-3}	-	4.4×10^{-6}	-
Te-125m	9.7×10^{-7}	3.4×10^{-4}	-	1.1×10^{-6}	-
Te-127m	6.1×10^{-7}	2.1×10^{-4}	-	1.7×10^{-6}	-
Te-127	6.0×10^{-7}	2.1×10^{-4}	-	1.7×10^{-6}	-
Te-129m	1.5×10^{-8}	5.1×10^{-6}	-	6.0×10^{-8}	-
Te-129	9.6×10^{-9}	3.3×10^{-6}	-	3.9×10^{-8}	-
I-129	6.7×10^{-3}	1.7×10^{-3}	-	3.0×10^{-3}	-
I-131	5.0×10^{-9}	1.2×10^{-9}	-	2.0×10^{-8}	-
Xe-131m	4.7×10^{-4}	-	-	1.5×10^{-3}	-
Cs-134	2.9×10^{-4}	1.0×10^{-1}	-	6.7×10^{-6}	-
Cs-136	1.2×10^{-12}	4.0×10^{-10}	-	-	-
Cs-137	8.9×10^{-5}	3.1×10^{-2}	-	2.0×10^{-5}	-
Ba-137m	8.4×10^{-5}	2.9×10^{-2}	-	1.9×10^{-5}	-
Ba-140	3.2×10^{-12}	1.1×10^{-9}	-	1.1×10^{-11}	-
La-140	3.7×10^{-12}	1.3×10^{-9}	-	1.3×10^{-11}	-
Ce-141	5.1×10^{-7}	1.8×10^{-4}	-	1.3×10^{-6}	-
Pr-143	9.5×10^{-12}	3.3×10^{-9}	-	3.5×10^{-11}	-
Pr-144	4.3×10^{-4}	1.5×10^{-1}	-	2.5×10^{-4}	-
Pr-144m	5.1×10^{-6}	1.8×10^{-3}	-	3.0×10^{-6}	-
Ce-144	4.3×10^{-4}	1.5×10^{-1}	-	2.5×10^{-4}	-
Nd-147	-	1.9×10^{-11}	-	-	-
Pm-148m	1.2×10^{-8}	4.2×10^{-6}	-	2.7×10^{-8}	-
Sm-151	2.7×10^{-7}	9.4×10^{-5}	-	6.1×10^{-7}	-

TABLE H.21 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)				
	Headend Fuel Processing	Fuel Processing	Tritium- Target Processing	Plutonium- Target Processing	Plutonium Oxide-to- Metal Conversion
Eu-154	1.0×10^{-5}	3.5×10^{-3}	-	3.5×10^{-7}	-
Eu-155	6.8×10^{-6}	2.3×10^{-3}	-	2.3×10^{-6}	-
Np-237	8.3×10^{-10}	2.9×10^{-7}	-	2.8×10^{-11}	-
Np-239	6.8×10^{-9}	2.4×10^{-6}	-	4.3×10^{-11}	-
Pu-238	2.0×10^{-5}	6.8×10^{-3}	-	4.5×10^{-8}	1.3×10^{-10}
Pu-239	1.1×10^{-8}	3.9×10^{-6}	-	1.5×10^{-6}	3.8×10^{-10}
Pu-240	1.3×10^{-8}	4.4×10^{-6}	-	7.1×10^{-7}	3.3×10^{-10}
Pu-241	8.8×10^{-6}	3.1×10^{-3}	-	6.4×10^{-5}	1.3×10^{-8}
Pu-242	2.6×10^{-10}	8.9×10^{-8}	-	5.4×10^{-11}	9.2×10^{-14}
Am-241	1.5×10^{-8}	5.4×10^{-6}	-	1.1×10^{-7}	2.7×10^{-10}
Am-243	6.8×10^{-9}	2.4×10^{-6}	-	4.3×10^{-11}	-
Cm-242	9.1×10^{-7}	3.1×10^{-4}	-	6.5×10^{-8}	-
Cm-244	3.0×10^{-6}	1.0×10^{-3}	-	3.1×10^{-10}	-

Source: EG&G 1989c.

TABLE H.22 Estimated Annual Incremental Releases from HWR Fuel and Target Processing Facilities at SRS^a

Radionuclide	Incremental Releases (Ci/yr)			
	Fuel Processing		Plutonium-Target Processing	
	Atmospheric	Liquid	Atmospheric	Liquid
H-3	3.8×10^3	2.8×10^3	3.8×10^3	2.7×10^3
C-14	5.6	-	5.6	-
Co-60	8.0×10^{-7}	-	8.0×10^{-7}	-
Se-75	3.0×10^{-6}	-	3.0×10^{-7}	-
Kr-85	1.4×10^5	-	1.4×10^5	-
Sr-90	2.8×10^{-4}	6.7×10^{-3}	2.8×10^{-4}	6.7×10^{-3}
Zr-95	3.4×10^{-3}	-	3.4×10^{-3}	-
Nb-95	5.8×10^{-3}	-	5.8×10^{-3}	-
Ru-103	1.5×10^{-3}	-	1.5×10^{-3}	-
Ru-106	1.0×10^{-2}	-	1.0×10^{-2}	-
Xe-131m	5.3×10^{-1}	-	5.3×10^{-1}	-
Xe-133	6.4×10^{-4}	-	6.4×10^{-4}	-
I-129	1.5×10^{-2}	4.3×10^{-4}	1.5×10^{-2}	4.4×10^{-4}
I-131	6.3×10^{-3}	-	6.3×10^{-3}	-
Cs-134	7.2×10^{-5}	-	7.2×10^{-5}	-
Cs-137	8.1×10^{-4}	3.7×10^{-5}	8.1×10^{-4}	3.7×10^{-5}
Ce-141	3.7×10^{-4}	-	3.7×10^{-4}	-
Ce-144	6.2×10^{-3}	-	6.2×10^{-3}	-
Os-185	7.2×10^{-5}	-	7.2×10^{-5}	-
U-235	4.0×10^{-4}	-	4.0×10^{-4}	-
Pu-238	2.6×10^{-4}	-	2.6×10^{-4}	-
Pu-239	8.0×10^{-5}	4.3×10^{-5}	8.0×10^{-5}	4.2×10^{-5}
Am-241	5.8×10^{-5}	-	5.8×10^{-5}	-
Cm-242	2.7×10^{-5}	-	2.7×10^{-5}	-

^aTritium-target processing would add 5×10^3 Ci/yr of tritium.

Source: WSRC 1991a.

TABLE H.23 Estimated Annual Incremental Atmospheric Releases from LWR Fuel and Target Processing Facilities at SRS

Radionuclide	Incremental Atmospheric Releases (Ci/yr)		
	Tritium Production		Plutonium Production
	Fuel Processing	Target Processing	Fuel and Target Processing
H-3	6.1×10^2	5.0×10^3	3.7×10^2
C-14	7.6	-	1.9×10^1
Cr-51	1.8×10^{-8}	-	2.0×10^{-6}
Mn-54	1.6×10^{-6}	-	7.9×10^{-7}
Fe-55	1.2×10^{-5}	-	3.8×10^{-6}
Fe-59	8.5×10^{-9}	-	1.2×10^{-7}
Co-58	6.2×10^{-7}	-	3.3×10^{-6}
Co-60	5.3×10^{-6}	-	1.5×10^{-6}
Ni-59	6.2×10^{-9}	-	1.7×10^{-9}
Ni-63	7.9×10^{-7}	-	2.2×10^{-7}
Kr-85	6.6×10^5	-	2.2×10^{-7}
Sr-89	1.8×10^{-6}	-	3.5×10^{-6}
Sr-90	5.7×10^{-6}	-	3.2×10^{-7}
Y-90	5.7×10^{-6}	-	3.2×10^{-7}
Y-91	4.3×10^{-6}	-	5.5×10^{-6}
Zr-95	9.4×10^{-6}	-	1.1×10^{-5}
Nb-95	2.0×10^{-5}	-	2.0×10^{-5}
Nb-95m	7.0×10^{-8}	-	8.2×10^{-8}
Tc-99	8.6×10^{-10}	-	-
Ru-103	3.3×10^{-7}	-	1.8×10^{-6}
Ru-106	7.9×10^{-6}	-	1.1×10^{-6}
Rh-103m	3.0×10^{-7}	-	1.6×10^{-6}
Ag-110m	7.7×10^{-9}	-	4.3×10^{-10}
In-113m	2.6×10^{-7}	-	1.2×10^{-7}
Sn-113	2.6×10^{-7}	-	1.2×10^{-7}
Cd-113m	1.2×10^{-9}	-	1.5×10^{-10}
Cd-115m	5.1×10^{-10}	-	3.8×10^{-9}
Sn-119m	4.5×10^{-6}	-	8.7×10^{-7}
Sn-123	1.6×10^{-7}	-	6.8×10^{-8}
Sb-124	3.6×10^{-10}	-	1.8×10^{-10}
Sb-125	2.1×10^{-6}	-	2.3×10^{-7}
Te-125m	5.0×10^{-7}	-	5.0×10^{-8}
Te-127	1.3×10^{-7}	-	6.2×10^{-8}
Te-127m	1.4×10^{-7}	-	6.3×10^{-8}
Te-129	2.3×10^{-9}	-	2.3×10^{-2}
Te-129m	3.6×10^{-9}	-	6.7×10^{-8}
I-129	2.3×10^{-2}	-	2.6×10^{-2}
I-131	2.9×10^{-8}	-	1.5
Cs-134	2.1×10^{-6}	-	3.1×10^{-8}
Cs-137	6.2×10^{-6}	-	3.6×10^{-7}
Ba-140	-	-	6.4×10^{-9}

TABLE H.23 (Cont'd)

Radionuclide	Incremental Atmospheric Releases (Ci/yr)		
	Tritium Production		Plutonium Production
	Fuel Processing	Target Processing	Fuel and Target Processing
La-140	-	-	7.4×10^{-9}
Ce-141	1.5×10^{-7}	-	1.6×10^{-6}
Ce-144	6.4×10^{-5}	-	7.3×10^{-6}
Pr-143	-	-	1.1×10^{-8}
Pr-144	6.4×10^{-5}	-	7.3×10^{-6}
Pr-144m	7.7×10^{-7}	-	8.8×10^{-8}
Nd-147	-	-	5.4×10^{-10}
Pm-147	1.5×10^{-5}	-	1.3×10^{-6}
Pm-148	4.4×10^{-10}	-	4.4×10^{-10}
Pm-148m	7.8×10^{-9}	-	7.9×10^{-9}
Sm-151	4.3×10^{-8}	-	6.4×10^{-9}
Eu-152	4.9×10^{-10}	-	-
Eu-154	1.2×10^{-7}	-	1.9×10^{-9}
Eu-155	1.3×10^{-7}	-	1.5×10^{-8}
Eu-156	-	-	2.3×10^{-10}
W-185	-	-	5.9×10^{-10}
U-234	2.3×10^{-9}	-	3.7×10^{-10}
U-238	-	-	1.3×10^{-10}
Pu-238	1.8×10^{-8}	-	7.6×10^{-10}
Pu-239	2.5×10^{-8}	-	1.1×10^{-8}
Pu-240	1.2×10^{-8}	-	3.1×10^{-9}
Pu-241	2.4×10^{-6}	-	2.6×10^{-7}
Am-241	4.6×10^{-9}	-	2.3×10^{-10}

Sources: WSRC 1991b; WHC 1990b.

TABLE H.24 Estimated Annual Atmospheric Releases from
MHTGR Fuel Processing and Target Processing Facilities at SRS

Radionuclide	Atmospheric Releases (Ci/yr)			
	Headend Fuel Processing	Fuel Processing	Tritium- Target Processing	Plutonium- Target Processing
H-3	4.5×10^2	-	5.0×10^3	4.4×10^1
C-14	3.2	-	-	1.6×10^2
Kr-85	3.4×10^5	-	-	8.1×10^4
Rb-86	-	6.2×10^{-9}	-	7.8×10^{-10}
Sr-89	5.7×10^{-6}	2.0×10^{-3}	-	1.7×10^{-4}
Sr-90	8.2×10^{-5}	2.8×10^{-2}	-	1.9×10^{-5}
Y-90	8.2×10^{-5}	2.8×10^{-2}	-	1.9×10^{-2}
Y-91	1.4×10^{-5}	5.0×10^{-3}	-	2.7×10^{-4}
Zr-95	2.5×10^{-5}	8.5×10^{-3}	-	3.3×10^{-4}
Nb-95	5.4×10^{-5}	1.9×10^{-2}	-	6.3×10^{-4}
Nb-95m	1.8×10^{-7}	6.3×10^{-5}	-	2.5×10^{-6}
Tc-99	7.1×10^{-9}	2.5×10^{-6}	-	2.7×10^{-9}
Ru-103	1.3×10^{-4}	5.1×10^{-4}	-	1.2×10^{-2}
Ru-106	7.2×10^{-3}	2.8×10^{-2}	-	2.3×10^{-2}
Rh-103m	1.3×10^{-6}	4.6×10^{-4}	-	1.2×10^{-4}
Ag-110	1.2×10^{-8}	4.3×10^{-6}	-	2.9×10^{-9}
Sn-123	2.0×10^{-7}	6.8×10^{-5}	-	1.8×10^{-6}
Sb-125	4.0×10^{-6}	1.4×10^{-3}	-	5.0×10^{-6}
Te-125m	9.7×10^{-7}	3.4×10^{-4}	-	1.1×10^{-6}
Te-127	6.0×10^{-7}	2.1×10^{-4}	-	5.4×10^{-6}
Te-127m	6.1×10^{-7}	2.1×10^{-4}	-	6.5×10^{-6}
Te-129	9.6×10^{-9}	3.3×10^{-6}	-	1.7×10^{-6}
Te-129m	1.5×10^{-8}	5.1×10^{-6}	-	2.6×10^{-6}
Xe-131m	4.7×10^{-4}	-	-	7.4×10^1
Xe-133	-	-	-	6.8×10^{-3}
I-129	6.7×10^{-1}	1.7×10^{-3}	-	3.0×10^{-1}
I-131	5.0×10^{-7}	1.2×10^{-9}	-	1.4×10^{-1}
I-132	-	-	-	1.8×10^{-9}
Cs-134	2.9×10^{-4}	1.0×10^{-1}	-	7.9×10^{-6}
Cs-136	-	4.0×10^{-10}	-	5.1×10^{-9}
Cs-137	8.9×10^{-5}	3.1×10^{-2}	-	2.0×10^{-5}
Ba-140	-	1.1×10^{-9}	-	2.2×10^{-7}
La-140	-	1.3×10^{-9}	-	2.5×10^{-7}
Ce-141	5.1×10^{-7}	1.8×10^{-4}	-	6.6×10^{-5}
Ce-144	4.3×10^{-4}	1.5×10^{-1}	-	3.9×10^{-4}
Pr-143	-	3.3×10^{-9}	-	3.9×10^{-7}
Pr-144	4.3×10^{-4}	1.5×10^{-1}	-	3.9×10^{-4}
Pr-144m	5.1×10^{-6}	1.8×10^{-3}	-	4.7×10^{-6}
Nd-147	-	-	-	1.8×10^{-8}
Pm-148m	1.2×10^{-8}	4.2×10^{-6}	-	5.7×10^{-7}
Sm-151	2.7×10^{-7}	9.4×10^{-5}	-	6.1×10^{-7}

TABLE H.24 (Cont'd)

Radionuclide	Atmospheric Releases (Ci/yr)			
	Headend Fuel Processing	Fuel Processing	Tritium- Target Processing	Plutonium- Target Processing
Eu-154	1.0×10^{-5}	3.5×10^{-3}	-	3.6×10^{-7}
Eu-155	6.8×10^{-6}	2.3×10^{-3}	-	2.5×10^{-6}
Np-237	-	2.9×10^{-7}	-	-
Np-239	6.8×10^{-9}	2.4×10^{-6}	-	-
Pu-238	2.0×10^{-5}	6.8×10^{-3}	-	1.2×10^{-6}
Pu-239	1.1×10^{-8}	3.9×10^{-6}	-	1.1×10^{-3}
Pu-240	1.3×10^{-8}	1.3×10^{-9}	-	5.3×10^{-4}
Pu-241	8.8×10^{-6}	8.8×10^{-7}	-	-
Am-241	1.5×10^{-8}	1.5×10^{-9}	-	7.0×10^{-5}
Am-243	6.8×10^{-9}	6.8×10^{-10}	-	6.7×10^{-8}
Cm-242	9.1×10^{-7}	9.1×10^{-8}	-	1.1×10^{-3}
Cm-243	3.0×10^{-6}	3.0×10^{-7}	-	2.2×10^{-6}

Source: WSRC 1991c; EG&G 1989c.

TABLE H.25 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Off-site Individual from 1-Year Exposure to Atmospheric Releases from Support Facilities at Hanford during Normal Operations

Mode/Facility ^a	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^b	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
<u>Tritium Mode</u>					
Fuel processing					
HWR	2.8×10^{-6}	Thyroid	5.3×10^{-5}	5.3×10^{-5}	1×10^{-9}
LWR	6.5×10^{-6}	Thyroid	4.6×10^{-5}	4.6×10^{-5}	3×10^{-9}
MHTGR	3.8×10^{-6}	Thyroid	5.4×10^{-5}	5.4×10^{-5}	2×10^{-9}
Tritium-target processing					
HWR	5.5×10^{-5}	S. int.	6.4×10^{-5}	6.1×10^{-5}	2×10^{-8}
LWR	5.5×10^{-5}	S. int.	6.4×10^{-5}	6.1×10^{-5}	2×10^{-8}
MHTGR	5.5×10^{-5}	S. int.	6.4×10^{-5}	6.1×10^{-5}	2×10^{-8}
<u>Plutonium Mode</u>					
Fuel processing					
HWR	2.9×10^{-6}	Thyroid	6.2×10^{-5}	6.2×10^{-5}	1×10^{-9}
LWR	1.5×10^{-5}	Thyroid	2.1×10^{-4}	2.1×10^{-4}	6×10^{-9}
MHTGR	6.7×10^{-5}	Thyroid	2.1×10^{-4}	2.1×10^{-4}	3×10^{-8}
Plutonium-target processing ^c					
HWR	-	-	-	-	-
LWR	-	-	-	-	-
MHTGR	-	-	-	-	-

^aRadiological impacts from fuel fabrication facilities and plutonium-target fabrication facilities are estimated to be several orders of magnitude smaller than those from other support facilities.

^bS. int. = small intestine.

^cPlutonium-target processing and fuel processing would take place in a combined facility for the plutonium production mode, the fuel and plutonium-target processing facility. Impacts from plutonium-target processing are included under fuel processing.

TABLE H.26 Estimated Population Doses and Health Effects from 1-Year Exposure to Atmospheric Releases from Support Facilities at Hanford during Normal Operations

Mode/Facility ^a	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
<u>Tritium Mode</u>					
Fuel processing					
HWR	2.6×10^{-1}	6.4×10^{-2}	5.3	1×10^{-4}	3×10^{-5}
LWR	4.8×10^{-1}	2.5×10^{-1}	4.4	2×10^{-4}	1×10^{-4}
MHTGR	3.0×10^{-1}	8.1×10^{-2}	5.3	1×10^{-4}	4×10^{-5}
Tritium-target processing					
HWR	2.4	2.6	2.6	1×10^{-3}	7×10^{-4}
LWR	2.4	2.6	2.6	1×10^{-3}	7×10^{-4}
MHTGR	2.4	2.6	2.6	1×10^{-3}	7×10^{-4}
<u>Plutonium Mode</u>					
Fuel processing					
HWR	2.8×10^{-1}	5.8×10^{-2}	6.2	1×10^{-4}	3×10^{-5}
LWR	1.0	5.1×10^{-1}	1.7×10^1	4×10^{-4}	2×10^{-4}
MHTGR	4.1	4.0	1.6×10^1	2×10^{-3}	1×10^{-3}
Plutonium-target processing ^b					
HWR	-	-	-	-	-
LWR	-	-	-	-	-
MHTGR	-	-	-	-	-

^aRadiological impacts from fuel fabrication facilities and plutonium-target fabrication facilities are estimated to be several orders of magnitude smaller than those from other support facilities.

^bPlutonium-target processing and fuel processing would take place in a combined facility for the plutonium production mode, the fuel and plutonium-target processing facility. Impacts from plutonium-target processing are included under fuel processing.

TABLE H.27 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Off-site Individual from 1-Year Exposure to Atmospheric Releases from Support Facilities at INEL during Normal Operations

Mode/Facility ^a	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^b	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
<u>Tritium Mode</u>					
Fuel processing					
HWR	8.1×10^{-6}	Thyroid	2.8×10^{-5}	2.8×10^{-5}	3×10^{-9}
LWR	3.0×10^{-6}	Thyroid	2.5×10^{-5}	2.5×10^{-5}	1×10^{-9}
MHTGR	1.6×10^{-5}	Endosteal	1.0×10^{-4}	3.2×10^{-5}	6×10^{-9}
Tritium-target processing					
HWR	9.8×10^{-6}	S. int.	1.1×10^{-5}	1.1×10^{-5}	4×10^{-9}
LWR	9.8×10^{-6}	S. int.	1.1×10^{-5}	1.1×10^{-5}	4×10^{-9}
MHTGR	9.8×10^{-6}	S. int.	1.1×10^{-5}	1.1×10^{-5}	4×10^{-9}
<u>Plutonium Mode</u>					
Fuel processing					
HWR	8.1×10^{-6}	Thyroid	2.8×10^{-5}	2.8×10^{-5}	3×10^{-9}
LWR ^c	8.0×10^{-6}	Thyroid	1.6×10^{-4}	1.6×10^{-4}	3×10^{-9}
MHTGR	1.6×10^{-5}	Endosteal	1.0×10^{-4}	3.2×10^{-5}	6×10^{-9}
Plutonium-target processing					
HWR	7.1×10^{-7}	Thyroid	1.5×10^{-6}	1.5×10^{-6}	3×10^{-10}
LWR ^c	-	-	-	-	-
MHTGR	3.7×10^{-6}	Thyroid	1.2×10^{-5}	1.2×10^{-5}	2×10^{-9}

^aRadiological impacts from fuel fabrication facilities and plutonium-target fabrication facilities are estimated to be several orders of magnitude smaller than those from other support facilities.

^bS. int. = small intestine.

^cFor the LWR technology, most plutonium-target processing functions would take place in a combined facility, the fuel and plutonium-target processing facility. Impacts of LWR plutonium-target processing are included under fuel processing.

TABLE H.28 Estimated Population Doses and Health Effects from 1-Year Exposure to Atmospheric Releases from Support Facilities at INEL during Normal Operations

Mode/Facility ^a	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
<u>Tritium Mode</u>					
Fuel processing					
HWR	1.1×10^{-1}	7.1×10^{-2}	4.6×10^{-1}	5×10^{-5}	2×10^{-5}
LWR	5.2×10^{-2}	2.4×10^{-2}	5.2×10^{-1}	2×10^{-5}	1×10^{-5}
MHTGR	1.8×10^{-1}	4.9×10^{-2}	5.8×10^{-1}	7×10^{-5}	2×10^{-5}
Tritium-target processing					
HWR	2.0×10^{-1}	2.1×10^{-1}	2.1×10^{-1}	8×10^{-5}	6×10^{-5}
LWR	2.0×10^{-1}	2.1×10^{-1}	2.1×10^{-1}	8×10^{-5}	6×10^{-5}
MHTGR	2.0×10^{-1}	2.1×10^{-1}	2.1×10^{-1}	8×10^{-5}	6×10^{-5}
<u>Plutonium Mode</u>					
Fuel processing					
HWR	1.1×10^{-1}	7.1×10^{-2}	4.6×10^{-1}	5×10^{-5}	2×10^{-5}
LWR ^b	1.3×10^{-1}	5.1×10^{-2}	2.6	5×10^{-5}	2×10^{-5}
MHTGR	1.8×10^{-1}	4.9×10^{-2}	5.8×10^{-1}	7×10^{-5}	2×10^{-5}
Plutonium-target processing					
HWR	1.2×10^{-2}	9.7×10^{-3}	2.8×10^{-2}	5×10^{-6}	3×10^{-6}
LWR ^b	-	-	-	-	-
MHTGR	6.5×10^{-2}	6.1×10^{-2}	2.2×10^{-1}	3×10^{-5}	2×10^{-5}

^aRadiological impacts from fuel fabrication facilities and plutonium-target fabrication facilities are estimated to be several orders of magnitude smaller than those from other support facilities.

^bFor the LWR technology, most plutonium-target processing functions would take place in a combined facility, the fuel and plutonium-target processing facility. Impacts of LWR plutonium-target processing are included under fuel processing.

TABLE H.29 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Off-Site Individual from 1-Year Exposure to Atmospheric Releases from Support Facilities at SRS during Normal Operations

Mode/Facility ^a	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^b	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
<u>Tritium Mode</u>					
Fuel processing					
HWR	1.9×10^{-5}	Thyroid	1.2×10^{-4}	1.2×10^{-4}	8×10^{-9}
LWR	1.7×10^{-5}	Thyroid	1.6×10^{-4}	1.6×10^{-4}	7×10^{-9}
MHTGR	1.6×10^{-4}	Thyroid	4.4×10^{-3}	4.4×10^{-3}	6×10^{-8}
Tritium-target processing					
HWR	1.3×10^{-5}	S. int.	1.5×10^{-5}	1.4×10^{-5}	5×10^{-9}
LWR	1.3×10^{-5}	S. int.	1.5×10^{-5}	1.4×10^{-5}	5×10^{-9}
MHTGR	1.3×10^{-5}	S. int.	1.5×10^{-5}	1.4×10^{-5}	5×10^{-9}
<u>Plutonium Mode</u>					
Fuel processing					
HWR	1.9×10^{-5}	Thyroid	1.2×10^{-4}	1.2×10^{-4}	8×10^{-9}
LWR ^c	3.7×10^{-5}	Thyroid	8.0×10^{-4}	8.0×10^{-4}	2×10^{-8}
MHTGR	1.6×10^{-4}	Thyroid	4.4×10^{-3}	4.4×10^{-3}	6×10^{-8}
Plutonium-target processing					
HWR	2.5×10^{-5}	Thyroid	1.5×10^{-4}	1.5×10^{-4}	1×10^{-8}
LWR ^c	-	-	-	-	-
MHTGR	1.7×10^{-4}	Thyroid	2.2×10^{-3}	2.2×10^{-3}	7×10^{-8}

^aRadiological impacts from fuel fabrication facilities and plutonium-target fabrication facilities are estimated to be several orders of magnitude smaller than those from other support facilities.

^bS. int. = small intestine.

^cFor the LWR technology, most plutonium-target processing functions would take place in a combined facility, the fuel and plutonium-target processing facility. Impacts of LWR plutonium-target processing are included under fuel processing.

TABLE H.30 Estimated Population Doses and Health Effects from 1-Year Exposure to Atmospheric Releases from Support Facilities at SRS during Normal Operations

Mode/Facility ^a	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
<u>Tritium Mode</u>					
Fuel processing					
HWR	1.5	1.1	6.7	6×10^{-4}	3×10^{-4}
LWR	1.6	5.0×10^{-1}	9.0	6×10^{-4}	4×10^{-4}
MHTGR	1.0×10^1	7.3×10^{-1}	2.5×10^2	4×10^{-3}	3×10^{-4}
Tritium-target processing					
HWR	9.4×10^{-1}	1.0	1.0	4×10^{-4}	3×10^{-4}
LWR	9.4×10^{-1}	1.0	1.0	4×10^{-4}	3×10^{-4}
MHTGR	9.4×10^{-1}	1.0	1.0	4×10^{-4}	3×10^{-4}
<u>Plutonium Mode</u>					
Fuel processing					
HWR	1.5	1.1	6.7	6×10^{-4}	3×10^{-4}
LWR ^b	1.9	1.0	3.1×10^1	8×10^{-4}	3×10^{-4}
MHTGR	1.0×10^1	7.3×10^{-1}	2.5×10^2	4×10^{-3}	3×10^{-4}
Plutonium-target processing					
HWR	1.5	1.1	6.7	6×10^{-4}	3×10^{-4}
LWR ^b	-	-	-	-	-
MHTGR	1.1×10^1	8.1	1.2×10^2	4×10^{-3}	2×10^{-3}

^aRadiological impacts from fuel fabrication facilities and plutonium-target fabrication facilities are estimated to be several orders of magnitude smaller than those from other support facilities.

^bFor the LWR technology, most plutonium-target processing functions would take place in a combined facility, the fuel and plutonium-target processing facility. Impacts of LWR plutonium-target processing are included under fuel processing.

TABLE H.31 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Off-site Individual from 1-Year Exposure to Liquid Releases from Support Facilities at SRS during Normal Operations

Mode/Facility ^a	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^b	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
<u>Tritium Mode</u>					
Fuel processing					
HWR	7.2×10^{-6}	R. marrow	8.7×10^{-6}	8.4×10^{-6}	3×10^{-9}
LWR ^c	-	-	-	-	-
MHTGR ^c	-	-	-	-	-
Tritium-target processing					
HWR ^d	-	-	-	-	-
LWR ^c	-	-	-	-	-
MHTGR ^c	-	-	-	-	-
<u>Plutonium Mode</u>					
Fuel processing					
HWR	7.2×10^{-6}	R. marrow	8.7×10^{-6}	8.4×10^{-6}	3×10^{-9}
LWR ^c	-	-	-	-	-
MHTGR ^c	-	-	-	-	-
Plutonium-target processing					
HWR	7.2×10^{-6}	R. marrow	8.7×10^{-6}	8.3×10^{-6}	3×10^{-9}
LWR ^c	-	-	-	-	-
MHTGR ^c	-	-	-	-	-

^aRadiological impacts from fuel fabrication facilities and plutonium-target fabrication facilities are estimated to be several orders of magnitude smaller than those from other support facilities.

^bR. marrow = red bone marrow.

^cThere are no liquid releases for this technology.

^dThere are no liquid releases from the HWR tritium-target processing facility.

TABLE H.32 Estimated Population Doses and Health Effects from 1-Year Exposure to Liquid Releases from Support Facilities at SRS during Normal Operations

Mode/Facility ^a	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
<u>Tritium Mode</u>					
Fuel processing					
HWR	2.7	2.9	3.0	1×10^{-3}	8×10^{-4}
LWR ^b	-	-	-	-	-
MHTGR ^b	-	-	-	-	-
Tritium-target processing					
HWR ^c	-	-	-	-	-
LWR ^b	-	-	-	-	-
MHTGR ^b	-	-	-	-	-
<u>Plutonium Mode</u>					
Fuel processing					
HWR	2.7	2.9	3.0	1×10^{-3}	8×10^{-4}
LWR ^b	-	-	-	-	-
MHTGR ^b	-	-	-	-	-
Plutonium-target processing					
HWR	2.7	2.9	3.0	1×10^{-3}	8×10^{-4}
LWR ^b	-	-	-	-	-
MHTGR ^b	-	-	-	-	-

^aRadiological impacts from fuel fabrication facilities are estimated to be several orders of magnitude smaller than those from other support facilities.

^bThere are no liquid releases for this technology.

^cThere are no liquid releases from the HWR tritium-target processing facility.

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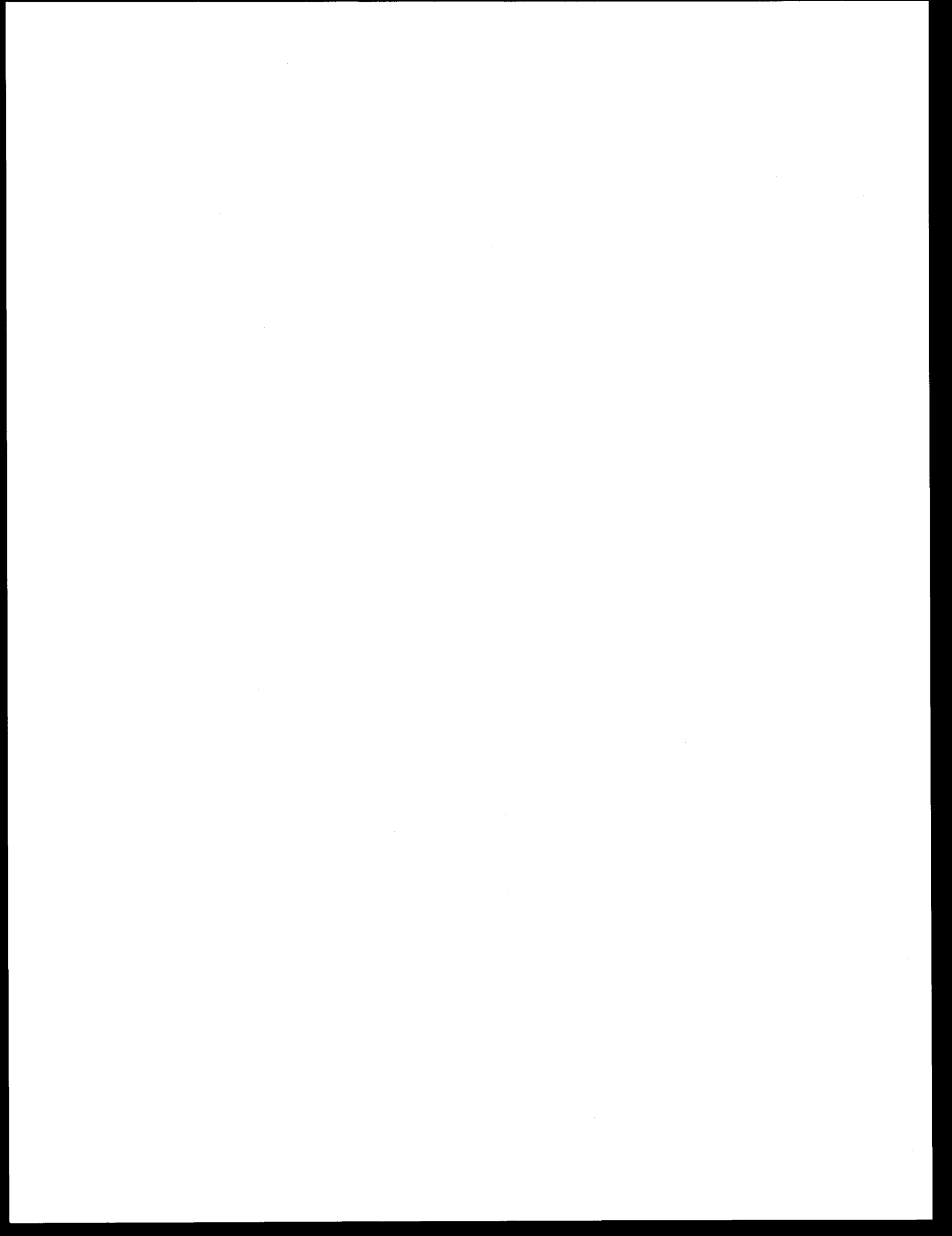
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APPENDIX I:
RADIOLOGICAL IMPACTS OF POSTULATED NONSEVERE ACCIDENTS



APPENDIX I:

RADIOLOGICAL IMPACTS OF POSTULATED NONSEVERE ACCIDENTS

I.1 INTRODUCTION

Detailed information on potential radiological impacts of postulated nonsevere accidents in new production reactors and their support facilities at the three alternative sites are presented in this appendix; this information is intended to support the impact assessments summarized in Sections 5.2.6.3, 5.3.6.3, and 5.4.6.3 of the main text. The methodology used to assess radiological impacts of nonsevere accidents is described in Section I.2. Descriptions of postulated accidents and their impacts are presented in Sections I.3 and I.4.

I.2 METHODOLOGY FOR ESTIMATING RADIOLOGICAL IMPACTS OF NONSEVERE ACCIDENTS

The GENII computer code was used to calculate environmental transport and consequences for postulated nonsevere accidents. This code was also used to estimate impacts of normal operations, and a summary description of GENII is provided in Appendix H, Section H.3.1. Complete descriptions of the models contained in GENII and its input and output options are given in Napier et al. (1988) and Rhoads (1990).

Doses were estimated for (1) the maximally exposed off-site individual, who lives at the place of maximum exposure on the site boundary and consumes only locally grown food, (2) the off-site population within an 80-km radius of the facility, and (3) the maximally exposed colocated worker, a person working on-site at a planned or existing facility other than the one being evaluated. Because atmospheric transport of material released from a short-term accident is expected to be highly directional, acute population dose was evaluated only in the sector where the release would result in maximum impact; the sector of maximum impact is based on statistical frequencies of atmospheric conditions and the number of people located in the sector. The dose received by the population in this sector is called the worst-sector population dose. This dose is a conservative estimate of the dose received by the entire 80-km population because, if the wind direction shifted and the wind blew into a sector other than the worst sector, the calculated population dose would be reduced.

The calculated doses were converted to health effects using the same risk estimators discussed in Appendix H, Section H.3.3. Numerically these factors are 4×10^{-4} incremental fatal cancers per person-rem of collective committed effective dose equivalent and 2.6×10^{-4} severe genetic effects per person-rem of collective committed dose equivalent to gonads plus collective external effective dose equivalent. These factors are consistent with recommendations in the BEIR V report (National Research Council 1990) of the National Research Council's Committee on the Biological Effects of Ionizing Radiation (Section 5.1.7.2).

Possible mitigative actions for off-site individuals in the event of a nonsevere accident are discussed in Section 5.1.7.2. Further guidance regarding when such actions might be taken by responsible authorities is provided in DOE (1989), FEMA (1980), EPA (1989), FDA (1982), and Shleien et al. (1982).

Estimated radiological impacts of postulated nonsevere accidents are presented in terms of potential doses and health effects to the maximally exposed off-site individual, worst-sector off-site population, and maximally exposed colocated worker. Impacts are presented for a postulated occurrence of each accident. Estimated doses to in-plant workers are also provided. Such accidents are usually analyzed in order to define the design envelope for safety-related structures, systems, and components important to safety at the facilities. Specific quantitative probabilities of occurrence are not assigned for each postulated nonsevere accident; instead, each accident is assigned to a category that is based on a probability range. Consequently, within this EIS, no attempt is made to develop frequencies and to estimate the total risk from a spectrum of nonsevere accidents.

I.2.1 Data and Assumptions

I.2.1.1 Data Descriptions

Meteorological Data

The meteorological data used for the Hanford site, INEL and SRS were in the form of joint wind frequency data (JFD) files. A JFD file is a table listing the fractions of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The JFD files were based on measurements over a 5-year period at various locations and at different heights at the three alternative sites. The JFD files for both normal operations and nonsevere accidents for each facility were the same, except that median air concentrations rather than average air concentrations were used for the nonsevere accident calculations. This is consistent with the final EIS for continued operation of K, L, and P reactors at SRS and with practices used in evaluations of commercial nuclear power plants (DOE 1990). The files used for the calculations in this EIS are provided in separate reports (Avci and Biwer 1991; Kier and Avci 1991).

Population Data

Population projections were based on U.S. Bureau of the Census statistics. Projections were determined for the year 2020 (projected midpoint in the life of the proposed NPR facilities) for individuals living in the vicinity of each alternative site. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances up to 80 km. The facility releasing the radionuclides was assumed to be at the center of the circular grid. A separate grid was established for each major NPR facility or for a group of facilities close to each other at each site. Population data are provided in separate reports (Avci and Biwer 1991; Kier and Avci 1991).

Source Term Data

A set of nonsevere accidents was postulated, and the source terms (quantities of radionuclides released to the environment) for each reactor technology were estimated on the basis of generic or conceptual designs and experience with similar facilities. The source terms used to generate the nonsevere accident impacts in this EIS are provided in Section I.3 for

reactors and Section I.4 for support facilities. They are also provided in separate reports (Avci and Biwer 1991; Kier and Avci 1991).

Food Production and Consumption Data

Site-specific data were used for food production and consumption rates when available. Otherwise, generic data from NRC regulatory guides (see, for example, Regulatory Guide 1.109 [NRC 1977]) were used. It was assumed that the eating habits of people living in the worst sector were representative of the people living within about 80 km of the site boundary. It was also assumed that the food for the population within the worst sector came only from that sector. Food production and consumption data are provided in separate reports (Avci and Biwer 1991; Kier and Avci 1991).

I.2.1.2 Calculational Parameters and Assumptions

To estimate the radiological impacts from nonsevere accidents at reactors and support facilities, additional assumptions and factors were considered, as follows:

- A finite plume model was used. Other pathways evaluated were ground exposure, inhalation, and ingestion of terrestrial food and animal products.
- All atmospheric releases for reactor accidents were assumed to be from a stack 10 m high (considered to be about an average release point for atmospheric releases from a reactor). The stack heights for support facilities were facility-specific. Whenever a stack release was considered, the actual stack height was used as the effective stack height, thus ignoring the plume rise and adding conservatism to the dose estimates.
- The calculated doses were 50-year committed doses from 1 year of ingestion intake.
- The fraction of time spent in the plume was 1.0 (used for plume exposure and inhalation intake).
- For the maximally exposed off-site individual, the annual exposure time to soil contamination was 0.7 years (based on NRC Regulatory Guide 1.109 [NRC 1977]).
- For the population, the annual exposure time to soil contamination was 0.5 years (based on NRC Regulatory Guide 1.109 [NRC 1977]).
- Resuspension of particulates was not considered because prior calculations of dust loading in the atmosphere showed that this pathway was negligible compared with others.
- For population exposures, the average individual parameters and options in GENII were exercised. Average individual parameters assume that the individual breathes and consumes food at average rates (maximum

individual parameters assume higher breathing and food consumption rates).

The maximally exposed colocated worker was assumed to be evacuated from the site 2 hours after the initiation of an accident. Therefore, this worker's soil exposure time was set at 2 hours and the ingestion pathway was not considered. It was also assumed that the colocated worker was exposed to the whole plume during the 2 hours.

1.2.2 Uncertainties and Conservatism in Estimated Impacts

The sequence of analyses performed to generate the radiological impact estimates from nonsevere accidents include (1) selection of accident sequences, (2) estimation of source terms, (3) estimation of environmental transport and uptake of radionuclides, (4) calculation of radiation doses to exposed individuals, and (5) estimation of health effects. There are uncertainties associated with each of these steps. Uncertainties exist in the way the physical systems being analyzed are represented by the computational models, in the data required to exercise the models (due to measurement errors, sampling errors, or natural variability), and in the calculations themselves (e.g., roundoff errors by the computers).

In principle, one can estimate the uncertainty associated with each source and predict the remaining uncertainty in the results of each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final results. However, conducting such a full-scale quantitative uncertainty analysis is neither practical nor a standard practice for an EIS. Instead, the analysis is designed to ensure -- through judicious selection of release scenarios, models, and parameters -- that the results are bounding. This is accomplished by making conservative assumptions in the calculations at each step. The release scenarios, models, and parameters used in the calculations are selected in such a way that most intermediate results and, consequently, the final estimates of impacts are greater than what would be expected. As a result, even though the range of uncertainty in a quantity might be large, the value calculated for the quantity is close to one of the extremes in the range of possible values, so that the chance of the actual quantity being greater than the calculated value is low (or the chance of the quantity being less than the calculated value if the criteria are such that the quantity has to be maximized). This has been the goal of the radiological assessment for nonsevere accidents in this EIS (i.e., to produce results that are conservative).

The degree of conservatism in the calculated results is closely related to the range of possible values the quantity can have. This range is determined by what can be expected to occur realistically. Thus, the only processes and events considered are those credible for the conditions under which the physical system being modeled operates. This consideration has also been employed for the analysis of nonsevere accidents in this EIS.

In summary, the radiological impact estimates presented in this EIS for nonsevere accidents were obtained by

- Using state-of-the-art computational tools,
- Considering the processes and events that can realistically be expected to occur in the proposed NPR facilities and the environment, and

- Making conservative assumptions when there is doubt about the exact nature of the processes and events taking place.

In most instances, the desire to provide a best estimate has been counterbalanced by the desire to be conservative. In some cases, however, the estimates presented might be overly conservative. One example is the estimated doses and health effects to the maximally exposed off-site individual from the ingestion pathway. It is not realistic to assume that, for all accidents analyzed, these individuals stay in the worst possible place at the site boundary, are unprotected throughout the accident, and continue to live there for an entire year consuming only locally grown food. Thus, considering this and all other conservative assumptions that have been made to estimate the source terms and environmental transport of radionuclides, the impacts estimated for maximally exposed off-site individuals are considered to be very conservative.

For population exposures, it has been assumed that the wind blows in the worst possible direction during the entire accident and that the people exposed and all the vegetation are always directly under the plume centerline. It was also assumed that all dietary needs of exposed persons were obtained from the sector in which they live (which is the worst of the 16 sectors around the facility where the accident is postulated to occur). No mitigating conditions in terms of evacuation, sheltering, or food interdiction were assumed. Considering these assumptions and other conservative assumptions made in the selection of accident sequences, calculation of source terms, and estimation of environmental transport and human consumption, the final estimates of population doses and health effects presented in this EIS are conservative (even though the meteorological conditions used represent median conditions).

The amount of conservatism is relatively smaller in the impacts estimated for maximally exposed colocated workers than for the maximally exposed off-site individual and off-site population because the ingestion pathway was not evaluated for the colocated workers. Most site evacuation plans call for evacuation of workers from the site in the event of a serious accident in much less than 2 hours. It is also likely that these personnel could be sheltered on-site, which would reduce their inhalation and external exposure doses. Because of these factors, the impacts to maximally exposed colocated workers presented in this EIS are also considered to be conservative.

Another major area of uncertainty for the nonsevere accident analyses is the lack of well-defined designs for the proposed facilities (Section 5.1.7.2). The design process is being carried out to ensure that any potential impacts from these postulated accidents are within any applicable guidelines and as low as reasonably achievable. Therefore, at this point, the nonsevere accident impacts presented in this EIS should be viewed as upper bound estimates of probable impacts.

I.3 RADIOLOGICAL IMPACTS FROM POSTULATED NONSEVERE ACCIDENTS AT REACTORS

I.3.1 Accident Descriptions

For each reactor technology (HWR, LWR, and MHTGR), the postulated nonsevere accidents and the environmental source terms of radioactive materials for these accidents were considered to be the same at each of the three sites. There might be differences among the sites

(e.g., seismic hazard potentials) that would require design modifications so that accident frequencies and source terms for a given reactor type would be the same for all sites. Alternatively, the same reactor might be designed such that the source terms and consequences of the postulated accidents might differ from one site to the next. These differences, however, are not expected to be viewed as major contrasts among the sites and are not considered in this EIS. Therefore, the accidents described in Sections I.3.1.1, I.3.1.2, and I.3.1.3 and the source terms listed in Section I.3.2 apply to all sites.

Nonsevere accidents are categorized by their relative frequency of occurrence. Moderate frequency events are events that are reasonably expected to occur once during any year of reactor operations, infrequent events are reasonably expected to occur once during the plant's lifetime, and limiting faults are highly unlikely events that are not expected to occur during the plant's lifetime.

The analysis for LWR nonsevere accidents included consideration of the standard list of prescriptive accidents usually presented for commercial LWRs. The analysis for HWR and MHTGR nonsevere accidents included a spectrum of conservative accidents that bounds the impacts of such accidents. For this reason, the number of nonsevere accidents analyzed for the LWR is larger than the number analyzed for the HWR and MHTGR.

I.3.1.1 HWR Accidents

The postulated HWR nonsevere accidents considered for evaluating environmental consequences and the expected frequency classes of these accidents are listed in Table I.1.* A short description of each accident is given below. More detailed information regarding these accidents, as well as other accidents considered but determined to have fewer impacts, is given in Section 5.3 of WSRC (1991a) and in Bickford (1990a-c). The accident identification (ID) codes listed in Table I.1 are used throughout this appendix in lieu of long names for accidents. The HWR facility is described in Section 2.1.3.1 and Appendix A of this EIS.

Detritiation System Failure (HWR1)

The heavy-water processing facility is a support facility located at the HWR site. This facility includes systems for the isotopic and chemical purification of the heavy-water primary coolant/moderator and also includes a detritiation module. The detritiation equipment will collect and concentrate tritium produced in the heavy-water coolant/moderator during routine HWR operations. A nonsevere accident (HWR1) is postulated in which a storage vessel for tritium gas fails during operation of the heavy-water processing facility, resulting in release to the environment of 3×10^5 Ci of tritium (maximum allowable inventory in the vessel).

*For readability, all tables are presented in sequence at the end of the text in this appendix.

Large-Break Loss-of-Coolant Accident for Tritium- and Plutonium-Producing Core (HWR2)

A large-break loss-of-coolant accident (HWR2) can release a significant quantity of primary heavy-water coolant into the reactor containment building. Radioactive material present in the coolant may then be available for release from the containment building to the environment.

Heat Exchanger Tube Rupture (HWR3)

An accident involving the rupture of a heat exchanger tube (HWR3) will allow primary heavy-water coolant to pass into the secondary-side coolant. Entrained radioactive material may then be released from the secondary-side coolant into the environment via the intake basin.

Charge-and-Discharge Mishap with Tritium-Producing Charge (HWR4)

In the HWR, fuel is assumed to be discharged from the reactor into a hot cell refueling canyon above the reactor head using a computer-controlled remote crane. The fuel is then transferred in air, but with water cooling, to a light-water-filled canal and then moved submerged in water to a storage and disassembly area. During this transfer operation, the potential exists for failure of the crane motive system and the water delivery system used to cool the fuel, leading to melting of one assembly in the atmosphere of the hot cell canyon. In this accident, such failures are postulated to occur during the charge-and-discharge operations of an irradiated fuel assembly containing tritium targets.

Charge-and-Discharge Mishap with Plutonium-Producing Charge (HWR5)

Accident HWR5 is the same as HWR4, except that it involves an assembly containing plutonium targets instead of tritium targets.

I.3.1.2 LWR Accidents

The postulated LWR nonsevere accidents considered for evaluating environmental consequences and the expected frequencies of occurrence of these accidents are listed in Table I.2. A short description of each accident is given below. More detailed information regarding these accidents is given in Section 5.1 of WHC (1990a). The accident ID codes listed in Table I.2 are used throughout this document in lieu of long names for accidents.

Two types of LWR core designs are considered: tritium getter-barrier core and plutonium core. The accidents listed in Table I.2 apply to both core designs. The letter "A" is appended at the end of the accident designation when it refers to the tritium getter-barrier core and the letter "C" when it refers to the plutonium core. For example, LWR1A refers to the tritium getter-barrier core and LWR1C to the plutonium core.

Equipment Leakage or Malfunction Resulting in Release of 25% of the Normal Inventory of a Waste-Gas Decay Tank (LWR1)

The waste-gas decay tanks are located in the waste processing area. These tanks accept gases stripped from reactor coolant and provide holdup to allow for radioactive decay. Accident LWR1 is postulated to occur through a pipe leak that is undetected for a time sufficient to result in the release of 25% of the normal tank inventory.

Equipment Leakage or Malfunction Resulting in Release of 100% of the Normal Inventory of a Waste-Gas Decay Tank (LWR2)

In accident LWR2, one of the waste-gas decay tanks is assumed to rupture, releasing 100% of its inventory.

Off-Design Transients that Induce Fuel Failures above Those Expected and Steam Generator Leak Combined with Station Blackout (LWR3)

Two classes of events are included in accident LWR3. The first class pertains to an incident in which a core transient might induce additional fuel failures and an increased release of effluents to the atmosphere through the main condenser exhaust; such an incident would occur if a primary-to-secondary leakage exists in the steam generator. The second class pertains to an incident in which secondary-system steam is released through the steam relief valves either due to valve malfunction or loss of load. This second class of events will not result in primary transients severe enough to cause additional fuel failures. Station blackout is the limiting case for both classes of events because it results in maximum steam dump from the secondary side. It is analyzed on the assumption that there will be additional fuel failures in the core.

Steam Generator Tube Rupture (LWR4)

The steam generator tube rupture (accident LWR4) is defined as the complete severance of one steam generator tube while the reactor is operating at full power. The complete severance of a steam generator tube will allow fission products in the primary-system coolant to contaminate the secondary-side coolant. This will lead to the release of radioactive material to the environment by way of the condenser off-gas and venting of the secondary system.

Refueling Accident - Fuel Assembly Drop (LWR5)

Equipment design is of major importance in preventing the dropping of fuel assemblies. For example, inadvertent disengagement of a fuel assembly from the fuel handling machine is prevented by positive grappling design features. The maximum elevation to which the fuel assemblies can be raised is limited by the design of the fuel handling hoists and manipulators. In addition, a mechanical stop prevents the hoist from raising fuel above the point where just enough water is available for shielding. Hence, at no time during the transfer from the reactor core to the spent fuel storage rack is a fuel assembly removed from water.

It has been assumed that the dropping of a fuel assembly (accident LWR5) would result in the failure of 19 fuel rods, an entire row. Because of the length of the fuel assembly, it is highly probable that the assembly would strike the reactor cavity wall or other high object as it rotated, thus breaking its fall and reducing the effects of impact. Hence, the assumption that the direct angle of strike would cause one row of fuel rods to fail is quite conservative. A glancing angle of strike that would reduce or preclude fuel damage is more probable. In the case of the tritium production core, the target rods are not located on the outside edges of the assembly and, therefore, are not subject to mechanical damage from this event.

During refueling, the containment building atmosphere is purged through charcoal filters. The containment building is isolated if detectors indicate that radio-active material has been released into it.

Refueling Accident - Heavy Object Drop onto Fuel in Core (LWR6)

The heaviest object routinely lifted over the reactor vessel is the reactor vessel head. Even if dropped, the head would probably not damage any fuel because it cannot physically fit inside the vessel. The total vertical lift at that point is limited to 1.8 m.

The upper core support assembly is also removed during each refueling. This structure fits inside the reactor vessel and if dropped could impose a load on the fuel. However, this would require the slots in the core plate to line up exactly with the pins in the core barrel. If this alignment did not occur, the core support assembly would be prevented from falling all the way into the reactor vessel. The only other heavy objects handled over the reactor vessel are the spent fuel assemblies. Thus, accident LWR6 is hypothetical, and the conditions are prescriptive in accordance with regulatory guidance (see proposed annex to Appendix D of 10 CFR 50).

Spent Fuel Handling Accident - Fuel Assembly Drop into Fuel Storage Pool (LWR7)

Accident LWR7 is essentially the same as accident LWR5 except that the location of the fuel assembly drop accident is the General Services Building outside of the primary containment. The fuel storage area in the General Services Building is ventilated through roughing and high-efficiency particulate air (HEPA) and charcoal filters.

Spent Fuel Handling Accident - Heavy Object Drop onto Fuel Rack (LWR8)

The largest item handled in the General Services Building is the spent fuel shipping cask. However, the crane and building arrangements are such that it is impossible for the cask to be carried over the fuel storage racks. The hoist that spans the storage pool serves as the fuel handling platform. This motor-driven walkway is designed to preclude the platform from tipping over and to safely withstand the forces associated with earthquakes. Thus, accident LWR8 is hypothetical in nature, and the prescriptive requirements of the guidance document are applied (see proposed annex to Appendix D of 10 CFR 50).

Loss-of-Coolant Accident - Small Pipe Break (LWR9)

A small pipe, 15 cm or less in diameter, in the primary coolant system is assumed to break in accident LWR9. The primary coolant discharges through the break into the containment, releasing any radioactive material contained in the coolant. The reactor is assumed to be operating with 0.5% of its fuel elements defective.

Loss-of-Coolant Accident - Large Pipe Break (LWR10)

A large pipe in the primary coolant system is assumed to break in accident LWR10. This results in rapid discharge of reactor coolant from the broken pipe into the containment. Any radioactive material contained in the coolant or released to the coolant as a result of fuel clad failures would escape to the containment.

Loss-of-Coolant Accident - Break in Letdown Line from Primary System that Penetrates the Containment (LWR11)

In accident LWR11, a fluid-bearing line from the primary system that penetrates the containment is assumed to break, resulting in the release of radioactive material to the environment. The lines that are connected to the primary system and penetrate the containment are the piping lines to the makeup and purification system and the decay heat removal system.

The accident of this type with the potential for the largest release of radioactive material is a pipe rupture in the makeup and purification system -- more specifically, a rupture of the letdown line just outside the containment but upstream of the letdown control valves. A rupture at this point will result in loss of reactor coolant until the reactor coolant system pressure drops below its low pressure setpoint of $1.1 \times 10^7 \text{ N/m}^2$ (gage). When this pressure is reached, the emergency injection signal initiates the closure of the letdown isolation valve inside the containment, thus terminating the accident.

Rod Ejection Accident (LWR12)

Accident LWR12 is defined as the mechanical failure in the pressure housing of a control rod mechanism, resulting in the ejection of a rod cluster control assembly and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. The radioactive material released to the coolant is expelled from the reactor vessel through the broken rod housing to the primary containment system.

Steam-Line Break Outside Containment (LWR13)

Accident LWR13 is defined as the instantaneous rupture of a main steam line outside the containment. From an environmental viewpoint, this type of break is much more serious than a break inside the containment because any radioactive material contained in the flashing mixture will be released directly to the environment.

I.3.1.3 MHTGR Accidents

An assessment of fission products and tritium release from the MHTGR during postulated accidents is provided in EG&G (1989c) and Bowman (1990a-e). On the basis of this assessment, six accidents were identified as bounding the environmental consequences from nonsevere MHTGR accidents. The postulated MHTGR accidents and their expected frequency classes are listed in Table I.3. A short description of each accident is given below. More detailed information regarding these accidents is presented in Section 5.3 of EG&G (1989c) and in Bowman (1990a-e). The accident ID codes listed in Table I.3 are used throughout this appendix in lieu of long names for accidents.

Small Primary Break with Normal Containment Leak Rate of 1% per Day (GCR1)

In accident GCR1, a small break in the primary-system piping results in release of the circulating radioactive material in the primary coolant into the containment. The containment leak rate to the environment is assumed to be 1% per day (a design criterion).

Moderate Primary Break with Normal Containment Leak Rate of 1% per Day (GCR2)

In accident GCR2, a moderate-sized break in the primary-system piping results in release of the circulating radioactive material in the primary coolant into the containment. The shear force of the coolant as it escapes the primary system is assumed to lift off any radioactive material deposited on the inside surfaces of the primary system and cause it to be released into the containment along with the circulating radioactive material. The containment leak rate to the environment is assumed to be 1% per day.

Small Primary Break with Containment Bypass Leak Rate of 100% per Day (GCR3)

Accident GCR3 is the same as GCR1 except it is assumed that the containment fails to be isolated from the environment. As a result, the leak rate is 100% per day.

Moderate Primary Break with Containment Bypass Leak Rate of 100% per Day (GCR4)

Accident GCR4 is the same as GCR2 except it is assumed that the containment fails to be isolated from the environment. As a result, the leak rate is 100% per day.

Depressurized Conduction Cooldown with Normal Containment Leak Rate of 1% per Day (GCR5)

In accident GCR5, a large break in the primary-system piping releases the circulating and plateout radioactive material into the containment. At 50 to 80 hours into the accident, the

radioactive material released from the core as a result of a temperature excursion to 1,320°C starts being released into the containment. The containment leak rate to the environment is assumed to be 1% per day, which is the design criterion of the MHTGR.

**Depressurized Conduction Cooldown with Containment Bypass
Leak Rate of 100% per Day followed by Normal Containment
Leak Rate of 1% per Day (GCR6)**

Accident GCR6 is the same as accident GCR5 except that the containment failure to isolate is assumed to exist throughout the release of circulating and plateout activity. The leak is assumed to be mitigated at the start of the release associated with the thermal transient (at 50 to 80 hours after the initiation of the accident).

I.3.2 Results of Analyses

The postulated NPR nonsevere accidents listed in Tables I.1, I.2, and I.3 have been analyzed using the methodology described in Section I.2. Each accident is associated with a certain frequency class. In this section, the consequences are presented only for those accidents that are bounding within each frequency class. For the HWR, these accidents are HWR4 (charge-and-discharge mishap involving a fuel assembly with tritium-producing charge) for the infrequent frequency class and HWR1 (detrification system failure) for the limiting frequency class. However, because accident HWR1 is associated with the heavy water processing facility, consequences are also presented for a limiting frequency class accident involving the reactor primary system and the core, i.e., accident HWR2 (large-break loss-of-coolant accident [LOCA]). These accidents bound the consequences for both the tritium and the plutonium production options in the HWR technology.

For the LWR, different accidents in the infrequent class cause the highest consequences for different receptors. For the maximally exposed off-site individual and the off-site population, this accident is LWR4A (steam generator tube rupture accident for the tritium production core design); for the maximally exposed colocated worker, it is LWR2A (waste-gas decay tank rupture for the tritium production core design). In the limiting frequency class, LWR10A (large pipe break LOCA for the tritium production core design) is the accident with the highest consequences for all receptors considered. Accidents associated with the plutonium production core design have lower consequences than those associated with the tritium production core design, so the consequences presented bound the consequences for both types of core designs.

For the MHTGR, the moderate frequency class accident is GCR1 (small primary break with normal containment leak rate of 1% per day) and the infrequent frequency class accident is GCR2 (moderate primary break with normal containment leak rate of 1% per day). In the limiting class, the bounding accident is GCR6 (depressurized conduction cooldown with failure to isolate the containment during the first 50 hours, followed by normal containment leak rate of 1% per day). All three accidents are associated with the tritium production core design. The releases and consequences of corresponding accidents for the plutonium production core design would be less.

The source terms for the selected accidents are given in Table I.4 for the HWR, Table I.5 for the LWR, and Table I.6 for the MHTGR. The columns entitled "0-2 hours" are the source

terms during the first two hours of the accidents and were used for the maximally exposed colocated worker calculations. The columns entitled "Entire" are the source terms for the entire duration of the accidents and were used for the maximally exposed off-site individual and off-site population calculations. These source terms are for atmospheric releases. No significant liquid releases would occur during these accidents.

The source terms for H-3 (tritium) are given as the tritiated water (HTO) equivalent. Tritium released to the atmosphere during the postulated nonsevere accidents could be either in the elemental form (mostly as tritiated hydrogen, HT) or in the form of an oxide (mostly as HTO). Per unit intake of tritium into the body, HTO is about 10,000 times more effective than HT in interacting with the human body and causing radiation dose. As a result, if tritium were released as HT, the bulk of the dose would come from the HTO that resulted from oxidation of HT in the environment rather than from HT itself. A conservative upper bound estimate of the HT to HTO conversion in the environment is 10% of the released HT (Avci 1991). Therefore, when tritium would be released as HT, this 10% conversion was applied at the point of release and the remaining calculations were performed as if the release would be HTO. Accidents in which tritium would be released as HT are HWR1 (detritiation system failure) and the MHTGR accidents. Part of the tritium released during accident LWR10A (large-break LOCA) would also be HT. For the remaining accidents in Tables I.4 through I.6, tritium would be released as HTO.

Results are presented in terms of radiological impacts to the maximally exposed off-site individual, worst-sector population (worst under typical weather conditions), and maximally exposed colocated worker. Impacts presented for the maximally exposed off-site individual and the maximally exposed colocated worker are the committed effective dose equivalent (CEDE), committed dose equivalent (CDE) to the thyroid and to the organ receiving the highest CDE (the maximum organ), and lifetime fatal cancer risk. Impacts presented for the worst-sector population are the collective CEDE, collective CDE to the gonads and thyroid, cancer deaths, and genetic effects. The CEDE listed for each receptor is actually the sum of the committed effective dose equivalent due to inhalation and ingestion uptake and the effective dose equivalent due to external exposure (air immersion and ground surface exposure). The contributions to CEDE by these pathways are also provided for the maximally exposed off-site individual and the worst-sector population. The ingestion pathway was not considered for the colocated worker. Most of the colocated worker's CEDE is due to inhalation and air immersion.

The results are given in Section I.3.2.1 for Hanford, Section I.3.2.2 for INEL, and Section I.3.2.3 for SRS. A non-site-specific discussion of possible occupational doses to in-plant workers due to postulated NPR nonsevere accidents is provided in Section I.3.2.4.

Some general considerations apply across the different sites and technologies. It is apparent from the analytical results that the CEDEs and the organ CDEs for the maximally exposed off-site individuals and off-site populations are largely dominated by the ingestion pathway. The ingestion pathway doses were calculated using extremely conservative assumptions, especially for the maximally exposed off-site individual (Section I.2). For example, the individual is assumed to consume only food grown locally for one entire year after the accident, but such an assumption is probably not realistic. In addition, if an accident of the kind considered here did occur, the area around the site would be closely monitored, and emergency response personnel would advise affected persons regarding protective measures such as washing certain kinds of food items or storing them for a certain period before eating. If necessary, food from the contaminated area would be interdicted and destroyed. In any case, individuals would be advised against consuming food from the contaminated area. The CEDEs

and organ CDEs provided here for the maximally exposed off-site individual include the ingestion pathway mainly because they constitute the absolute, although somewhat unrealistic, upper limit to any individual's dose in the unlikely event that one of these accidents did occur. Another reason for providing these results is to obtain some indication regarding the necessity for implementation of protective measures. In an actual accident, emergency response authorities would take necessary steps to keep doses to off-site individuals as low as reasonably achievable. Therefore, because of these assumptions, the dose estimates provided in this section for off-site individuals are very conservative.

It is assumed that the maximally exposed colocated worker is evacuated from the site 2 hours after the initiation of an accident and does not return to the site until control is reestablished (see Section I.2). It is also assumed that the worker does not consume any contaminated food or water while on-site. Because of these assumptions, the estimated dose to the maximally exposed colocated worker is sometimes less than the estimated dose to the maximally exposed off-site individual.

I.3.2.1 Hanford Site

Potential radiological impacts to the hypothetical maximally exposed off-site individual from the selected NPR nonsevere accidents at Hanford are presented in Tables I.7 and I.8. The location proposed for the HWR and MHTGR at Hanford is different from the WNP-1 location for the LWR. The LWR location is considerably closer to the site boundary, and the maximally exposed off-site individual is therefore much closer to the LWR than to the other two reactors (5,000 m vs. 15,600 m). Because of this, the CEDEs and organ CDEs are generally higher for the LWR at Hanford. Otherwise, the differences between technologies at a given site are due solely to the differences in source terms.

Given the extremely conservative estimates calculated in this analysis (Table I.7), the only NPR nonsevere accident at Hanford that would require emergency action is LWR10A. The thyroid CDE of 21 rem (2.1×10^1 rem) is greater than 15 rem, the Food and Drug Administration protective action guide (PAG) limit for considering emergency action. If this accident occurred and if the maximally exposed off-site individual is assumed to be prevented from consuming contaminated food, the individual's CEDE would fall to about 0.04 rem and the thyroid CDE would be about 0.2. Preventive PAG limits (0.5 rem CEDE and 1.5 rem thyroid CDE) are also exceeded for HWR4 (thyroid CDE) and GCR6 (CEDE and thyroid CDE). The maximally exposed off-site individual CEDEs and thyroid CDEs for these accidents, without the ingestion pathway, would be 6.9×10^{-3} rem and 4.3×10^{-2} rem for HWR4 and 2×10^{-2} rem and 9.4×10^{-2} rem for GCR6, respectively.

Potential radiological impacts to the worst-sector population within 80 km of the site from the selected NPR nonsevere accidents are given in Tables I.9 and I.10. For all technologies, the worst sector is south-southeast of the site toward the Tri-Cities. As for the maximally exposed off-site individual, the doses to the worst-sector population are dominated by the ingestion pathway and the ingestion doses were calculated using very conservative assumptions. The differences in population doses among technologies are not as apparent for the worst-sector population as for the maximally exposed off-site individual. The reason is that the population centers are relatively farther away from the reactors than the locations of the maximally exposed off-site individuals, and the distance between the locations of reactors becomes relatively unimportant for the population compared with the off-site individuals.

Potential impacts to the maximally exposed colocated worker are given in Table I.11. All estimates are within the DOE occupational limit of 5 rem/yr. The highest estimate is for accident HWR1 (detritiation system failure in the heavy-water processing facility).

I.3.2.2 Idaho National Engineering Laboratory

Potential radiological impacts from the selected NPR nonsevere accidents at INEL are presented in Tables I.12 and I.13 for the hypothetical maximally exposed off-site individual, Tables I.14 and I.15 for the worst-sector off-site population, and Table I.16 for the maximally exposed colocated worker. The proposed location of the NPR is the same at INEL for all three technologies. The locations and sectors of the three receptors are also the same (see footnotes to Tables I.12 through I.16). The worst-sector off-site population is in the east sector toward Idaho Falls, Idaho.

Even with the extremely conservative assumptions used for this analysis, the emergency action PAG limits (5 rem CEDE and 15 rem thyroid CDE) are not exceeded by any of the postulated NPR nonsevere accidents at INEL. Preventive PAG limits (0.5 rem CEDE and 1.5 rem thyroid CDE) are exceeded for accidents HWR4, LWR10A, and GCR6. The CEDEs and thyroid CDEs to the maximally exposed off-site individuals for these accidents, without the ingestion pathway, are 1.1×10^{-2} rem and 1.1×10^{-1} rem for HWR4, 2.7×10^{-2} rem and 1.4×10^{-1} rem for LWR10A, and 3.4×10^{-2} rem and 7×10^{-2} rem for GCR6, respectively.

I.3.2.3 Savannah River Site

Potential radiological impacts from the selected NPR nonsevere accidents at SRS are presented in Tables I.17 and I.18 for the hypothetical maximally exposed off-site individual, Table I.19 and I.20 for the worst-sector off-site population, and Table I.21 for the maximally exposed colocated worker. The proposed location of the NPR is the same at SRS for all three technologies. The locations and sectors of the three receptors are also the same (see footnotes to Tables I.17 through I.21). The worst-sector off-site population is in the west-northwest sector toward Augusta, Georgia.

Even with the extremely conservative assumptions used for this analysis, the emergency action PAG limits (5 rem CEDE and 15 rem thyroid CDE) are not exceeded by any of the postulated NPR nonsevere accidents at SRS. Preventive PAG limits (0.5 rem CEDE and 1.5 rem thyroid CDE) are exceeded for accidents HWR4 and LWR10A. The CEDEs and thyroid CDEs to the maximally exposed off-site individuals for these accidents, without the ingestion pathway, are 2.9×10^{-3} rem and 3×10^{-2} rem for HWR4, 7.1×10^{-3} rem and 3.5×10^{-2} rem for LWR10A, and 8.9×10^{-3} rem and 1.8×10^{-2} rem for GCR6.

I.3.2.4 In-Plant Workers

Postulated nonsevere accidents in the proposed NPRs could have radiological impacts to in-plant workers (i.e., workers in the reactor buildings rather than the colocated workers in other buildings on the site). The magnitude of these impacts would be related to the severity of the accident and the amount of radioactive material released from the primary-system boundary into areas where workers could be exposed. Such areas are the containment, control

room, connecting hallways, and service areas. The discussion of in-plant worker impacts in this section is based on historical perspective and on detailed analyses performed for other studies (e.g., Safety Analysis Reports for commercial reactors).

No LWR or MHTGR has been used as a production reactor by DOE thus far. Therefore, any knowledge of occupational exposures in these reactors is based on commercial reactor experience. The commercial experience has been scaled and extrapolated to a facility in the production mode for this analysis.

Factors potentially influencing occupational exposures include reactor design differences, technology differences, and differences in receptors of the highest consequences from an accident. For example, some minor design differences exist between the sites for the same reactor. However, at this stage of development and at a qualitative level, these differences are not significant enough to distinguish occupational exposures to in-plant workers in a particular reactor type from one site to another. Therefore, only technology differences are discussed.

Two types of occupational exposure are related to accidents, immediate and long-term. Immediate exposure occurs at the time of the accident and during the immediate management of the emergency. Immediate exposure is prompt and not controllable. Long-term exposure, presumably at significantly lower individual rates, is associated with cleanup, refurbishment, or decommissioning of the damaged facility. Because such actions are managed by responsible authorities, the long-term exposure is kept below a predetermined amount to comply with applicable regulatory limits. In this analysis, occupational exposures from nonsevere accidents were estimated on the basis of both the immediate and long-term components.

With regard to consequences, nonsevere accidents that cause the highest consequences to in-plant workers during the immediate management of the accident are usually not the same accidents that cause the highest consequences to persons outside the facility. For example, in an LWR, accidents involving waste-gas decay tank rupture and steam generator tube rupture cause the highest immediate occupational exposures whereas the large-break LOCA causes the highest consequences to the off-site population.

Table I.22 lists the estimated upper bound occupational doses to in-plant workers from nonsevere accidents in the proposed NPRs (see Tables I.1, I.2, and I.3 for accident descriptions). For the HWR, the highest dose due to immediate exposure would result from a detritiation system failure (accident HWR1) and the highest dose due to long-term exposure would result from a charge-and-discharge mishap (accident HWR4) (Bickford 1990d). For the LWR, the highest doses due to immediate and long-term exposure would result from a waste-gas decay tank failure with 100% release of the tank contents (accident LWR2) and from a large-break LOCA (accident LWR10), respectively (Heacock 1990b). For the MHTGR, the highest doses due to both exposure components would result from a depressurized conduction cooldown (accident GCR6) (Luptak 1990). Because there has been relatively little experience with an MHTGR in this regard, the uncertainty associated with the estimates provided in Table I.22 is relatively large.

The occupational exposures following nonsevere accidents are very design-specific and facility-dependent. Therefore, there is considerable uncertainty associated with the values given in Table I.22. They are given only to provide some indication of the levels of exposure that would be expected in the case of a nonsevere accident in the proposed NPRs.

I.4 RADIOLOGICAL IMPACTS OF POSTULATED NONSEVERE ACCIDENTS AT SUPPORT FACILITIES

I.4.1 Introduction

A number of nonsevere accidents that could result in atmospheric releases of radioactive material to the off-site environment have been postulated for NPR support facilities (see Appendixes A, B, and C for descriptions of support facilities and their functions). No accidents with liquid releases from support facilities have been postulated because the buildings housing these facilities would be designed to contain liquids. (One accident has been postulated at SRS that assumes a truck crashes into and ruptures a solvent tank near a fuel processing facility, resulting in liquid releases off-site.) The types of postulated accidents range from inadvertent criticalities, fires, and explosions to earthquakes and tornadoes. The extent to which radioactive material would be released to the environment in the event of such accidents depends on a number of factors, including the inventory and volatility of the radioactive material and the operating status or efficiency of the systems that filter out these materials. Not all types of accidents are possible at each type of support facility. For example, criticality accidents, which have the highest radiological consequences at fuel fabrication facilities, are not possible at tritium-target processing facilities because of the absence of fissile material. Also, some types of accidents are restricted to only one reactor technology. For example, a Zircaloy hull fire accident is specific to the LWR technology. The nomenclature developed for identifying the postulated accidents is explained in Section I.4.1.1. The accidents are described briefly in Section I.4.1.2.

Because detailed facility designs and safety analyses were not available for the NPR support facilities, a list of postulated nonsevere accidents was selected from a review of Safety Analysis Reports and other reports that address the hazards of operations similar to those to be utilized in NPR support facilities for a given reactor technology. Based on information gathered from this review, a broad range of postulated accidents was identified for each facility. Each of the postulated accidents has an initiator -- that is, a human error, equipment failure, or natural phenomenon. An initiator, alone or combined with other events, acting together with the failure of accident prevention or mitigation mechanisms could lead to release of radioactive material to the environment. A subset of these postulated accidents that could provide a reasonable and credible upper bound estimate of the maximum radionuclide release during a nonsevere accident at a support facility was chosen for analysis at each alternative site for each reactor technology. Generally, postulated accidents chosen for analysis have large radionuclide inventories at risk and either direct release paths or increased potential for release. Values of parameters that affect the amount of radioactive material released to the environment were conservatively chosen in accordance with information in the DOE accident analysis guide for nonreactor nuclear facilities (Elder et al. 1986) or NRC regulatory guides.

The frequency of nonsevere support facility accidents are classified as anticipated, unlikely, and extremely unlikely (Elder et al. 1986). Accidents in the "anticipated" frequency class may be expected to occur once or more during the lifetime of the facility; accidents in the "unlikely" frequency class are not expected but may occur during the lifetime of the facility; and accidents in the "extremely unlikely" frequency class will probably not occur during the lifetime of the facility.

I.4.1.1 Identification of Accidents

Each nonsevere accident was assigned a five-character identifier. These characters define the site, reactor technology, generic type of support facility where the accident is postulated to occur, type of accident, and sequence number of the accident. The first character defines the site where the accident would occur and is either H for Hanford, I for INEL, or S for SRS. The second character defines the reactor technology involved in the accident and is either H for HWR, L for LWR, or M for MHTGR. The third character defines the type of support facility; it is either 1 for fuel fabrication facilities, 2 for fuel processing facilities, 3 for plutonium-target fabrication facilities, 4 for plutonium-target processing facilities, 5 for tritium-target fabrication facilities, or 6 for tritium-target processing facilities. Fabrication facilities are facilities at which reactor fuels or targets are manufactured and assembled prior to irradiation in the NPR; processing facilities are facilities at which fuels or targets are handled subsequent to irradiation to recover tritium, plutonium, or recoverable uranium. No accidents involving release of radioactive material have been postulated at tritium-target fabrication facilities because significant quantities of radioactive material are not expected to be present in these facilities. The fourth character defines the kind of accident; it is either B for break, C for criticality, D for deflagration, E for explosion, F for fire, L for leak, R for rupture, S for seismic (earthquake), W for weather (tornado), or N for none of the above. The fifth character defines the sequence number of a particular kind of accident at a support facility for a reactor technology at an alternative site. For example, HH1C1 is criticality accident number 1 in the fuel fabrication facility supporting an HWR at Hanford, and SL6D1 is deflagration accident number 1 in the tritium-target processing facility supporting an LWR at SRS.

I.4.1.2 Description of Accidents

The postulated bounding nonsevere accidents (those with the highest consequences) for each type of support facility associated with the three reactor technologies are listed in Tables I.23, I.24, and I.25 for Hanford, INEL, and SRS, respectively. This section provides a brief generic description of the postulated accidents listed in these tables, as well as other postulated nonsevere accidents that have been analyzed but found to be less impacting. The source terms and results for the accidents listed in Tables I.23, I.24, and I.25 are provided in Sections I.4.2 and I.4.3. A separate document (Kier and Avci 1991) provides the source terms and results for all postulated nonsevere accidents analyzed. All the accidents described herein have been analyzed but not for all technology and site combinations. Instead, a representative subset of the postulated accidents has been analyzed for each such combination.

The postulated nonsevere accidents with the highest consequences can be grouped under four general headings: criticalities, 30-day cooled fuel or target processing, tritium deflagration, and Zircaloy hull fire. Other postulated accidents include other types of fires, explosions, equipment failures, and so forth. For more detailed descriptions of these accidents, the following references should be consulted: WHC (1989a-b, 1990b) for Hanford; EG&G (1989a-c) for INEL; and WSRC (1991b-d) for SRS.

Postulated Accidents with the Highest Consequences

Criticalities. Whenever fissile materials are handled in significant quantities, the potential exists for sufficient quantities of these materials to be in such close proximity that an unintended nuclear chain reaction (criticality) can occur. A number of scenarios have been postulated to result in bounding criticality accidents. For example, a criticality accident for the HWR fuel fabrication facility would most likely result from inadvertent dissolution of fuel tubes during cleaning operations or from inadvertent spill of uranium-aluminum machining chips from an approved scrap/cyclone can into a large container of moderator (WSRC 1991b). A criticality accident for the LWR fuel fabrication facility could be caused by an accumulation of uranium oxide powder at a single location.

Each of these criticality accidents is assumed to result in 10^{19} fissions before being terminated. Because support facilities, unlike reactors, do not have means of controlling a nuclear chain reaction, the reaction would either terminate when the critical mass disassembles from the energy generated by the accident or as the result of inherent negative feedback mechanisms.

30-Day Cooled Fuel or Target Processing. The 30-day cooled fuel, or short-cooled fuel, type of accident is postulated during fuel and plutonium-target processing. Fission products and other radioactive material created in the nuclear fuel and plutonium targets during irradiation in the NPR have a broad range of half-lives, from fractions of a second to thousands of years. To reduce the radioactivity of fuel and plutonium targets before they undergo disassembly during processing, there would be a delay of 1 year between discharge from the reactor and disassembly and processing, which would allow time for the fuel and plutonium targets to cool (i.e., allow short-lived radionuclides to decay). For this type of postulated accident, a series of administrative errors would have to occur that would result in fuel or targets being taken to the disassembly process after being cooled for only 30 days. A 30-day minimum appears to be a reasonable "worst-case" estimate of the time that would elapse before used fuel or targets could be disassembled and sheared at a processing facility based on the requirements for (1) fuel and target preparation and packaging at the reactor prior to shipment to the reprocessing plant, (2) fuel handling logistics at both facilities, and (3) cask handling requirements.

Tritium Deflagration. A tritium-target processing facility would receive tritium (irradiated lithium) targets shipped from an NPR in casks. At the processing facility, the tritium would be removed from the targets and purified for shipment off-site. The tritium targets would be processed in gloveboxes in an oxygen-free or oxygen-deficient environment at subatmospheric pressures. A postulated deflagration accident could occur if an operator failed to remove tritium from the equipment before opening a glovebox for maintenance and an ignition source were present (e.g., air leaking into the glovebox from the rupture of a line to a pump).

Zircaloy Hull Fire. The LWR fuel rods would have hulls, or cladding, made of Zircaloy-4, a pyrophoric metal. For this type of accident, it is assumed that after the cladding had been removed from the fuel during fuel processing, Zircaloy-4 chips would ignite to release uranium, plutonium, and activation products of Zircaloy-4. (Zircaloy-4 chips are produced during shearing of spent fuel rods and subsequent handling of empty hulls after the fuel is leached.)

Other Postulated Accidents

Interim Waste Tank Explosion. During fuel processing, fission products and actinides would be stored in interim storage tanks. These tanks would be continuously vented to preclude the accumulation of explosive mixtures of hydrogen and air. For this type of postulated accident to occur, the venting would have to fail for a sufficient duration to cause an explosion, resulting in the release of radioactive material through the facility's ventilation system.

Red Oil Explosion. Red oil, the organic degradation product of the combination of solvent and hot nitric acid, is unstable and can explode at temperatures above 130°C. Red oil would be formed, for example, during the process in which uranium is purified for recycling to fuel fabrication. In this postulated accident, the oil explodes in a concentrator in the uranium purification process, and uranium isotopes are released to the environment after passing through HEPA filters.

Double-Batched Burner Rupture. For the MHTGR technology, fuel and plutonium targets would be crushed and burned in a furnace during processing. The postulated accident assumes that because of high temperatures and/or failure of materials of construction, the integrity of the furnace is breached when it contains a double batch of fuel or plutonium targets. A fire or explosion could result, and radioactive material could be released.

Sintering Furnace Explosions. For the LWR technology, uranium oxide powder would be pelletized, and the pellets would be sintered in a reducing atmosphere in a furnace at the fuel fabrication facility. In the postulated accident, air leaks into the furnace, and a detonable mixture of hydrogen and oxygen explodes with the subsequent release of uranium particles through a ventilation system.

Other Explosions. Other accidents involving explosions have been postulated and analyzed. At Hanford and INEL, such accidents could involve a hydrogen explosion in the concentrator during processing of fuel and plutonium targets.

1A Column Fire. The 1A column is used in a decontamination cell where fission products are separated from plutonium and uranium during fuel processing. In this postulated accident, a leak in the column is assumed to result in combustion of the solution and in release of radioactive material through a ventilation exhaust.

Scrap Fire. For the LWR technology, combustion of scrap and waste (e.g., improperly packaged Zircaloy-4 chips) at the fuel fabrication facility could cause a fire. Fuel rods could be dispersed among the ignited waste and cause the release of radioactive material. If this occurred, the release would be in the form of particulates through a HEPA filter.

Other Fires. Several other kinds of fires have been postulated and analyzed. They include solvent extraction fires during fuel processing and plutonium-target processing, fires and explosions during plutonium-target fabrication, and furnace carbon fires during tritium-target processing. In the tritium-target processing facilities, tritium targets would be crushed and then heated in an evacuated furnace in a hot cell to extract their tritium. The postulated accident assumes that the furnace contents are exposed to air and burned while the detritiation system for the hot cell is inoperable.

HEPA Filter Failure. The purpose of a HEPA filter is to filter out particulate matter (including radioactive particulates) in a facility atmosphere so that it will not reach the outside environment. A HEPA filter does not usually fail completely but rather develops holes and tears that release only small amounts of radioactive material. However, in the HEPA filter failure accidents postulated for this EIS, it is assumed that the filter fails completely.

Uranium Hexafluoride Releases. For the LWR technology, uranium hexafluoride (UF_6) would be received in cylindrical containers at a uranium processing facility for conversion to uranium dioxide. The postulated accidents involve the release of UF_6 to the environment from assumed breaches in the containers.

Monorail Hoist Failure. Monorails for moving equipment would be used in some gloveboxes in which tritium targets are processed. A mechanical failure of the monorail could cause the hoist to fail, which could result in the breaking of a process line, cracking of a glovebox panel, or opening of a gloveport.

Glovebox Spill. At tritium-target processing facilities, a postulated accident involves a spill of tritium to a glovebox while the tritium containment system is unavailable.

Leak during Maintenance. A number of accidents have been postulated that involve leaks during maintenance activities at tritium-target processing facilities. An initiating event for such an accident would be failure to remove tritium from equipment before opening a glovebox for maintenance. The types of equipment failures accompanying the initiating event could be glovebox leaks, pipe leaks, zeolite bed leaks, solid waste leaks, and liquid waste leaks.

Canister Accidents. Tritium targets would be contained in canisters (target canisters) when they are moved to a glovebox for processing. After processing, the extracted tritium would be moved in transfer canisters from a glovebox to a shipping cask. If the canisters were accidentally dropped or otherwise broken, tritium could be released.

Other Postulated Accidents for the HWR Technology at SRS Support Facilities

There have been incidents in the past at SRS, where no accident resulted, in which radioactive liquid materials from fuel processing have been transferred erroneously outside of

a facility. Therefore, because existing facilities will be utilized, other types of accidents are postulated for the HWR technology at SRS support facilities. For example, in one postulated accident, it is assumed that during fuel processing, process material is transferred erroneously to an outside chemical storage tank, the tank fails, and liquid containing radioactive material is released to the atmosphere through evaporation. A similar accident is postulated during plutonium-target processing. Earthquake-initiated accidents that have a severity of either VIII or IX on the Modified Mercalli Intensity Scale have been postulated at existing HWR technology support facilities. (New facilities at SRS and NPR facilities at Hanford and INEL will be built to withstand such earthquakes.) In these accidents, the earthquake could cause either partial destruction of the building or local overstressing of equipment so that part of the inventory of radioactive materials could be released. At the tritium-target processing facility for the HWR technology, a tornado of Fujita Intensity Four is postulated to release part of the tritium inventory.

I.4.2 Source Terms for Highest Consequence Nonsevere Accidents

Estimated releases from the highest consequence nonsevere accidents at support facilities are presented in Sections I.4.2.1, I.4.2.2, and I.4.2.3 for the Hanford Site, INEL, and SRS, respectively. For plutonium production in the LWR, fuel/target assemblies would be purchased from off-site commercial fabricators, and consequently there would be no plutonium-target fabrication facility on-site. For the other two technologies, the consequences of postulated accidents in plutonium-target fabrication facilities have been found to be negligible compared with consequences of accidents postulated for other types of support facilities. Therefore, source terms and radiological impacts for nonsevere accidents at plutonium-target fabrication facilities are not discussed in this section. However, they are discussed in a backup document (Kier and Avci 1991).

For each site and reactor technology, the source terms for releases from criticality accidents at fuel fabrication facilities were taken from NRC's Regulatory Guide 3.34, Rev. 1 (NRC 1979), and the source term for the highest consequence accident at a tritium-target processing facility, a tritium deflagration, is taken to be 1×10^6 Ci of tritium in the form of HTO (Cornwell 1990). The tritium deflagration release is based on historical data contained in SARs for similar facilities at SRS. All releases are assumed to be gaseous or particulate releases to the atmosphere.

I.4.2.1 Hanford Site

Source terms for the highest consequence nonsevere accidents at Hanford are presented in Tables I.26, I.27, and I.28 for HWR, LWR, and MHTGR support facilities, respectively. The postulated accidents are identified in Table I.23. For the HWR and MHTGR fuel processing and plutonium-target processing facilities, different accidents have the highest consequences for different receptors. The highest consequence accidents are the processing of short-cooled fuel or targets for the maximally exposed off-site individual and worst-sector population and a criticality for the maximally exposed colocated worker.

I.4.2.2 Idaho National Engineering Laboratory

Source terms for the highest consequence nonsevere accidents at INEL are presented in Tables I.29, I.30, and I.31 for HWR, LWR, and MHTGR support facilities, respectively. The postulated accidents are identified in Table I.24. For the MHTGR fuel fabrication and plutonium-target processing facilities and for the HWR plutonium-target processing facility, the accidents with the highest consequences are the processing of short-cooled fuel or targets for the maximally exposed off-site individual and worst-sector population and a criticality for the maximally exposed colocated worker. For HWR fuel processing, the highest consequence accidents are the processing of short-cooled fuel for the maximally exposed off-site individual and a criticality for the worst-sector population and maximally exposed colocated worker.

I.4.2.3 Savannah River Site

Source terms for the highest consequence nonsevere accidents at SRS are presented in Tables I.32, I.33, and I.34 for HWR, LWR, and MHTGR support facilities, respectively. The postulated accidents are identified in Table I.25. For LWR fuel processing, the highest consequence accidents are a Zircaloy hull fire for the maximally exposed off-site individual and worst-sector population and a criticality for the maximally exposed colocated worker.

I.4.3 Results of Analyses

The analytical results of postulated nonsevere accidents at support facilities are presented in Sections I.4.3.1, I.4.3.2, and I.4.3.3 for the Hanford Site, INEL, and SRS, respectively. Estimated radiological impacts are presented for the maximally exposed off-site individual, worst-sector population, and maximally exposed colocated worker for the highest consequence accident during fuel fabrication, fuel processing, plutonium-target processing, and tritium-target processing. Impacts as well as source terms for all postulated accidents are presented in a separate document (Kier and Avci 1991). Impacts presented for the maximally exposed off-site individual and maximally exposed colocated worker are the CEDE, CDE to the thyroid and maximally exposed organ, and incremental lifetime fatal cancer risk. Impacts presented for the worst-sector population are the collective CEDE, collective CDE to the gonads and thyroid, incremental number of cancer deaths, and incremental number of genetic effects. Also, doses by pathway are presented for the maximally exposed off-site individual and worst-sector population. Doses by pathway are important for assessing the impact of protective actions that might be recommended by off-site emergency response authorities to mitigate doses from the ingestion pathway or for determining whether to recommend evacuation or sheltering of the off-site population to mitigate doses from the radioactive plume. Protective action guides used by off-site emergency response authorities in deciding whether to recommend protective actions are discussed in Section 5.1.7.2.

I.4.3.1 Hanford Site

The estimated consequences of the highest consequence accidents for support facilities at Hanford are presented in Tables I.35 through I.39. For the maximally exposed off-site individual (Table I.35), the estimated CEDEs from the tritium deflagration accidents (HH6D1, HL6D1, HM6D1) and the estimated maximally exposed organ doses from the tritium

deflagration accidents, the short-cooled fuel accident at the LWR fuel processing facility (HL2N2), and the short-cooled target accident at the HWR tritium-target processing facility (HH4N1) range from 4.3 to 7.0 rem. These doses are primarily from the ingestion pathway (Table I.36). For the tritium deflagration accidents, these estimates indicate, therefore, that the FDA preventive PAG of 0.5 rem for the CEDE and organs other than the thyroid would be exceeded. For the short-cooled fuel accident (HL2N2) and short-cooled target accident (HH4N1), these estimates indicate that the FDA preventive PAG of 1.5 rem for the thyroid would be exceeded. For the tritium deflagration accidents, the estimated number of cancer deaths and genetic effects are 30 and 20, respectively. If protective actions were to eliminate exposure from the ingestion pathway, the estimated number of cancer deaths would be reduced to 1 and no genetic effects would be expected.

I.4.3.2 Idaho National Engineering Laboratory

The estimated impacts from the highest consequence accidents for support facilities at INEL are presented in Tables I.40 through I.44. For the maximally exposed off-site individual (Table I.40), the estimated CEDEs and maximally exposed organ doses from the tritium deflagration accidents (IH6D1, IL6D1, IM6D1) range from 2.1 to 2.5 rem. These doses are primarily from the ingestion pathway (Table I.41). For the tritium deflagration accidents, these estimates indicate, therefore, that the FDA preventive PAG of 0.5 rem for the CEDE and organs other than the thyroid would be exceeded. The estimated number of cancer deaths and genetic effects for these accidents are 2 and 1, respectively (Table I.42). If protective actions were used to eliminate exposure from the ingestion pathway, no cancer deaths or genetic effects would be expected.

I.4.3.3 Savannah River Site

The estimated impacts of the highest consequence accidents for support facilities at SRS are presented in Tables I.45 through I.49. For the estimated doses to the maximally exposed off-site individual (Tables I.45 and I.46), the estimated doses from several accidents exceed FDA ingestion pathway PAGs. For the tritium deflagration accidents (SH6D1, SL6D1, and SM6D1), the estimated CEDEs and maximally exposed organ doses exceed the FDA preventive PAG of 0.5 rem for the CEDE and organs other than the thyroid. For the criticality accidents at fuel fabrication facilities (SH1C1, SL1C1, and SM1C1) and the short-cooled fuel accident at the LWR fuel processing facility (SL2N2), the estimated thyroid dose exceeds the FDA preventive PAG of 1.5 rem for the thyroid.

The estimated dose to the thyroid of the maximally exposed off-site individual from the short-cooled fuel accident at the HWR fuel processing facility (SH2N1) is 23 rem, which exceeds the FDA emergency PAG of 15 rem for the thyroid. This dose is primarily from the ingestion pathway (Table I.46). For the tritium deflagration accidents, the estimated number of cancer deaths and genetic effects are 6 and 4, respectively (Table I.47). If protective actions were taken to eliminate exposure from the ingestion pathway, no cancer deaths or genetic effects would be expected.

I.4.3.4 In-Plant Workers

Nonsevere accidents at NPR support facilities would give radiological impacts to in-plant workers as well as to colocated workers and the general public off-site. Detailed designs are needed for quantitative estimates of these impacts. For example, one key parameter governing the exposure of workers to a criticality accident is their distance from the critical mass, which depends on the design of the facility. The design of most support facilities is not sufficiently detailed at this time to permit quantitative estimates; therefore, radiological impacts to in-plant workers are discussed qualitatively in this EIS. One exception is HWR fuel fabrication at SRS, which would use an existing facility for which it has been estimated that an average of 2.7 operating personnel would be expected to receive a fatal radiation dose from a criticality incident; from 0 to 8 persons would receive fatal doses, depending on the location of the accident (WSRC 1991b).

The NPR support facility functions include fuel and target fabrication and fuel and target processing. Fabrication operations occur prior to irradiation in the reactor and therefore involve materials with substantially lower inventories of radioactive material than processing operations, which occur subsequent to irradiation in the reactor. In fuel and plutonium-target processing, uranium and plutonium are separated and fission products are removed. Operations involving fission products are performed within thick-walled (e.g., concrete) gamma-ray shielding. Operations involving enriched uranium and plutonium, both fabrication and processing, are also typically performed within a shielded area to protect in-plant workers from radioactive contamination.

In general, the types of accidents with the highest radiological consequences to in-plant workers are criticalities and explosions. The NPR design requirements will minimize the frequency of accidents as well as worker exposures and number of workers affected. However, explosions and criticalities at NPR support facilities are possible and have been postulated for this EIS.

Documented criticality accidents at nuclear fuel cycle facilities have been reviewed by Knief (1985). All of these accidents occurred before 1980 in facilities that have since been shut down or extensively redesigned to preclude the type of accident that occurred. Although these specific criticality accidents may not be possible at NPR support facilities, they indicate how criticality accidents have occurred in the past and what their radiological impacts to in-plant workers have been.

All eight documented criticality accidents occurred in aqueous solutions (five involved highly enriched uranium and three involved plutonium). Two of these accidents resulted in fatalities, one fatality per accident. One of the fatal accidents occurred in 1964 at the Wood River Junction Plant, a commercial nuclear facility in Rhode Island, when a solution containing highly enriched uranium was poured into a tank. The other fatal accident occurred in 1958 at the Los Alamos Scientific Laboratory (now Los Alamos National Laboratory) when plutonium became concentrated in a solvent layer in a tank.

Three criticality accidents occurred in the Idaho Chemical Processing Plant (ICPP) at the National Reactor Testing Station (now INEL). Only the first of these accidents resulted in significant doses to in-plant workers (one was estimated to have received 50 rad, another 32 rad). This accident, which occurred in 1959, resulted when a highly enriched uranium solution was

inadvertently siphoned from a bank of geometrically favorable* storage tanks to a process equipment waste tank, where the criticality occurred. In 1962, a criticality accident occurred at Hanford in the original Recuplex Plant, which was used to recover plutonium from various processes conducted at Hanford. The accident took place in a solvent extraction area that was enclosed in a room-sized glovebox. During a general cleanup operation, concentrated plutonium solution is thought to have overflowed from a geometrically favorable tank to a sump, where it collected and then was sucked into a vessel where the criticality occurred. Three workers received doses ranging from 16 to 87 rad. Other documented criticality accidents occurred at the Y-12 Plant at Oak Ridge, Tennessee, and at Windscale in the United Kingdom.

From the doses received by in-plant workers in the ICPP accidents, it has been estimated that for a reference solution criticality of 1×10^{17} fissions, the threshold distance from the solution undergoing criticality for a lethal dose (50% of those exposed dying within 30 days of exposure) is approximately 4 m; the threshold distance for detectable medical effects (an exposure of approximately 25 rad) is about 20 m.

The historical data presented above are from facilities that were built many years ago with design requirements that were much different than those which exist today. Under the current regulatory climate, all NPR facilities will be designed, built, and operated to comply with stringent safety requirements. As a result, these requirements will preclude most potential accidents, and in-plant worker safety at the NPR facilities should be much better than the historical data indicate.

*Geometrically favorable means a shape for which leakage of neutrons is so high that criticality is precluded.

TABLE I.1 Postulated HWR Nonsevere Accidents

Accident ID	Frequency Class ^a	Description
HWR1	Limiting	Detritiation system failure in the heavy water processing facility
HWR2	Limiting	Large-break loss-of-coolant accident for tritium- and plutonium-producing core
HWR3	Infrequent	Heat exchanger tube rupture
HWR4	Infrequent	Charge-and-discharge mishap with tritium-producing charge
HWR5	Infrequent	Charge-and-discharge mishap with plutonium-producing charge

^aSee Section I.3.1 for definition of frequency classes.

Sources: Section 5.3 of WSRC 1989a; Bickford 1990a-c.

TABLE I.2 Postulated LWR Nonsevere Accidents

Accident ID	Frequency Class ^a	Description
LWR1	Infrequent	Equipment leakage or malfunction resulting in release of 25% of the normal inventory of a waste-gas decay tank
LWR2	Infrequent	Equipment leakage or malfunction resulting in release of 100% of the normal inventory of a waste-gas decay tank
LWR3	Infrequent	Off-design transients that induce fuel failures above those expected and steam generator leak combined with station blackout
LWR4	Infrequent	Steam generator tube rupture
LWR5	Infrequent	Refueling accident - fuel assembly drop
LWR6	Infrequent	Refueling accident - heavy object drop onto fuel in core
LWR7	Infrequent	Spent fuel handling accident - fuel assembly drop into fuel storage pool
LWR8	Infrequent	Spent fuel handling accident - heavy object drop onto fuel rack
LWR9	Limiting	Loss-of-coolant accident - small pipe break
LWR10	Limiting	Loss-of-coolant accident - large pipe break
LWR11	Limiting	Loss-of-coolant accident - break in letdown line from primary system that penetrates the containment
LWR12	Limiting	Rod ejection accident
LWR13	Limiting	Steam-line break outside containment

^aSee Section I.3.1 for definition of frequency classes.

Source: Section 5.1 of WHC 1990a.

TABLE I.3 Postulated MHTGR Nonsevere Accidents

Accident ID	Frequency Class ^a	Description
GCR1	Moderate	Small primary break with normal containment leak rate of 1% per day.
GCR2	Infrequent	Moderate primary break with normal containment leak rate of 1% per day.
GCR3	Limiting	Small primary break with containment bypass leak rate of 100% per day.
GCR4	Limiting	Moderate primary break with containment bypass leak rate of 100% per day.
GCR5	Limiting	Depressurized conduction cooldown with normal containment leak rate of 1% per day.
GCR6	Limiting	Depressurized conduction cooldown with containment bypass leak rate of 100% per day followed by normal containment leak rate of 1% per day.

^aSee Section I.3.1 for definition of frequency classes.

Source: Bowman 1990c.

TABLE I.4 Source Terms for Selected HWR Nonsevere Accidents

Radio-nuclide	HWR1 ^a (Ci)		HWR2 (Ci)		HWR4 (Ci)	
	0-2 hours	Entire	0-2 hours	Entire	0-2 hours	Entire
H-3 ^b	3.0×10^4	3.0×10^4	6.4×10^1	2.3×10^4	1.6×10^3	2.2×10^3
Br-83	-	-	-	-	4.8×10^{-2}	4.8×10^{-2}
Br-84	-	-	-	-	6.2×10^{-8}	6.2×10^{-8}
Kr-83m	-	-	-	-	1.4	1.4
Kr-85	-	-	-	-	6.6	9.2
Kr-85m	-	-	-	-	8.3×10^1	8.3×10^1
Kr-87	-	-	-	-	6.9×10^{-1}	6.9×10^{-1}
Kr-88	-	-	-	-	6.3×10^1	6.3×10^1
Rb-86	-	-	-	-	4.7×10^{-3}	9.7×10^{-2}
Xe-131m	-	-	-	-	1.1×10^1	1.3×10^1
Xe-133	-	-	-	-	3.5×10^3	3.7×10^3
Xe-133m	-	-	-	-	4.3×10^2	4.3×10^2
Xe-135	-	-	-	-	1.1×10^2	1.1×10^2
I-129	-	-	-	-	5.2×10^{-8}	2.1×10^{-6}
I-131	-	-	1.7×10^{-4}	2.2×10^{-2}	7.9	7.9×10^1
I-132	-	-	-	-	1.8×10^{-1}	1.8×10^{-1}
I-133	-	-	-	-	1.2×10^1	1.4×10^1
I-134	-	-	-	-	3.4×10^{-4}	3.4×10^{-4}
I-135	-	-	-	-	4.2	4.2
Cs-134	-	-	-	-	1.4×10^{-1}	5.7
Cs-136	-	-	-	-	4.6×10^{-2}	7.4×10^{-1}
Cs-137	-	-	-	-	1.3×10^{-1}	5.5
Cs-138	-	-	-	-	1.4×10^{-7}	1.4×10^{-7}
Cs-139	-	-	-	-	-	-

^aSee Table I.1 for accident descriptions.^bHTO equivalent.

Sources: Section 5.3 of WSRC 1989a; Bickford 1990a-c.

TABLE I.5 Source Terms for Selected LWR Nonsevere Accidents

Radio-nuclide	LWR2A (Ci)		LWR4A (Ci)		LWR10A (Ci)	
	0-2 hours	Entire	0-2 hours	Entire	0-2 hours	Entire
H-3 ^b	0	0	1.4×10^3	1.4×10^3	1.1×10^3	1.7×10^5
Kr-83m	0	0	0	0	1.1×10^{-1}	9.0×10^{-1}
Kr-85m	2.9×10^2	2.9×10^2	3.1×10^1	3.1×10^1	4.9×10^{-1}	7.1
Kr-85	1.8×10^4	1.8×10^4	4.5×10^1	4.5×10^1	2.3×10^{-1}	5.2×10^1
Kr-87	2.4×10^2	2.4×10^2	3.1×10^1	3.1×10^1	3.7×10^{-1}	2.6
Kr-88	4.9×10^2	4.9×10^2	5.8×10^1	5.8×10^1	1.0	1.1×10^1
I-131	4.0	4.0	5.5	5.5	1.9	7.6×10^1
I-132	1.6	1.6	2.6×10^1	2.6×10^1	2.7×10^{-1}	1.6
I-133	2.7	2.7	2.1×10^1	2.1×10^1	1.4	2.2×10^1
I-134	2.0	2.0	3.7×10^1	3.7×10^1	2.3×10^{-1}	1.3
I-135	2.4	2.4	3.2×10^1	3.2×10^1	7.2×10^{-1}	7.1
Xe-131m	2.8×10^4	2.8×10^4	2.0×10^2	2.0×10^2	1.3×10^{-1}	1.0×10^1
Xe-133m	5.0×10^2	5.0×10^2	1.5×10^1	1.5×10^1	3.1×10^{-1}	1.7×10^1
Xe-133	3.3×10^4	3.3×10^4	4.9×10^2	4.9×10^2	1.6×10^1	1.5×10^3
Xe-135m	3.5×10^2	3.5×10^2	1.5×10^1	1.5×10^1	2.4×10^{-1}	5.2
Xe-135	3.8×10^3	3.8×10^3	2.8×10^2	2.8×10^2	1.1	3.1×10^1
Xe-138	1.5×10^2	1.5×10^2	2.0×10^1	2.0×10^1	1.2×10^{-1}	5.8×10^{-1}

^aSee Table I.2 for accident descriptions.

^bHTO equivalent.

Sources: WHC 1990a; Heacock 1990a; Schultz 1990.

TABLE I.6 Source Terms for Selected MHTGR Nonsevere Accidents

Radio-nuclide	GCR1 ^a (Ci)		GCR2 (Ci)		GCR6 (Ci)	
	0-2 hours	Entire	0-2 hours	Entire	0-2 hours	Entire
H-3 ^b	1.9×10^{-3}	2.3	1.8×10^{-1}	2.3	1.8×10^{-1}	7.2×10^4
Kr-85m	4.3×10^{-3}	1.7×10^{-2}	4.1×10^{-1}	1.3	4.1×10^{-1}	1.3
Kr-85	1.2×10^{-5}	1.4×10^{-2}	1.1×10^{-3}	1.4×10^{-2}	1.1×10^{-3}	1.4×10^{-2}
Kr-87	6.2×10^{-3}	9.6×10^{-3}	6.0×10^{-1}	8.9×10^{-1}	6.0×10^{-1}	8.9×10^{-1}
Kr-88	1.2×10^{-2}	3.1×10^{-2}	1.1	2.7	1.1	2.7
Kr-89	2.2×10^{-4}	2.2×10^{-4}	2.2×10^{-2}	2.2×10^{-2}	2.2×10^{-2}	2.2×10^{-2}
Kr-90	1.6×10^{-5}	1.6×10^{-5}	1.6×10^{-3}	1.6×10^{-3}	1.6×10^{-3}	1.6×10^{-3}
Rb-86	1.7×10^{-6}	1.6×10^{-5}	1.6×10^{-4}	9.3×10^{-4}	2.9×10^{-1}	1.6
Sr-89	1.2×10^{-6}	1.1×10^{-5}	1.1×10^{-4}	6.5×10^{-4}	5.3×10^{-1}	2.4×10^1
Sr-90	2.3×10^{-9}	2.3×10^{-8}	2.2×10^{-7}	1.3×10^{-6}	2.2×10^{-1}	2.6
Sr-91	2.8×10^{-9}	1.3×10^{-8}	2.7×10^{-7}	9.6×10^{-7}	1.0×10^{-5}	3.2×10^{-2}
Y-90	2.8×10^{-11}	2.3×10^{-10}	2.7×10^{-9}	1.4×10^{-8}	6.7×10^{-7}	5.6×10^{-1}
Y-91	2.3×10^{-11}	2.2×10^{-10}	2.2×10^{-9}	1.3×10^{-8}	2.3×10^{-5}	2.7×10^1
Zr-95	2.3×10^{-11}	2.2×10^{-10}	2.2×10^{-9}	1.3×10^{-8}	1.3×10^{-5}	3.0×10^1
Zr-97	1.9×10^{-9}	1.1×10^{-8}	1.8×10^{-7}	7.5×10^{-7}	1.2×10^{-5}	5.9×10^{-1}
Nb-95	4.2×10^{-11}	4.0×10^{-10}	4.0×10^{-9}	2.3×10^{-8}	2.7×10^{-5}	2.9×10^1
Mo-99	5.1×10^{-10}	4.2×10^{-9}	4.9×10^{-8}	2.6×10^{-7}	1.3×10^{-5}	1.1×10^1
Tc-99m	4.3×10^{-9}	1.5×10^{-8}	4.1×10^{-7}	1.2×10^{-6}	2.0×10^{-5}	8.9×10^{-4}
Ru-103	2.2×10^{-11}	2.1×10^{-10}	2.1×10^{-9}	1.2×10^{-8}	7.7×10^{-6}	1.7×10^1
Ru-105	1.8×10^{-9}	5.3×10^{-9}	1.8×10^{-7}	4.4×10^{-7}	3.2×10^{-6}	1.5×10^{-5}
Ru-106	2.1×10^{-13}	2.0×10^{-12}	2.0×10^{-11}	1.2×10^{-10}	7.0×10^{-7}	1.2
Rh-105	2.4×10^{-10}	1.8×10^{-9}	2.3×10^{-8}	1.1×10^{-7}	6.6×10^{-6}	1.1
Ag-110m	2.3×10^{-15}	2.2×10^{-14}	2.2×10^{-13}	1.3×10^{-12}	5.1×10^{-9}	1.0×10^{-2}
Sb-127	1.2×10^{-11}	1.0×10^{-10}	1.2×10^{-9}	6.3×10^{-9}	4.2×10^{-7}	4.8×10^{-1}
Sb-129	4.5×10^{-10}	4.5×10^{-10}	4.5×10^{-8}	4.5×10^{-8}	5.3×10^{-8}	5.3×10^{-8}
Te-127m	1.4×10^{-7}	1.4×10^{-6}	1.4×10^{-5}	8.0×10^{-5}	1.4×10^{-1}	8.1×10^{-1}
Te-127	4.7×10^{-7}	4.7×10^{-6}	4.5×10^{-5}	2.7×10^{-4}	1.7	9.9
Te-129m	1.2×10^{-6}	1.2×10^{-5}	1.2×10^{-4}	6.8×10^{-4}	3.7×10^{-1}	2.1
Te-129	1.1×10^{-4}	1.5×10^{-4}	1.0×10^{-2}	1.4×10^{-2}	5.7×10^{-2}	7.6×10^{-2}
Te-131m	2.1×10^{-5}	1.5×10^{-4}	2.0×10^{-3}	9.5×10^{-3}	2.5×10^{-1}	1.2
Te-132	4.8×10^{-5}	4.6×10^{-4}	4.6×10^{-3}	2.6×10^{-2}	1.4×10^1	8.0×10^1
I-131	6.7×10^{-5}	6.1×10^{-4}	6.4×10^{-3}	3.6×10^{-2}	5.1	2.8×10^1
I-132	6.2×10^{-4}	1.2×10^{-3}	6.0×10^{-2}	1.1×10^{-1}	1.1×10^1	2.1×10^1
I-133	4.3×10^{-4}	2.7×10^{-3}	4.1×10^{-2}	1.8×10^{-1}	3.4	1.5×10^1
I-134	9.9×10^{-4}	1.2×10^{-3}	9.6×10^{-2}	1.2×10^1	4.2×10^{-1}	5.1×10^{-1}
I-135	6.5×10^{-4}	2.4×10^{-3}	6.2×10^{-2}	1.9×10^{-1}	1.7	5.0
Xe-133	2.3×10^{-3}	2.0×10^{-1}	2.2×10^{-1}	2.5	2.2×10^{-1}	2.5
Xe-135	4.6×10^{-3}	3.3×10^{-2}	4.4×10^{-1}	2.1	4.4×10^{-1}	2.1
Cs-134	3.3×10^{-11}	3.2×10^{-10}	3.1×10^{-9}	1.8×10^{-8}	2.2×10^{-4}	4.4
Cs-136	9.4×10^{-12}	8.9×10^{-11}	9.1×10^{-10}	5.1×10^{-9}	1.1×10^{-6}	3.0
Cs-137	1.1×10^{-11}	1.0×10^{-10}	1.0×10^{-9}	5.9×10^{-9}	1.1×10^{-3}	2.0

TABLE I.6 (Cont'd)

Radio-Nuclide	GCR1 ^a (Ci)		GCR2 (Ci)		GCR6 (Ci)	
	0-2 hours	Entire	0-2 hours	Entire	0-2 hours	Entire
Ba-140	8.5×10^{-11}	8.0×10^{-10}	8.2×10^{-9}	4.6×10^{-8}	9.8×10^{-6}	1.8×10^1
La-140	9.7×10^{-10}	7.3×10^{-9}	9.4×10^{-8}	4.6×10^{-7}	2.4×10^{-5}	6.7
Ce-141	4.3×10^{-11}	4.2×10^{-10}	4.1×10^{-9}	2.4×10^{-8}	1.3×10^{-5}	2.7×10^1
Ce-143	9.4×10^{-10}	6.7×10^{-9}	9.0×10^{-8}	4.3×10^{-7}	1.2×10^{-5}	3.7
Ce-144	2.0×10^{-12}	1.9×10^{-11}	1.9×10^{-10}	1.1×10^{-9}	5.0×10^{-6}	9.6
Pr-143	9.8×10^{-11}	9.2×10^{-10}	9.4×10^{-9}	5.3×10^{-8}	2.4×10^{-5}	2.3×10^1
Nd-147	4.9×10^{-11}	4.6×10^{-10}	4.7×10^{-9}	2.6×10^{-8}	4.8×10^{-6}	8.8

^aSee Table I.3 for accident descriptions.

^bHTO equivalent.

Sources: Bowman 1990c-d.

TABLE I.7 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Off-site Individual from Postulated NPR Nonsevere Accidents at Hanford

Accident ^a	Committed Effective Dose Equivalent (rem)	Maximum Organ ^b	Maximum Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR1	1.8×10^{-1}	S. int.	2.1×10^{-1}	2.0×10^{-1}	7×10^{-5}
HWR2	1.4×10^{-1}	LL. int.	1.6×10^{-1}	1.5×10^{-1}	6×10^{-5}
HWR4	3.7×10^{-1}	Thyroid	7.6	7.6	2×10^{-4}
LWR2A	3.4×10^{-2}	Thyroid	9.8×10^{-1}	9.8×10^{-1}	1×10^{-5}
LWR4A	6.4×10^{-2}	Thyroid	1.4	1.4	3×10^{-5}
LWR10A	3.1	Thyroid	2.1×10^1	2.1×10^1	1×10^{-3}
GCR1	1.5×10^{-5}	Thyroid	5.9×10^{-5}	5.9×10^{-5}	6.0×10^{-9}
GCR2	1.0×10^{-4}	Thyroid	2.6×10^{-3}	2.6×10^{-3}	4.0×10^{-8}
GCR6	6.7×10^{-1}	Thyroid	2.5	2.5	3×10^{-4}

^aLocation of maximally exposed off-site individual: 15,600 m east-southeast for HWR and MHTGR; 5,000 m east for LWR. See Tables I.1, I.2, and I.3 for accident descriptions.

^bMaximum organ is the organ receiving the highest dose. S. int. = small intestine; LL. int. = lower large intestine.

TABLE I.8 Estimated Doses to the Maximally Exposed Off-site Individual by Pathway from Postulated NPR Nonsevere Accidents at Hanford

Accident ^a	CEDE (rem)		EDE (rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR1	2.1×10^{-3}	1.8×10^{-1}	0	0
HWR2	1.6×10^{-3}	1.4×10^{-1}	2.7×10^{-9}	5.3×10^{-7}
HWR4	2.6×10^{-3}	3.6×10^{-1}	1.3×10^{-4}	4.2×10^{-3}
LWR2A	3.5×10^{-4}	3.0×10^{-2}	3.6×10^{-3}	3.2×10^{-4}
LWR4A	9.4×10^{-4}	6.2×10^{-2}	3.4×10^{-4}	1.0×10^{-3}
LWR10A	3.6×10^{-2}	3.1	1.3×10^{-4}	5.2×10^{-3}
GCR1	2.0×10^{-7}	1.5×10^{-5}	2.6×10^{-8}	4.0×10^{-8}
GCR2	2.8×10^{-6}	9.5×10^{-5}	2.0×10^{-6}	2.6×10^{-6}
GCR6	1.4×10^{-2}	6.4×10^{-1}	7.9×10^{-5}	6.4×10^{-3}

^aLocation of maximally exposed individual: 15,600 m east-southeast for HWR and MHTGR; 5,000 m east for LWR. See Tables I.1, I.2, and I.3 for accident descriptions.

TABLE I.9 Estimated Population Doses and Health Effects from Postulated NPR Nonsevere Accidents at Hanford

Accident ^a	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR1	3.8×10^3	4.1×10^3	4.1×10^3	2	1
HWR2	2.9×10^3	3.1×10^3	3.2×10^3	1	8×10^{-1}
HWR4	7.8×10^3	3.9×10^3	1.5×10^5	3	1
LWR2A	2.7×10^2	1.2	7.1×10^3	1×10^{-1}	1×10^{-2}
LWR4A	4.9×10^2	1.9×10^2	1.0×10^4	2×10^1	5×10^{-2}
LWR10A	2.5×10^4	2.3×10^4	1.6×10^5	1×10^1	6
GCR1	3.0×10^{-1}	3.2×10^{-1}	6.0×10^{-1}	1×10^{-4}	8×10^{-5}
GCR2	1.0	5.3×10^{-1}	1.7×10^1	4×10^{-4}	2×10^{-4}
GCR6	1.4×10^4	1.4×10^4	2.5×10^4	6	4

^aWorst-sector population: 104,000 persons south-southeast for HWR and MHTGR; 96,000 persons south-southeast for LWR. See Tables I.1, I.2, and I.3 for accident descriptions.

TABLE I.10 Estimated Population Doses by Pathway from Postulated NPR Nonsevere Accidents at Hanford

Accident	CEDE (rem)		EDE (rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR1	7.7×10^1	3.7×10^3	0	0
HWR2	5.8×10^1	2.8×10^3	9.3×10^{-5}	1.4×10^{-2}
HWR4	9.4×10^1	7.6×10^3	4.6	1.1×10^2
LWR2A	4.9	2.1×10^2	4.6×10^1	3.0
LWR4A	1.3×10^1	4.6×10^2	3.2	8.5
LWR10A	4.9×10^2	2.5×10^4	1.8	5.2×10^1
GCR1	7.4×10^{-3}	2.9×10^{-1}	1.0×10^{-3}	1.0×10^{-3}
GCR2	1.0×10^{-1}	7.7×10^{-2}	7.7×10^{-2}	6.8×10^{-2}
GCR6	5.1×10^2	1.3×10^4	2.6	1.6×10^2

^aWorst-sector population: 104,000 persons south-southeast for HWR and LWR; 96,000 persons south-southeast for MHTGR. See Tables I.1, I.2, and I.3 for accident descriptions.

TABLE I.11 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Colocated Worker from Postulated NPR Nonsevere Accidents at Hanford

Accident ^a	Committed Effective Dose Equivalent (rem)	Maximum Organ ^b	Maximum Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR1	1.5×10^{-1}	S. int.	1.8×10^{-1}	1.7×10^{-1}	6×10^{-5}
HWR2	3.3×10^{-4}	LL. int.	3.9×10^{-4}	3.7×10^{-4}	1×10^{-7}
HWR4	3.1×10^{-2}	Thyroid	6.1×10^{-1}	6.1×10^{-1}	1×10^{-5}
LWR2A	3.5×10^{-2}	Thyroid	3.0×10^{-1}	3.0×10^{-1}	1×10^{-5}
LWR4A	3.3×10^{-2}	Thyroid	6.7×10^{-1}	6.7×10^{-1}	1×10^{-5}
LWR10A	1.1×10^{-2}	Thyroid	1.5×10^{-1}	1.5×10^{-1}	4×10^{-6}
GCR1	6.8×10^{-7}	Thyroid	1.0×10^{-5}	1.0×10^{-5}	3×10^{-10}
GCR2	6.5×10^{-5}	Thyroid	9.6×10^{-4}	9.6×10^{-4}	3×10^{-8}
GCR6	3.7×10^{-2}	Thyroid	4.9×10^{-1}	4.9×10^{-1}	2×10^{-5}

^aLocation of maximally exposed colocated worker: 700 m west-southwest for HWR and MHTGR; 500 m south-southwest for LWR. See Tables I.1, I.2, and I.3 for accident descriptions.

^bMaximum organ is the organ receiving the highest dose. S. int. = small intestine; LL. int. = lower large intestine.

TABLE I.12 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Off-site Individual from Postulated Nonsevere NPR Accidents at INEL

Accident ^a	Committed Effective Dose Equivalent (rem)	Maximum Organ ^b	Maximum Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR1	2.7×10^{-1}	S. int.	3.2×10^{-1}	3.0×10^{-1}	1×10^{-4}
HWR2	2.1×10^{-1}	LL. int.	2.4×10^{-1}	2.3×10^{-1}	8×10^{-5}
HWR4	5.5×10^{-1}	Thyroid	1.2×10^1	1.2×10^1	2×10^{-4}
LWR2A	2.1×10^{-2}	Thyroid	6.2×10^{-1}	6.2×10^{-1}	8×10^{-6}
LWR4A	4.0×10^{-2}	Thyroid	8.8×10^{-1}	8.8×10^{-1}	2×10^{-5}
LWR10A	1.9	Thyroid	1.3×10^1	1.3×10^1	8×10^{-4}
GCR1	2.3×10^{-5}	Thyroid	8.9×10^{-5}	8.9×10^{-5}	9×10^{-9}
GCR2	1.5×10^{-4}	Thyroid	3.9×10^{-3}	3.9×10^{-3}	6×10^{-8}
GCR6	1.0	Thyroid	3.8	3.8	4×10^{-4}

^aLocation of maximally exposed off-site individual: 14,800 m south-southeast. See Tables I.1, I.2, and I.3 for accident descriptions.

^bMaximum organ is the organ receiving the highest dose. S. int. = small intestine; L.L. int. = lower large intestine.

TABLE I.13 Estimated Doses to the Maximally Exposed Off-site Individual by Pathway from Postulated NPR Nonsevere Accidents at INEL

Accident ^a	CEDE (rem)		EDE (rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR1	3.6×10^{-3}	2.7×10^{-1}	0	0
HWR2	2.7×10^{-3}	2.0×10^{-1}	3.4×10^{-9}	9.3×10^{-7}
HWR4	4.3×10^{-3}	5.4×10^{-1}	1.7×10^{-4}	6.9×10^{-3}
LWR2A	2.5×10^{-4}	1.9×10^{-2}	1.8×10^{-3}	2.1×10^{-4}
LWR4A	6.3×10^{-4}	3.8×10^{-2}	1.6×10^{-4}	6.4×10^{-4}
LWR10A	2.4×10^{-2}	1.9	6.7×10^{-5}	3.4×10^{-3}
GCR1	3.4×10^{-7}	2.2×10^{-5}	4.0×10^{-8}	6.8×10^{-8}
GCR2	4.9×10^{-6}	1.4×10^{-4}	3.1×10^{-6}	4.5×10^{-6}
GCR6	2.3×10^{-2}	9.7×10^{-1}	9.3×10^{-5}	1.1×10^{-2}

^aLocation of maximally exposed off-site individual: 14,800 m south-southeast. See Tables I.1, I.2, and I.3 for accident descriptions.

TABLE I.14 Estimated Population Doses and Health Effects from Postulated NPR Nonsevere Accidents at INEL

Accident ^a	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR1	2.5×10^3	2.7×10^3	2.7×10^3	1	7×10^{-1}
HWR2	1.9×10^3	2.1×10^3	2.1×10^3	8×10^{-1}	6×10^{-1}
HWR4	4.1×10^3	1.9×10^3	8.1×10^4	2	5×10^{-1}
LWR2A	1.5×10^2	6.7×10^{-1}	4.0×10^3	6×10^{-2}	8×10^{-3}
LWR4A	2.9×10^2	1.3×10^2	5.7×10^3	1×10^{-1}	4×10^{-2}
LWR10A	1.6×10^4	1.5×10^4	9.2×10^4	6	4
GCR1	2.0×10^{-1}	2.1×10^{-1}	3.6×10^{-1}	8×10^{-5}	6×10^{-5}
GCR2	5.7×10^{-1}	3.1×10^{-1}	9.2	2×10^{-4}	1×10^{-4}
GCR6	8.6×10^3	8.6×10^3	1.4×10^4	3	2

^aWorst-sector population: 113,000 persons east. See Tables I.1, I.2, and I.3 for accident descriptions.

TABLE I.15 Estimated Population Doses by Pathway from Postulated NPR Nonsevere Accidents at INEL

Accident ^a	CEDE (rem)		EDE (rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR1	4.2×10^1	2.4×10^3	0	0
HWR2	3.2×10^1	1.9×10^3	5.7×10^{-5}	7.6×10^{-3}
HWR4	5.1×10^1	4.0×10^3	2.7	5.8×10^1
LWR2A	2.7	1.2×10^2	2.7×10^1	1.7
LWR4A	7.1	2.8×10^2	1.9	4.7
LWR10A	2.8×10^2	1.6×10^4	1.0	2.8×10^1
GCR1	4.0×10^{-3}	1.9×10^{-1}	4.9×10^{-4}	5.3×10^{-4}
GCR2	5.4×10^{-2}	4.5×10^{-1}	3.7×10^{-2}	3.5×10^{-2}
GCR6	2.8×10^2	8.2×10^3	1.9	9.1×10^1

^aWorst-sector population: 113,000 persons east; see Tables I.1, I.2, and I.3 for accident descriptions.

TABLE I.16 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Colocated Worker from Postulated NPR Nonsevere Accidents at INEL

Accident ^a	Committed Effective Dose Equivalent (rem)	Maximum Organ ^b	Maximum Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR1	1.9×10^{-2}	S. int.	2.2×10^{-2}	2.0×10^{-2}	8×10^{-6}
HWR2	4.0×10^{-5}	LL. int.	4.7×10^{-5}	4.5×10^{-5}	2×10^{-8}
HWR4	3.9×10^{-3}	Thyroid	7.4×10^{-2}	7.4×10^{-2}	2×10^{-6}
LWR2A	7.1×10^{-3}	Thyroid	3.4×10^{-2}	3.4×10^{-2}	3×10^{-6}
LWR4A	3.9×10^{-3}	Thyroid	7.3×10^{-2}	7.3×10^{-2}	2×10^{-6}
LWR10A	1.2×10^{-3}	Thyroid	1.6×10^{-2}	1.6×10^{-2}	5×10^{-7}
GCR1	9.8×10^{-8}	Thyroid	1.2×10^{-6}	1.2×10^{-6}	4×10^{-11}
GCR2	9.2×10^{-6}	Thyroid	1.2×10^{-4}	1.2×10^{-4}	4×10^{-9}
GCR6	4.4×10^{-3}	Thyroid	5.9×10^{-2}	5.9×10^{-2}	2×10^{-6}

^aLocation of maximally exposed colocated worker: 2,900 m south. See Tables I.1, I.2, and I.3 for accident descriptions.

^bMaximum organ is the organ receiving the highest dose. S. int. = small intestine; LL. int. = lower large intestine.

TABLE I.17 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Off-site Individual from Postulated NPR Nonsevere Accidents at SRS

Accident ^a	Committed Effective Dose Equivalent (rem)	Maximum Organ ^b	Maximum Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR1	4.9×10^{-2}	S. int.	5.7×10^{-2}	5.3×10^{-2}	2×10^{-5}
HWR2	3.8×10^{-2}	LL. int.	4.4×10^{-2}	4.3×10^{-2}	2×10^{-5}
HWR4	1.8×10^{-1}	Thyroid	3.8	3.8	7×10^{-5}
LWR2A	6.5×10^{-3}	Thyroid	1.9×10^{-1}	1.9×10^{-1}	3×10^{-6}
LWR4A	1.1×10^{-2}	Thyroid	2.6×10^{-1}	2.6×10^{-1}	4×10^{-6}
LWR10A	3.9×10^{-1}	Thyroid	3.8	3.8	2×10^{-4}
GCR1	4.6×10^{-6}	Thyroid	2.7×10^{-5}	2.7×10^{-5}	2×10^{-9}
GCR2	5.0×10^{-5}	Thyroid	1.4×10^{-3}	1.4×10^{-3}	2×10^{-8}
GCR6	2.4×10^{-1}	Thyroid	1.2	1.2	1×10^{-4}

^aLocation of maximally exposed off-site individual: 14,500 m north-northwest. See Tables I.1, I.2, and I.3 for accident descriptions.

^bMaximum organ is the organ receiving the highest dose. S. int. = small intestine; LL. int. = lower large intestine.

TABLE I.18 Estimated Doses to the Maximally Exposed Off-site Individual by Pathway from Postulated NPR Nonsevere Accidents at SRS

Accident ^a	CEDE (rem)		EDE (rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR1	9.3×10^{-4}	4.8×10^{-2}	0	0
HWR2	7.1×10^{-4}	3.7×10^{-2}	1.3×10^{-9}	2.4×10^{-7}
HWR4	1.1×10^{-3}	1.7×10^{-1}	5.9×10^{-5}	1.8×10^{-3}
LWR2A	6.4×10^{-5}	5.8×10^{-3}	6.2×10^{-4}	5.4×10^{-5}
LWR4A	1.6×10^{-4}	1.0×10^{-2}	4.8×10^{-5}	1.6×10^{-4}
LWR10A	6.2×10^{-3}	3.8×10^{-1}	2.3×10^{-5}	9.0×10^{-4}
GCR1	9.0×10^{-8}	4.4×10^{-6}	1.2×10^{-8}	1.7×10^{-8}
GCR2	1.3×10^{-6}	4.6×10^{-5}	8.8×10^{-7}	1.2×10^{-6}
GCR6	6.1×10^{-3}	2.3×10^{-1}	3.5×10^{-5}	2.8×10^{-3}

^aLocation of maximally exposed off-site individual: 14,500 m north-northwest. See Tables I.1, I.2, and I.3 for accident descriptions.

TABLE I.19 Estimated Population Doses and Health Effects from Postulated NPR Nonsevere Accidents at SRS

Accident ^a	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR1	8.0×10^2	8.8×10^2	8.8×10^2	3×10^{-1}	2×10^{-1}
HWR2	6.1×10^2	6.7×10^2	6.8×10^2	2×10^{-1}	2×10^{-1}
HWR4	2.3×10^3	1.5×10^3	3.4×10^4	9×10^{-1}	4×10^{-1}
LWR2A	7.4×10^1	2.9×10^{-1}	1.7×10^3	3×10^{-2}	6×10^{-3}
LWR4A	1.1×10^2	4.2×10^1	2.3×10^3	4×10^{-2}	1×10^{-2}
LWR10A	5.5×10^3	5.0×10^3	3.6×10^4	2	1
GCR1	6.6×10^{-2}	6.9×10^{-2}	1.7×10^{-1}	3×10^{-5}	2×10^{-5}
GCR2	3.3×10^{-1}	1.8×10^{-1}	6.1	1×10^{-4}	6×10^{-5}
GCR6	3.8×10^3	3.7×10^3	7.6×10^3	2	1

^aWorst-sector population: 254,000 persons west-northwest. See Tables I.1, I.2, and I.3 for accident descriptions.

TABLE I.20 Estimated Population Doses by Pathway from Postulated NPR Nonsevere Accidents at SRS

Accident ^a	CEDE (rem)		EDE (rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR1	3.4×10^1	7.6×10^2	0	0
HWR2	2.6×10^1	5.9×10^2	5.2×10^{-5}	6.2×10^{-3}
HWR4	4.0×10^1	2.2×10^3	2.2	4.7×10^1
LWR2A	2.1	4.9×10^1	2.2×10^1	1.3
LWR4A	5.6	1.0×10^2	1.4	3.6
LWR10A	2.3×10^2	5.3×10^3	8.7×10^{-1}	2.2×10^1
GCR1	3.3×10^{-3}	6.2×10^{-2}	3.7×10^{-4}	4.0×10^{-4}
GCR2	4.3×10^{-2}	2.4×10^{-1}	2.6×10^{-2}	2.6×10^{-2}
GCR6	2.2×10^2	3.5×10^3	1.8	7.3×10^1

^aWorst-sector population: 254,000 persons west-northwest. See Tables I.1, I.2, and I.3 for accident descriptions.

TABLE I.21 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Colocated Worker from Postulated NPR Nonsevere Accidents at SRS

Accident ^a	Committed Effective Dose Equivalent (rem)	Maximum Organ ^b	Maximum Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR1	1.0×10^{-2}	S. int.	1.2×10^{-2}	1.1×10^{-2}	4×10^{-6}
HWR2	2.1×10^{-5}	LL. int.	2.5×10^{-5}	2.4×10^{-5}	8×10^{-9}
HWR4	2.2×10^{-3}	Thyroid	3.9×10^{-2}	3.9×10^{-2}	9×10^{-7}
LWR2A	5.2×10^{-3}	Thyroid	1.8×10^{-2}	1.8×10^{-2}	2×10^{-6}
LWR4A	2.2×10^{-3}	Thyroid	3.8×10^{-2}	3.8×10^{-2}	9×10^{-7}
LWR10A	6.3×10^{-4}	Thyroid	8.7×10^{-3}	8.7×10^{-3}	3×10^{-7}
GCR1	6.2×10^{-8}	Thyroid	6.5×10^{-7}	6.5×10^{-7}	3×10^{-11}
GCR2	5.8×10^{-6}	Thyroid	6.2×10^{-5}	6.2×10^{-5}	2×10^{-9}
GCR6	2.3×10^{-3}	Thyroid	3.1×10^{-2}	3.1×10^{-2}	9×10^{-7}

^aLocation of maximally exposed colocated worker: 2,100 m west-southwest. See Tables I.1, I.2, and I.3 for accident descriptions.

^bMaximum organ is the organ receiving the highest dose. S. int. = small intestine; LL. int. = lower large intestine.

TABLE I.22 Estimated Upper Bound Occupational Doses to In-Plant Workers from Postulated Nonsevere Accidents in the Proposed NPRs^a

Reactor	Dose from Immediate Exposure		Collective CEDE from Long-Term Exposure (person-rem)
	Individual CEDE (rem)	Collective CEDE (person-rem)	
HWR	3.0×10^{-2}	2.0×10^{-1}	6.0×10^2
LWR	1.0×10^1	2.0×10^1	2.0×10^3
MHTGR	0.5	2.0×10^1	3.0×10^2

^aSee Section I.3.2.4 and Tables I.1, I.2, and I.3 for descriptions of the accidents resulting in these estimated doses.

TABLE I.23 Identification of Bounding Postulated Nonsevere Accidents at Hanford Support Facilities

Facility/ Accident ID ^{a,b}	Description
Fuel Fabrication	
HH1C1 ^{c,d}	Uranium criticality
HL1C1 ^{c,d}	Uranium criticality
HM1C1 ^{c,d}	Uranium criticality
Fuel Processing	
HH2C1 ^c	Solution criticality - fuel and plutonium-target processing
HH2C2 ^d	
HL2C2 ^c	Uranium powder criticality - uranium processing
HL2N2 ^d	
HM2C2 ^c	Uranium powder criticality - uranium processing
HM2N1 ^d	
	Damage 30-day cooled fuel - fuel and plutonium-target processing
	Uranium criticality - uranium processing
	Process 30-day cooled fuel - fuel and plutonium-target processing
Plutonium-Target Processing	
HH4C1 ^c	Plutonium oxide criticality
HH4N1 ^d	
	Process 30-day cooled target
HL4C1 ^{c,d}	Plutonium oxide criticality
HM4C1 ^c	Plutonium criticality
HM4N1 ^d	
	Process 30-day cooled target
Tritium-Target Processing	
HH6D1 ^{c,d}	Tritium deflagration
HL6D1 ^{c,d}	
HM6D1 ^{c,d}	

^aSee Section I.4.1.1 for explanation of accident IDs.

^bAll accidents listed belong to the "extremely unlikely" frequency class.

^cBounding accident for the maximally exposed colocated worker.

^dBounding accident for the maximally exposed off-site individual and worst-sector population.

TABLE I.24 Identification of Bounding Postulated Nonsevere Accidents at INEL Support Facilities

Facility/ Accident ID ^{a,b}	Description
Fuel Fabrication	
IH1C1 ^{c,d}	Criticality
IL1C1 ^{c,d}	Criticality
IM1C1 ^{c,d}	Criticality
Fuel Processing	
IH2C2 ^c	Criticality - fuel processing
IH2N1 ^d	30-day cooled fuel dissolution - headend fuel processing
IL2C1 ^{c,d}	IBX column criticality - fuel and plutonium-target processing
IM2C2 ^c	Criticality - fuel processing
IM2F1 ^d	30-day cooled fuel burning - headend fuel processing
Plutonium-Target Processing	
IH4C2 ^e	Criticality - plutonium oxide-to-metal conversion
IH4N1 ^f	30-day cooled target dissolution - plutonium-target processing
IL4C1 ^{c,d}	Criticality - plutonium oxide-to-metal conversion
IM4C2 ^c	Criticality - plutonium oxide-to-metal conversion
IM4F1 ^d	30-day cooled fuel burning - plutonium-target processing
Tritium-Target Processing	
IH6D1 ^{c,d}	Tritium deflagration
IL6D1 ^{c,d}	Tritium deflagration
IM6D1 ^{c,d}	Tritium deflagration

^aSee Section I.4.1.1 for explanation of accident IDs.^bAll accidents listed belong to the "extremely unlikely" frequency class.^cBounding accident for the maximally exposed colocated worker.^dBounding accident for the maximally exposed off-site individual and worst-sector population.^eBounding accident for the worst-sector population and maximally exposed colocated worker.^fBounding accident for the maximally exposed off-site individual.

**TABLE I.25 Identification of Bounding Postulated
Nonsevere Accidents at SRS Support Facilities**

Facility/ Accident ID ^{a,b}	Description
Fuel Fabrication	
SH1C1 ^{c,d}	Criticality
SL1C1 ^{c,d}	Criticality
SM1C1 ^{c,d}	Criticality
Fuel Processing	
SH2N1 ^{c,d}	Process 30-day cooled fuel
SL2F1 ^c	Zircaloy hull fire
SL2N2 ^d	Process 30-day cooled fuel
SM2F1 ^{c,d}	30-day cooled fuel fire
Plutonium-Target Processing	
SH4N1 ^{c,d}	Process 30-day cooled fuel
SM4F1 ^{c,d}	30-day cooled fuel burning
Tritium-Target Processing	
SH6D1 ^{c,d}	Tritium deflagration
SL6D1 ^{c,d}	Tritium deflagration
SM6D1 ^{c,d}	Tritium deflagration

^aSee Section I.4.1.1 for explanation of accident IDs.

^bAll accidents listed belong to the "extremely unlikely" frequency class, with the exceptions of accidents SH2N1, SL2F1, and SH4N1 which are in the "anticipated" frequency class.

^cBounding accident for the maximally exposed colocated worker.

^dBounding accident for the maximally exposed off-site individual and worst-sector population.

TABLE I.26 Estimated Releases from the Highest Consequence HWR
Nonsevere Accidents at Hanford Support Facilities^a

Radio-nuclide	Releases (Ci)				
	Fuel Fabrication (HH1C1)	Fuel Processing (HH2C2) ^b	Fuel Processing (HH2C1) ^c	Plutonium-Target Processing (HH4N1) ^b	Plutonium-Target Processing (HH4C1) ^c
Kr-83m	1.6×10^2	1.1×10^2	1.6×10^2	-	1.1×10^2
Kr-85m	1.5×10^2	7.1×10^1	1.5×10^2	-	7.1×10^1
Kr-85	1.6×10^{-3}	8.1×10^{-4}	1.6×10^{-3}	1.1×10^3	8.1×10^{-4}
Kr-87	9.9×10^2	4.3×10^2	9.9×10^2	-	4.3×10^2
Kr-88	6.5×10^2	2.3×10^2	6.5×10^2	-	2.3×10^2
Kr-89	4.2×10^4	-	4.2×10^4	-	-
Xe-131m	8.2×10^{-2}	1.0×10^{-1}	8.2×10^{-2}	-	1.0×10^{-1}
Xe-133m	1.8	2.2	1.8	-	2.2
Xe-133	2.7×10^1	2.7×10^1	2.7×10^1	1.2×10^5	2.7×10^1
Xe-135m	2.2×10^3	-	2.2×10^3	-	-
Xe-135	3.6×10^2	4.1×10^2	3.6×10^2	-	4.1×10^2
Xe-137	4.9×10^4	-	4.9×10^4	-	-
Xe-138	1.3×10^4	1.1×10^4	1.3×10^4	-	1.1×10^4
I-131	8.7	2.2×10^1	8.7	1.7×10^2	2.2×10^2
I-132	1.1×10^3	2.0×10^3	1.1×10^3	5.4	2.0×10^3
I-133	1.6×10^2	3.2×10^2	1.6×10^2	-	3.2×10^2
I-134	4.5×10^3	8.6×10^3	4.5×10^3	-	8.6×10^3
I-135	4.7×10^2	9.0×10^2	4.7×10^2	-	9.0×10^2
Pu-238	-	-	-	-	1.4×10^{-9}
Pu-239	-	-	-	-	5.9×10^{-8}
Pu-240	-	-	-	-	1.1×10^{-8}
Pu-241	-	-	-	-	7.3×10^{-7}

^aThe source term for the highest consequence accident at the tritium-target processing facility (HH6D1) is estimated at 1×10^6 Ci of HTO.

^bHighest consequence accident for the maximally exposed off-site individual and worst-sector population.

^cHighest consequence accident for the maximally exposed colocated worker.

Sources: HH1C1 - NRC 1979; HH2C1, HH2C2, HH4C1 - WHC 1989a;
HH4N1 - Heacock 1990c; HH6D1 - Cornwell 1990.

TABLE I.27 Estimated Releases from the Highest Consequence LWR Nonsevere Accidents at Hanford Support Facilities

Radio-nuclide	Releases (Ci)				
	Fuel Fabrication (HL1C1)	Fuel Processing (HL2N2) ^a	Fuel Processing (HL2C2) ^b	Plutonium-Target Processing (HL4C1)	Tritium-Target Processing (HL6D1)
H-3	-	-	-	-	1.0 × 10 ⁶
Kr-83m	1.6 × 10 ²	-	1.1 × 10 ²	1.1 × 10 ²	-
Kr-85m	1.5 × 10 ²	-	7.1 × 10 ¹	7.1 × 10 ¹	-
Kr-85	1.6 × 10 ⁻³	3.8 × 10 ¹	8.1 × 10 ⁻⁴	8.1 × 10 ⁻⁴	-
Kr-87	9.9 × 10 ²	-	4.3 × 10 ²	4.3 × 10 ²	-
Kr-88	6.5 × 10 ²	-	2.3 × 10 ²	2.3 × 10 ²	-
Kr-89	4.2 × 10 ⁴	-	-	-	-
Xe-131m	8.2 × 10 ⁻²	3.9 × 10 ¹	1.0 × 10 ⁻¹	1.0 × 10 ⁻¹	-
Xe-133m	1.8	-	2.2	2.2	-
Xe-133	2.7 × 10 ¹	4.6 × 10 ²	2.7 × 10 ¹	2.7 × 10 ¹	-
Xe-135m	2.2 × 10 ³	-	-	-	-
Xe-135	3.6 × 10 ²	-	4.1 × 10 ²	4.1 × 10 ²	-
Xe-137	4.9 × 10 ⁴	-	-	-	-
Xe-138	1.3 × 10 ⁴	-	1.1 × 10 ⁴	1.1 × 10 ⁴	-
I-131	8.7	2.3 × 10 ²	2.2 × 10 ¹	2.2 × 10 ¹	-
I-132	1.1 × 10 ³	7.7	2.0 × 10 ³	2.0 × 10 ³	-
I-133	1.6 × 10 ²	-	3.2 × 10 ²	3.2 × 10 ²	-
I-134	4.5 × 10 ³	-	8.6 × 10 ³	8.6 × 10 ³	-
I-135	4.7 × 10 ²	-	9.0 × 10 ²	9.0 × 10 ²	-
U-234	-	-	5.6 × 10 ⁻⁸	-	-
U-235	-	-	1.7 × 10 ⁻⁹	-	-
U-236	-	-	3.1 × 10 ⁻⁹	-	-
U-238	-	-	3.1 × 10 ⁻⁹	-	-
Pu-238	-	-	-	4.1 × 10 ⁻⁸	-
Pu-239	-	-	-	5.2 × 10 ⁻⁸	-
Pu-240	-	-	-	2.5 × 10 ⁻⁸	-
Pu-241	-	-	-	5.1 × 10 ⁻⁶	-
Pu-242	-	-	-	1.2 × 10 ⁻¹¹	-

^aHighest consequence accident for the maximally exposed off-site individual and worst-sector population.

^bHighest consequence accident for the maximally exposed colocated worker.

Sources: HL1C1 - NRC 1979; HL2C2, HL2N2, HL4C1 - WHC 1990b; HL6D1 - Cornwell 1990.

TABLE I.28 Estimated Releases from the Highest Consequence MHTGR
Nonsevere Accidents at Hanford Support Facilities^a

Radio-nuclide	Releases (Ci)				
	Fuel Fabrication (HM1C1)	Fuel Processing (HM2N1) ^b	Fuel Processing (HM2C2) ^c	Plutonium-Target Processing (HM4N1) ^b	Plutonium-Target Processing (HM4C1) ^c
H-3	-	1.0	1.5×10^2	1.5×10^{-2}	1.4
C-14	-	1.2×10^{-3}	1.9×10^{-1}	1.1×10^{-2}	1.1
Kr-83m	1.6×10^2	-	3.7×10^1	-	1.1×10^2
Kr-85m	1.5×10^2	-	1.7×10^2	-	7.1×10^1
Kr-85	1.6×10^{-3}	7.8×10^2	1.1×10^3	2.7×10^1	2.5×10^1
Kr-87	9.9×10^2	-	1.0×10^3	-	4.3×10^2
Kr-88	6.5×10^2	-	6.6×10^2	-	2.3×10^2
Kr-89	4.2×10^4	-	4.1×10^4	-	1.3×10^4
Rb-86	-	1.0×10^{-8}	2.1×10^{-11}	7.4×10^{-11}	-
Sr-89	-	1.2×10^{-6}	6.6×10^{-6}	4.5×10^{-7}	1.6×10^{-6}
Sr-90	-	1.8×10^{-7}	9.5×10^{-5}	6.1×10^{-9}	2.1×10^{-6}
Y-90	-	1.8×10^{-7}	9.5×10^{-5}	6.1×10^{-9}	2.1×10^{-6}
Y-91	-	1.7×10^{-6}	1.7×10^{-5}	5.3×10^{-7}	3.5×10^{-6}
Zr-95	-	2.0×10^{-6}	2.9×10^{-5}	5.7×10^{-7}	5.2×10^{-6}
Nb-95	-	2.6×10^{-6}	6.2×10^{-5}	5.2×10^{-7}	1.1×10^{-5}
Nb-95m	-	1.5×10^{-8}	2.1×10^{-7}	4.2×10^{-9}	3.9×10^{-8}
Mo-99	-	1.4×10^{-9}	-	7.7×10^{-10}	-
Tc-99	-	1.6×10^{-11}	8.3×10^{-9}	-	3.0×10^{-10}
Tc-99m	-	1.4×10^{-9}	-	7.4×10^{-10}	-
Ru-103	-	1.1×10^{-4}	1.7×10^{-6}	5.7×10^{-5}	5.9×10^{-7}
Ru-106	-	3.0×10^{-5}	9.3×10^{-5}	1.0×10^{-5}	2.1×10^{-5}
Rh-103m	-	1.1×10^{-6}	1.5×10^{-6}	5.7×10^{-7}	5.3×10^{-7}
Sn-123	-	2.6×10^{-9}	2.3×10^{-7}	1.3×10^{-9}	7.7×10^{-8}
Sb-125	-	1.1×10^{-8}	4.6×10^{-6}	1.8×10^{-9}	5.0×10^{-7}
Sb-127	-	5.5×10^{-10}	-	5.3×10^{-10}	-
Te-125m	-	2.4×10^{-9}	1.1×10^{-6}	2.1×10^{-10}	1.2×10^{-7}
Te-127	-	1.2×10^{-8}	7.0×10^{-7}	5.1×10^{-9}	1.9×10^{-7}
Te-127m	-	1.1×10^{-8}	7.2×10^{-7}	4.7×10^{-9}	1.9×10^{-7}
Te-129	-	2.1×10^{-8}	1.1×10^{-8}	1.3×10^{-8}	4.4×10^{-9}
Te-129m	-	3.3×10^{-8}	1.7×10^{-8}	2.0×10^{-8}	6.7×10^{-9}
Te-132	-	3.4×10^{-9}	-	2.0×10^{-9}	-
Xe-131m	8.2×10^{-2}	2.2×10^2	3.9×10^{-3}	1.0×10^2	1.0×10^{-1}
Xe-133m	1.8	3.8×10^{-1}	5.5×10^{-1}	1.3×10^3	2.7×10^1
Xe-133	2.7×10^1	2.4×10^3	1.3×10^1	2.1×10^{-1}	2.2
Xe-135m	2.2×10^3	-	1.1×10^2	-	3.3×10^3
Xe-135	3.6×10^2	-	1.7×10^2	-	4.1×10^2
Xe-137	4.9×10^4	-	3.9×10^4	-	4.9×10^4
Xe-138	1.3×10^4	-	1.1×10^4	-	1.1×10^4
I-129	-	1.5×10^{-5}	5.6×10^{-4}	9.4×10^{-7}	2.5×10^{-5}
I-131	8.7	3.9×10^1	4.5×10^{-1}	2.3×10^1	2.8

TABLE I.28 (Cont'd)

Radio-nuclide	Releases (Ci)				
	Fuel Fabrication (HM1C1)	Fuel Processing (HM2N1) ^b	Fuel Processing (HM2C2) ^c	Plutonium-Target Processing (HM4N1) ^b	Plutonium-Target Processing (HM4C1) ^c
I-132	1.1×10^3	1.2	1.7	7.2×10^{-1}	3.0×10^2
I-133	1.6×10^2	3.9×10^{-8}	8.8	2.2×10^{-8}	4.0×10^1
I-134	4.5×10^3	-	1.2×10^2	-	1.1×10^3
I-135	4.7×10^2	-	3.0×10^1	-	1.1×10^2
Cs-134	-	8.7×10^{-7}	3.4×10^{-4}	3.0×10^{-9}	7.5×10^{-7}
Cs-136	-	1.3×10^{-7}	1.4×10^{-12}	5.3×10^{-9}	-
Cs-137	-	2.0×10^{-7}	1.0×10^{-4}	6.5×10^{-9}	2.2×10^{-6}
Ba-140	-	5.5×10^{-7}	3.7×10^{-12}	2.8×10^{-7}	1.3×10^{-12}
La-140	-	6.3×10^{-7}	4.3×10^{-12}	3.2×10^{-7}	1.4×10^{-12}
Ce-141	-	1.4×10^{-6}	5.9×10^{-7}	5.5×10^{-7}	1.5×10^{-7}
Ce-144	-	2.1×10^{-6}	5.0×10^{-4}	1.8×10^{-7}	2.8×10^{-5}
Pr-143	-	5.7×10^{-7}	1.1×10^{-11}	3.1×10^{-7}	3.9×10^{-12}
Pr-144	-	2.1×10^{-6}	5.0×10^{-4}	1.8×10^{-7}	2.8×10^{-5}
Pr-144m	-	2.5×10^{-8}	6.0×10^{-6}	2.2×10^{-9}	3.4×10^{-7}
Nd-147	-	1.6×10^{-7}	-	8.1×10^{-8}	-
Pm-148m	-	5.2×10^{-9}	7.9×10^{-10}	2.4×10^{-9}	6.0×10^{-8}
Sm-151	-	6.0×10^{-10}	3.2×10^{-7}	2.0×10^{-10}	6.9×10^{-8}
Eu-154	-	2.4×10^{-8}	1.2×10^{-5}	1.2×10^{-10}	3.9×10^{-8}
Eu-155	-	1.7×10^{-8}	7.9×10^{-6}	8.5×10^{-10}	2.6×10^{-7}
Np-237	-	1.8×10^{-12}	9.7×10^{-10}	-	3.2×10^{-12}
Np-239	-	1.0×10^{-9}	8.0×10^{-9}	-	4.9×10^{-12}
Pu-238	-	4.3×10^{-8}	2.3×10^{-5}	1.5×10^{-11}	1.1×10^{-8}
Pu-239	-	2.5×10^{-11}	1.3×10^{-8}	4.8×10^{-10}	1.7×10^{-7}
Pu-240	-	2.7×10^{-11}	1.5×10^{-8}	2.3×10^{-10}	8.1×10^{-8}
Pu-241	-	2.0×10^{-8}	1.0×10^{-5}	2.2×10^{-8}	7.4×10^{-6}
Pu-242	-	5.7×10^{-13}	3.0×10^{-10}	-	1.0×10^{-11}
Am-241	-	4.9×10^{-12}	1.8×10^{-8}	5.0×10^{-12}	1.3×10^{-8}
Am-243	-	1.5×10^{-11}	8.0×10^{-9}	-	4.9×10^{-12}
Cm-242	-	8.3×10^{-9}	1.1×10^{-6}	8.8×10^{-11}	7.3×10^{-9}
Cm-244	-	6.8×10^{-9}	3.5×10^{-6}	-	3.4×10^{-11}

^aThe source term for the highest consequence accident at the tritium-target processing facility (HM6D1) is estimated at 1×10^6 Ci of HTO.

^bHighest consequence accident for the maximally exposed off-site individual and worst-sector population.

^cHighest consequence accident for the maximally exposed colocated worker.

Sources: HM6D1 - Cornwell 1990; HM1C1 - NRC 1979; all other accidents - WHC 1989b, EG&G 1989c.

TABLE I.29 Estimated Releases from the Highest Consequence HWR Nonsevere Accidents at INEL Support Facilities^a

Radionuclide	Releases (Ci)				
	Fuel Fabrication (IH1C1)	Fuel Processing (IH2N1) ^b	Fuel Processing (IH2C2) ^c	Plutonium-Target Processing (IH4N1) ^d	Plutonium-Target Processing (IH4C2) ^e
H-3	-	2.1	2.8	1.8×10^{-1}	-
Kr-83m	1.6×10^2	-	3.7×10^1	-	1.1×10^2
Kr-85m	1.5×10^2	-	1.7×10^2	-	7.1×10^1
Kr-85	1.6×10^{-3}	5.2×10^2	7.6×10^{-1}	3.8×10^1	8.1×10^{-4}
Kr-87	9.9×10^2	-	1.0×10^3	-	4.3×10^2
Kr-88	6.5×10^2	-	6.6×10^2	-	2.3×10^2
Kr-89	4.2×10^4	-	4.1×10^4	-	1.3×10^4
Rb-86	-	1.4×10^{-10}	-	8.1×10^{-12}	-
Sr-89	-	4.0×10^{-7}	2.5×10^{-6}	6.9×10^{-8}	-
Sr-90	-	1.2×10^{-8}	6.5×10^{-6}	8.7×10^{-10}	-
Y-90	-	1.2×10^{-8}	6.5×10^{-6}	8.7×10^{-10}	-
Y-91	-	5.2×10^{-7}	5.4×10^{-6}	8.7×10^{-8}	-
Zr-95	-	5.8×10^{-7}	9.0×10^{-6}	1.0×10^{-7}	-
Nb-95	-	6.9×10^{-7}	1.9×10^{-5}	8.1×10^{-8}	-
Nb-95m	-	7.5×10^{-9}	1.1×10^{-7}	1.3×10^{-9}	-
Tc-99	-	1.7×10^{-12}	9.0×10^{-10}	-	-
Ru-103	-	2.5×10^{-7}	3.9×10^{-7}	9.5×10^{-8}	-
Ru-106	-	2.0×10^{-8}	5.7×10^{-6}	8.7×10^{-9}	-
Rh-103m	-	2.5×10^{-7}	3.9×10^{-7}	9.5×10^{-8}	-
Rh-106	-	2.0×10^{-8}	5.7×10^{-6}	8.7×10^{-9}	-
Ag-110m	-	6.9×10^{-10}	1.5×10^{-7}	1.2×10^{-10}	-
Sn-123	-	8.1×10^{-10}	6.9×10^{-8}	2.8×10^{-10}	-
Sb-125	-	5.2×10^{-10}	2.2×10^{-7}	1.2×10^{-10}	-
Sb-127	-	6.6×10^{-11}	-	7.5×10^{-11}	-
Te-125m	-	9.0×10^{-11}	5.2×10^{-8}	1.1×10^{-11}	-
Te-127m	-	1.6×10^{-9}	1.0×10^{-7}	4.9×10^{-10}	-
Te-127	-	1.7×10^{-9}	1.0×10^{-7}	5.5×10^{-10}	-
Te-129	-	7.8×10^{-9}	2.0×10^{-9}	3.2×10^{-9}	-
Te-129m	-	1.2×10^{-8}	3.1×10^{-9}	5.2×10^{-9}	-
Te-132	-	9.5×10^{-10}	-	5.2×10^{-10}	-
Xe-131m	8.2×10^{-2}	-	3.9×10^{-3}	-	1.0×10^{-1}
Xe-133m	1.8	-	5.5×10^{-1}	-	2.2
Xe-133	2.7×10^1	7.9×10^3	1.3×10^1	3.9×10^3	2.7×10^1
Xe-135m	2.2×10^3	-	1.2×10^2	-	3.3×10^3
Xe-135	3.6×10^2	-	1.7×10^2	-	4.1×10^2
Xe-137	4.9×10^4	-	3.9×10^4	-	4.9×10^4
Xe-138	1.3×10^4	-	1.1×10^4	-	1.1×10^4
I-129	-	2.1×10^{-6}	2.5×10^{-5}	2.8×10^{-7}	-
I-131	8.7	2.5×10^1	4.5×10^{-1}	1.4×10^1	2.8
I-132	1.1×10^3	8.3×10^{-1}	1.7	4.5×10^{-1}	3.0×10^2
I-133	1.6×10^2	-	8.8	-	4.0×10^1
I-134	4.5×10^3	-	1.2×10^2	-	1.1×10^3

TABLE I.29 (Cont'd)

Radionuclide	Releases (Ci)				
	Fuel Fabrication (IH1C1)	Fuel Processing (IH2N1) ^b	Fuel Processing (IH2C2) ^c	Plutonium-Target Processing (IH4N1) ^d	Plutonium-Target Processing (IH4C2) ^e
I-135	4.7×10^2	-	3.0×10^1	-	1.1×10^2
Cs-134	-	1.3×10^{-8}	4.9×10^{-6}	1.6×10^{-10}	-
Cs-136	-	8.7×10^{-10}	-	4.9×10^{-10}	-
Cs-137	-	1.2×10^{-8}	6.5×10^{-6}	1.1×10^{-9}	-
Ba-137m	-	1.2×10^{-8}	6.2×10^{-6}	1.1×10^{-9}	-
Ba-140	-	1.7×10^{-7}	1.2×10^{-12}	6.9×10^{-8}	-
La-140	-	1.7×10^{-7}	1.2×10^{-12}	6.9×10^{-8}	-
Ce-141	-	4.3×10^{-7}	1.9×10^{-7}	1.2×10^{-7}	-
Ce-144	-	3.2×10^{-7}	7.1×10^{-5}	2.9×10^{-8}	-
Pr-143	-	1.9×10^{-7}	3.7×10^{-12}	7.2×10^{-8}	-
Pr-144	-	3.2×10^{-7}	7.1×10^{-5}	2.9×10^{-8}	-
Pr-144m	-	3.8×10^{-9}	8.6×10^{-7}	3.8×10^{-10}	-
Pm-147	-	2.2×10^{-8}	-	4.0×10^{-9}	-
Pm-148m	-	4.3×10^{-9}	8.3×10^{-9}	5.2×10^{-10}	-
Nd-147	-	4.6×10^{-8}	-	2.0×10^{-8}	-
Sm-151	-	3.8×10^{-11}	2.0×10^{-8}	1.8×10^{-11}	-
Eu-154	-	4.9×10^{-10}	2.5×10^{-7}	1.1×10^{-11}	-
Eu-155	-	2.9×10^{-10}	1.4×10^{-7}	6.9×10^{-11}	-
Np-237	-	-	3.7×10^{-11}	-	-
Pu-238	-	2.9×10^{-10}	1.5×10^{-7}	1.3×10^{-12}	1.2×10^{-9}
Pu-239	-	2.0×10^{-12}	1.0×10^{-9}	5.8×10^{-11}	5.4×10^{-11}
Pu-240	-	1.6×10^{-12}	8.8×10^{-10}	1.0×10^{-11}	1.2×10^{-10}
Pu-241	-	5.5×10^{-10}	2.8×10^{-7}	7.5×10^{-10}	3.6×10^{-8}
Pu-242	-	-	-	-	8.6×10^{-13}
Am-241	-	-	5.1×10^{-10}	-	4.8×10^{-11}
Cm-242	-	2.9×10^{-11}	3.7×10^{-9}	1.6×10^{-12}	-
Cm-244	-	-	2.6×10^{-12}	-	-

^aThe source term for the highest consequence accident at the tritium-target processing facility (IH6D1) is estimated at 1×10^6 Ci of HTO.

^bHighest consequence accident for the maximally exposed off-site individual and worst-sector population.

^cHighest consequence accident for the maximally exposed colocated worker.

^dHighest consequence accident for the maximally exposed off-site individual.

^eHighest consequence accident for the worst-sector population and maximally exposed colocated worker.

Sources: IH1C1 - NRC 1979; IH6D1 - Cornwell 1990; all other accidents - EG&G 1989a.

TABLE I.30 Estimated Releases from the Highest Consequence LWR
Nonsevere Accidents at INEL Support Facilities

Radionuclide	Releases (Ci)			
	Fuel Fabrication (IL1C1)	Fuel Processing (IL2C1)	Plutonium- Target Processing (IL4C1)	Tritium- Target Processing (IL6D1)
H-3	-	-	-	1.0×10^6
Kr-83m	1.6×10^2	3.7×10^1	1.1×10^2	-
Kr-85m	1.5×10^2	1.7×10^2	7.1×10^1	-
Kr-85	1.6×10^{-3}	1.6×10^{-3}	8.1×10^{-4}	-
Kr-87	9.9×10^2	1.0×10^3	4.3×10^2	-
Kr-88	6.5×10^2	6.6×10^2	2.3×10^2	-
Kr-89	4.2×10^4	-	1.3×10^4	-
Xe-131m	8.2×10^{-2}	3.9×10^{-3}	1.0×10^{-1}	-
Xe-133m	1.8	5.5×10^{-1}	2.2	-
Xe-133	2.7×10^1	1.3×10^1	2.7×10^1	-
Xe-135m	2.2×10^3	-	3.3×10^3	-
Xe-135	3.6×10^2	1.7×10^2	4.1×10^2	-
Xe-137	4.9×10^4	-	4.9×10^4	-
Xe-138	1.3×10^4	1.1×10^4	1.1×10^4	-
I-129	-	4.3×10^{-10}	-	-
I-131	8.7	1.8	2.8	-
I-132	1.1×10^3	6.7	3.0×10^2	-
I-133	1.6×10^2	3.5×10^1	4.0×10^1	-
I-134	4.5×10^3	4.8×10^2	1.1×10^3	-
I-135	4.7×10^2	1.2×10^2	1.1×10^2	-
Pu-238	-	-	1.2×10^{-9}	-
Pu-239	-	-	5.4×10^{-11}	-
Pu-240	-	-	1.2×10^{-10}	-
Pu-241	-	-	3.6×10^{-8}	-
Pu-242	-	-	8.6×10^{-13}	-
Am-241	-	-	4.8×10^{-11}	-

Sources: IL1C1 - NRC 1979; IL2C1 - WHC 1990b; IL4C1 - EG&G 1989c;
IL6D1 - Cornwell 1990.

TABLE I.31 Estimated Releases from the Highest Consequence MHTGR Nonsevere Accidents at INEL Support Facilities^a

Radionuclide	Releases (Ci)				
	Fuel Fabrication (IM1C1)	Fuel Processing (IM2F1) ^b	Fuel Processing (IM2C2) ^c	Plutonium-Target Processing (IM4F1) ^b	Plutonium-Target Processing (IM4C2) ^c
H-3	-	1.0	-	1.5×10^{-2}	-
C-14	-	5.1×10^{-4}	-	4.7×10^{-3}	-
Kr-83m	1.6×10^2	-	3.7×10^1	-	1.1×10^2
Kr-85m	1.5×10^2	-	1.7×10^2	-	7.1×10^1
Kr-85	1.6×10^{-3}	7.8×10^2	1.6×10^{-3}	2.7×10^1	8.1×10^{-4}
Kr-87	9.9×10^2	-	1.0×10^3	-	4.3×10^2
Kr-88	6.5×10^2	-	6.6×10^2	-	2.3×10^2
Kr-89	4.2×10^4	-	4.1×10^4	-	1.3×10^4
Rb-86	-	1.0×10^{-8}	6.2×10^{-12}	7.4×10^{-11}	-
Sr-89	-	1.2×10^{-6}	2.0×10^{-6}	4.5×10^{-7}	-
Sr-90	-	1.8×10^{-7}	2.8×10^{-5}	6.1×10^{-9}	-
Y-90	-	1.8×10^{-7}	2.8×10^{-5}	6.1×10^{-9}	-
Y-91	-	1.7×10^{-6}	5.0×10^{-6}	5.3×10^{-7}	-
Zr-95	-	2.0×10^{-6}	8.5×10^{-6}	5.7×10^{-7}	-
Nb-95	-	2.6×10^{-6}	1.9×10^{-5}	5.2×10^{-7}	-
Nb-95m	-	1.5×10^{-8}	6.3×10^{-8}	4.2×10^{-9}	-
Mo-99	-	1.4×10^{-9}	-	7.7×10^{-10}	-
Tc-99	-	1.6×10^{-11}	2.5×10^{-9}	-	-
Tc-99m	-	1.4×10^{-9}	-	7.4×10^{-10}	-
Ru-103	-	1.1×10^{-4}	5.1×10^{-7}	5.7×10^{-5}	-
Ru-106	-	3.0×10^{-5}	2.8×10^{-5}	1.0×10^{-5}	-
Rh-103m	-	1.1×10^{-6}	4.6×10^{-7}	5.7×10^{-7}	-
Rh-106	-	3.3×10^{-7}	2.8×10^{-5}	1.1×10^{-7}	-
Ag-110m	-	6.9×10^{-11}	4.3×10^{-9}	1.4×10^{-12}	-
Sn-123	-	2.9×10^{-9}	6.8×10^{-8}	1.3×10^{-9}	-
Sb-125	-	1.1×10^{-8}	1.4×10^{-6}	1.8×10^{-9}	-
Sb-127	-	5.5×10^{-10}	-	5.3×10^{-10}	-
Te-125m	-	2.4×10^{-9}	3.4×10^{-7}	2.1×10^{-10}	-
Te-127m	-	1.1×10^{-8}	2.1×10^{-7}	4.7×10^{-9}	-
Te-127	-	1.2×10^{-7}	2.1×10^{-7}	5.1×10^{-9}	-
Te-129	-	2.1×10^{-8}	3.3×10^{-9}	1.3×10^{-8}	-
Te-129m	-	3.3×10^{-8}	5.1×10^{-9}	2.0×10^{-8}	-
Te-132	-	3.4×10^{-9}	-	2.0×10^{-9}	-
Xe-131m	8.2×10^{-2}	2.2×10^2	3.9×10^{-3}	1.0×10^2	1.0×10^{-1}
Xe-133m	1.8	3.8×10^{-1}	5.5×10^{-1}	2.1×10^{-1}	2.2
Xe-133	2.7×10^1	2.4×10^3	1.3×10^1	1.3×10^3	2.7×10^1
Xe-135m	2.2×10^3	-	1.1×10^2	-	3.3×10^3
Xe-135	3.6×10^2	-	1.7×10^2	-	4.1×10^2
Xe-137	4.9×10^4	-	3.9×10^4	-	4.9×10^4
Xe-138	1.3×10^4	-	1.1×10^4	-	1.1×10^4
I-129	-	1.5×10^{-5}	1.7×10^{-6}	9.4×10^{-7}	-

TABLE I.31 (Cont'd)

Radionuclide	Releases (Ci)				
	Fuel Fabrication (IM1C1)	Fuel Processing (IM2F1) ^b	Fuel Processing (IM2C2) ^c	Plutonium-Target Processing (IM4F1) ^b	Plutonium-Target Processing (IM4C2) ^c
I-131	8.7	4.5×10^1	3.9×10^1	2.3×10^1	2.8
I-132	1.1×10^3	1.2	1.7	7.2×10^{-1}	3.0×10^2
I-133	1.6×10^2	3.9×10^{-8}	8.8	2.2×10^{-8}	4.0×10^1
I-134	4.5×10^3	-	1.2×10^2	-	1.1×10^3
I-135	4.7×10^2	-	3.0×10^1	-	1.1×10^2
Cs-134	-	8.7×10^{-7}	1.0×10^{-4}	3.0×10^{-9}	-
Cs-136	-	1.3×10^{-7}	-	5.3×10^{-9}	-
Cs-137	-	2.0×10^{-7}	3.1×10^{-5}	6.5×10^{-9}	-
Ba-137m	-	1.9×10^{-7}	2.9×10^{-5}	6.2×10^{-9}	-
Ba-140	-	5.5×10^{-7}	1.1×10^{-12}	2.8×10^{-7}	-
La-140	-	6.3×10^{-7}	1.3×10^{-12}	3.2×10^{-7}	-
Ce-141	-	1.4×10^{-6}	1.8×10^{-7}	5.5×10^{-7}	-
Ce-144	-	2.1×10^{-6}	1.5×10^{-4}	1.8×10^{-7}	-
Pr-143	-	5.7×10^{-7}	3.3×10^{-12}	3.1×10^{-7}	-
Pr-144	-	2.1×10^{-6}	1.5×10^{-4}	1.8×10^{-7}	-
Pr-144m	-	2.5×10^{-8}	1.8×10^{-6}	2.2×10^{-9}	-
Pm-148m	-	5.2×10^{-9}	2.4×10^{-10}	2.4×10^{-9}	-
Nd-147	-	1.6×10^{-7}	-	8.1×10^{-8}	-
Sm-151	-	6.0×10^{-10}	9.4×10^{-8}	2.0×10^{-10}	-
Eu-154	-	2.4×10^{-8}	3.5×10^{-6}	1.2×10^{-10}	-
Eu-155	-	1.7×10^{-8}	2.3×10^{-6}	8.5×10^{-10}	-
Np-237	-	1.8×10^{-12}	2.9×10^{-10}	-	-
Np-239	-	1.0×10^{-9}	2.4×10^{-9}	-	-
Pu-238	-	4.3×10^{-8}	6.8×10^{-6}	1.5×10^{-11}	1.2×10^{-9}
Pu-239	-	2.5×10^{-11}	3.9×10^{-9}	4.8×10^{-10}	5.4×10^{-11}
Pu-240	-	2.7×10^{-11}	4.4×10^{-9}	2.3×10^{-10}	1.2×10^{-10}
Pu-241	-	2.0×10^{-8}	3.1×10^{-6}	2.2×10^{-8}	3.6×10^{-8}
Pu-242	-	5.7×10^{-13}	8.9×10^{-11}	1.7×10^{-14}	8.6×10^{-13}
Am-241	-	4.9×10^{-12}	5.4×10^{-9}	5.0×10^{-12}	4.8×10^{-11}
Am-243	-	1.5×10^{-11}	2.4×10^{-9}	-	-
Cm-242	-	8.3×10^{-9}	3.1×10^{-7}	8.8×10^{-11}	-
Cm-244	-	6.8×10^{-9}	1.0×10^{-6}	1.0×10^{-13}	-

^aThe source term for the highest consequence accident at the tritium-target processing facility (IM6D1) is estimated at 1×10^6 Ci of HTO.

^bHighest consequence accident for the maximally exposed off-site individual and worst-sector population.

^cHighest consequence accident for the maximally exposed colocated worker.

Sources: IM1C1 - NRC 1979; IM6D1 - Cornwell 1990; all other accidents - EG&G 1989c.

**TABLE I.32 Estimated Releases from the Highest Consequence
HWR Nonsevere Accidents at SRS Support Facilities**

Radio- nuclide	Releases (Ci)			
	Fuel Fabrication (SH1C1)	Fuel Processing (SH2N1)	Plutonium- Target Processing (SH4N1)	Tritium- Target Processing (SH6D1)
H-3	-	-	-	1.0×10^6
Kr-83m	1.6×10^2	-	-	-
Kr-85m	1.5×10^2	-	-	-
Kr-85	1.6×10^{-3}	6.2×10^3	1.7×10^3	-
Kr-87	9.9×10^2	-	-	-
Kr-88	6.5×10^2	-	-	-
Kr-89	4.2×10^4	-	-	-
Xe-131m	8.2×10^{-2}	-	-	-
Xe-133m	1.8	-	-	-
Xe-133	2.7×10^1	1.0×10^5	1.8×10^5	-
Xe-135m	2.2×10^3	-	-	-
Xe-135	3.6×10^2	-	-	-
Xe-137	4.9×10^4	-	-	-
Xe-138	1.3×10^4	-	-	-
I-131	8.7	6.6×10^2	2.6×10^2	-
I-132	1.1×10^3	2.1×10^1	8.3	-
I-133	1.6×10^2	-	-	-
I-134	4.5×10^3	-	-	-
I-135	4.7×10^2	-	-	-

Sources: SH1C1 - NRC 1979; SH2N1, SN4N1 - WSRC 1991b; SH6D1 - Cornwell 1990.

TABLE I.33 Estimated Releases from the Highest Consequence
LWR Nonsevere Accidents at SRS Support Facilities

Radio- nuclide	Releases (Ci)			
	Fuel Fabrication (SL1C1)	Fuel Processing (SL2N2) ^a	Fuel Processing (SL2F1) ^b	Tritium- Target Processing (SL6D1)
H-3	-	-	2.3×10^1	1.0×10^6
Mn-54	-	-	9.6×10^{-8}	-
Fe-55	-	-	6.4×10^{-8}	-
Co-60	-	-	2.1×10^{-7}	-
Ni-59	-	-	2.5×10^{-10}	-
Ni-63	-	-	3.2×10^{-8}	-
Kr-83m	1.6×10^2	-	-	-
Kr-85m	1.5×10^2	-	-	-
Kr-85	1.6×10^{-3}	3.8×10^1	-	-
Kr-87	9.9×10^2	-	-	-
Kr-88	6.5×10^2	-	-	-
Kr-89	4.2×10^4	-	-	-
Zr-95	-	-	9.2×10^{-8}	-
Ru-103	-	-	7.6×10^{-2}	-
Ru-106	-	-	1.8	-
Sn-119n	-	-	1.6×10^{-7}	-
Sb-125	-	-	6.4×10^{-7}	-
Xe-131m	8.2×10^{-2}	3.9×10^1	-	-
Xe-133m	1.8	-	-	-
Xe-133	2.7×10^1	4.6×10^2	-	-
Xe-135m	2.2×10^3	-	-	-
Xe-135	3.6×10^2	-	-	-
Xe-137	4.9×10^4	-	-	-
Xe-138	1.3×10^4	-	-	-
I-131	8.7	2.3×10^2	-	-
I-132	1.1×10^3	7.7	-	-
I-133	1.6×10^2	-	-	-
I-134	4.5×10^3	-	-	-
I-135	4.7×10^2	-	-	-
U-235	-	-	1.1×10^{-6}	-
Pu-239	-	-	6.2×10^{-9}	-

^aHighest consequence accident for the maximally exposed off-site individual and worst-sector population.

^bHighest consequence accident for the maximally exposed colocated worker.

Sources: SL1C1 - NRC 1979; SL2N2, SL2F1 - WSRC 1991c; SL6D1 - Cornwell 1990.

TABLE I.34 Estimated Releases from the Highest Consequence MHTGR Nonsevere Accidents at SRS Support Facilities

Radio-nuclide	Releases (Ci)			
	Fuel Fabrication (SM1C1)	Fuel Processing (SM2F1)	Plutonium-Target Processing (SM4F1)	Tritium-Target Processing (SM6D1)
H-3	-	1.0	1.0	1.0×10^6
C-14	-	1.2×10^{-3}	1.1×10^{-2}	-
Kr-83m	1.6×10^2	-	-	-
Kr-85m	1.5×10^2	-	-	-
Kr-85	1.6×10^{-3}	7.8×10^2	2.7×10^1	-
Kr-87	9.9×10^2	-	-	-
Kr-88	6.5×10^2	-	-	-
Kr-89	4.2×10^4	-	-	-
Rb-86	-	1.0×10^{-8}	-	-
Sr-89	-	1.2×10^{-6}	4.5×10^{-7}	-
Sr-90	-	1.8×10^{-7}	6.1×10^{-9}	-
Y-90	-	1.8×10^{-7}	6.1×10^{-9}	-
Y-91	-	1.7×10^{-6}	5.3×10^{-7}	-
Zr-95	-	2.0×10^{-6}	5.7×10^{-7}	-
Nb-95	-	2.6×10^{-6}	5.2×10^{-7}	-
Nb-95m	-	1.5×10^{-8}	4.2×10^{-9}	-
Mo-99	-	1.4×10^{-9}	7.7×10^{-10}	-
Tc-99m	-	1.4×10^{-9}	7.4×10^{-10}	-
Ru-103	-	1.1×10^{-4}	5.7×10^{-5}	-
Ru-106	-	3.0×10^{-5}	1.0×10^{-5}	-
Rh-103m	-	1.1×10^{-6}	5.7×10^{-7}	-
Sn-123	-	2.6×10^{-9}	1.3×10^{-9}	-
Sb-125	-	1.1×10^{-8}	1.8×10^{-9}	-
Sb-127	-	5.5×10^{-10}	5.3×10^{-10}	-
Te-125m	-	2.4×10^{-9}	2.1×10^{-10}	-
Te-127	-	1.2×10^{-8}	5.1×10^{-9}	-
Te-127m	-	1.1×10^{-8}	4.7×10^{-9}	-
Te-129	-	2.1×10^{-8}	1.3×10^{-8}	-
Te-129m	-	3.3×10^{-8}	2.0×10^{-8}	-
Te-132	-	3.4×10^{-9}	2.0×10^{-9}	-
Xe-131m	8.2×10^{-2}	2.2×10^2	1.0×10^2	-
Xe-133m	1.8	3.8×10^{-1}	2.1×10^{-1}	-
Xe-133	2.7×10^1	2.4×10^3	1.3×10^3	-
Xe-135m	2.2×10^3	-	-	-
Xe-135	3.6×10^2	-	-	-
Xe-137	4.9×10^4	-	-	-
Xe-138	1.3×10^4	-	-	-
I-129	-	1.0×10^{-5}	9.4×10^{-7}	-
I-131	8.7	3.9×10^1	2.3×10^1	-
I-132	1.1×10^3	1.2	7.2×10^{-1}	-
I-133	1.6×10^2	3.9×10^{-8}	2.2×10^{-8}	-
I-134	4.5×10^3	-	-	-

TABLE I.34 (Cont'd)

Radio-nuclide	Releases			
	Fuel Fabrication (SM1C1)	Fuel Processing (SM2F1)	Plutonium-Target Processing (SM4F1)	Tritium-Target Processing (SM6D1)
I-135	4.7×10^2	-	-	-
Cs-134	-	8.7×10^{-7}	3.0×10^{-9}	-
Cs-136	-	1.3×10^{-7}	5.3×10^{-9}	-
Cs-137	-	2.0×10^{-7}	6.5×10^{-9}	-
Ba-140	-	5.5×10^{-7}	2.8×10^{-7}	-
La-140	-	6.3×10^{-7}	3.2×10^{-7}	-
Ce-141	-	1.4×10^{-6}	5.5×10^{-7}	-
Ce-144	-	2.1×10^{-6}	1.8×10^{-7}	-
Pr-143	-	5.7×10^{-7}	3.1×10^{-7}	-
Pr-144	-	2.1×10^{-6}	1.8×10^{-7}	-
Pr-144m	-	2.8×10^{-8}	2.2×10^{-9}	-
Nd-147	-	1.6×10^{-7}	8.1×10^{-8}	-
Pm-148m	-	5.2×10^{-9}	2.4×10^{-9}	-
Sm-151	-	6.0×10^{-10}	2.0×10^{-10}	-
Eu-154	-	2.4×10^{-8}	1.2×10^{-10}	-
Eu-155	-	1.7×10^{-8}	8.5×10^{-10}	-
Np-239	-	1.0×10^{-9}	-	-
Pu-238	-	4.3×10^{-8}	-	-
Pu-239	-	-	4.8×10^{-10}	-
Pu-240	-	-	2.3×10^{-10}	-
Pu-241	-	2.0×10^{-8}	2.3×10^{-8}	-
Cm-242	-	8.3×10^{-9}	-	-
Cm-244	-	6.8×10^{-9}	-	-

Sources: SM1C1 - NRC 1979; SM2F1, SM4F1 - WSRC 1991d; SM6D1 - Cornwell 1990.

TABLE I.35 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Off-site Individual from Nonsevere Accidents at Hanford Support Facilities^a

Technology/ Accident ^b	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^c	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR					
HH1C1	3.8×10^{-2}	Thyroid	9.7×10^{-1}	9.7×10^{-1}	2×10^{-5}
HH2C2	3.9×10^{-2}	Thyroid	9.6×10^{-1}	9.6×10^{-1}	2×10^{-3}
HH4N1	1.6×10^{-1}	Thyroid	5.2	5.2	6×10^{-5}
HH6D1	4.3	S. int.	4.9	4.7	2×10^{-3}
LWR					
HL1C1	3.5×10^{-2}	Thyroid	7.0×10^{-1}	7.0×10^{-1}	1×10^{-4}
HL2N2	2.2×10^{-1}	Thyroid	7.0	7.0	9×10^{-5}
HL4C1	3.1×10^{-2}	Thyroid	7.8×10^{-1}	7.8×10^{-1}	1×10^{-5}
HL6D1	4.3	S. int.	4.9	4.7	2×10^{-3}
MHTGR					
HM1C1	3.8×10^{-2}	Thyroid	9.7×10^{-1}	9.7×10^{-1}	2×10^{-5}
HM2N1	3.7×10^{-2}	Thyroid	1.2	1.2	2×10^{-5}
HM4N1	2.2×10^{-2}	Thyroid	7.0×10^{-1}	7.0×10^{-1}	9×10^{-6}
HM6D1	4.3	S. int.	4.9	4.7	2×10^{-3}

^aLocation of maximally exposed off-site individual: 7,500 m south-southwest for tritium-target processing and LWR fuel fabrication accidents; 15,600 m east-southeast for HWR and MHTGR fuel fabrication accidents; and 17,000 m east for fuel processing and plutonium-target processing accidents.

^bSee Table I.23 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

^cS. int. = small intestine.

TABLE I.36 Estimated Doses by Pathway to the Maximally Exposed Off-Site Individual from Nonsevere Accidents at Hanford Support Facilities^a

Technology/ Accident ^b	CEDE (rem)		EDE (rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR				
HH1C1	1.8×10^{-3}	2.9×10^{-2}	2.9×10^{-3}	4.3×10^{-3}
HH2C2	1.7×10^{-3}	2.9×10^{-2}	3.7×10^{-3}	4.7×10^{-3}
HH4N1	1.4×10^{-3}	1.6×10^{-1}	6.6×10^{-4}	1.4×10^{-3}
HH6D1	5.0×10^{-2}	4.2	0	0
LWR				
HL1C1	1.8×10^{-3}	2.9×10^{-2}	2.9×10^{-3}	4.3×10^{-3}
HL2N2	1.9×10^{-3}	2.1×10^{-1}	1.4×10^{-5}	1.9×10^{-3}
HL4C1	1.4×10^{-3}	2.3×10^{-2}	2.9×10^{-3}	3.7×10^{-3}
HL6D1	5.0×10^{-2}	4.2	0	0
MHTGR				
HM1C1	1.8×10^{-3}	2.9×10^{-2}	2.9×10^{-3}	4.3×10^{-3}
HM2N1	3.4×10^{-4}	3.6×10^{-2}	1.6×10^{-5}	3.4×10^{-4}
HM4N1	1.9×10^{-4}	2.1×10^{-2}	8.5×10^{-6}	1.9×10^{-4}
HM6D1	5.0×10^{-2}	4.2	0	0

^aLocation of maximally exposed off-site individual: 7,500 m south-southeast for tritium-target processing and LWR fuel fabrication accidents; 15,600 m east-southeast for HWR and MHTGR fuel fabrication accidents; and 17,000 m east for fuel processing and plutonium-target processing accidents.

^bSee Table I.23 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

TABLE I.37 Estimated Population Doses and Health Effects from Nonsevere Accidents at Hanford Support Facilities^a

Technology/ Accident ^b	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR					
HH1C1	9.9×10^2	9.3	1.7×10^4	4×10^{-1}	1×10^{-1}
HH2C2	4.1×10^2	3.5	1.0×10^4	2×10^{-1}	3×10^{-2}
HH4N1	2.1×10^3	1.1×10^1	6.8×10^4	8×10^{-1}	1×10^{-2}
HH6D1	6.7×10^4	7.4×10^4	7.4×10^4	3×10^1	2×10^1
LWR					
HL1C1	7.3×10^2	6.1	9.3×10^3	3×10^{-1}	1×10^{-1}
HL2N2	2.8×10^3	1.5×10^1	9.1×10^4	1	1×10^{-2}
HL4C1	3.4×10^2	2.6	9.3×10^3	1×10^{-1}	1×10^{-2}
HL6D1	6.7×10^4	7.4×10^4	7.4×10^4	3×10^1	2×10^1
MHTGR					
HM1C1	9.9×10^2	9.3	1.7×10^4	4×10^{-1}	1×10^{-1}
HM2N1	4.8×10^2	2.5	1.5×10^4	2×10^{-1}	2×10^{-3}
HM4N1	2.8×10^2	1.5	9.1×10^3	1×10^{-1}	1×10^{-3}
HM6D1	6.7×10^4	7.4×10^4	7.4×10^4	3×10^1	2×10^1

^aWorst-sector population: 84,000 persons south-southeast for tritium-target processing and LWR fuel fabrication accidents; 104,000 persons south-southeast for HWR and MHTGR fuel fabrication accidents; and 146,000 persons southeast for fuel processing and plutonium-target processing accidents.

^bSee Table I.23 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

TABLE I.38 Estimated Doses by Pathway to the Worst-Sector Population from Nonsevere Accidents at Hanford Support Facilities^a

Technology/ Accident ^b	CEDE (person-rem)		EDE (person-rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR				
HH1C1	7.2×10^1	4.8×10^2	3.1×10^2	1.3×10^2
HH2C2	2.9×10^1	2.9×10^2	4.6×10^1	4.8×10^1
HH4N1	3.4×10^1	2.0×10^3	1.8×10^1	2.4×10^1
HH6D1	1.4×10^3	6.6×10^4	1.1×10^{-6}	0
LWR				
HL1C1	7.2×10^1	4.8×10^2	3.1×10^2	1.3×10^2
HL2N2	4.6×10^1	2.7×10^3	3.6×10^{-1}	3.2×10^1
HL4C1	2.2×10^1	2.7×10^2	2.0×10^1	3.2×10^1
HL6D1	1.4×10^3	6.6×10^4	1.1×10^{-6}	0
MHTGR				
HM1C1	7.2×10^1	4.8×10^2	3.1×10^2	1.3×10^2
HM2N1	8.0	4.6×10^2	4.2×10^{-1}	5.3
HM4N1	4.6	2.7×10^2	2.3×10^{-1}	3.2
HM6D1	1.4×10^3	6.6×10^4	1.1×10^{-6}	0

^aWorst-sector population: 84,000 persons south-southeast for tritium-target processing and LWR fuel fabrication accidents; 104,000 persons south-southeast for HWR and MHTGR fuel fabrication accidents; and 146,000 persons southeast for fuel processing and plutonium-target processing accidents.

^bSee Table I.23 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

TABLE I.39 Estimated Maximum Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Colocated Worker from Nonsevere Accidents at Hanford Support Facilities^a

Technology/ Accident ^b	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^c	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR					
HH1C1	2.4	Thyroid	6.9	6.9	1×10^{-3}
HH2C1	1.3×10^{-1}	Thyroid	9.9×10^{-2}	9.9×10^{-2}	5×10^{-5}
HH4C1	6.3×10^{-2}	Thyroid	2.0×10^{-1}	2.0×10^{-1}	3×10^{-5}
HH6D1	3.6×10^{-1}	S. int.	4.3×10^{-1}	4.0×10^{-1}	1×10^{-4}
LWR					
HL1C1	1.9×10^{-1}	Thyroid	2.3×10^{-1}	2.3×10^{-1}	8×10^{-5}
HL2C2	8.2×10^{-2}	Thyroid	4.2×10^{-1}	4.2×10^{-1}	3×10^{-5}
HL4C1	6.3×10^{-2}	Thyroid	2.0×10^{-1}	2.0×10^{-1}	3×10^{-5}
HL6D1	3.6×10^{-1}	S. int.	4.3×10^{-1}	4.0×10^{-1}	1×10^{-4}
MHTGR					
HM1C1	2.4	Thyroid	6.9	6.9	1×10^{-3}
HM2C2	1.0×10^{-1}	Thyroid	1.0×10^{-2}	1.0×10^{-2}	4×10^{-5}
HM4C1	5.9×10^{-2}	Thyroid	2.6×10^{-2}	2.6×10^{-2}	2×10^{-5}
HM6D1	3.6×10^{-1}	S. int.	4.3×10^{-1}	4.0×10^{-1}	1×10^{-4}

^aLocation of maximally exposed colocated worker: 200 m west-southwest for HWR and MHTGR fuel fabrication accidents; 400 m west-southwest for tritium-target processing and LWR fuel fabrication accidents; 600 m southwest for accidents in plutonium-target processing and fuel processing accidents in the fuel and plutonium-target processing facility; and 700 m west-southwest for fuel processing accidents in the uranium processing facility.

^bSee Table I.23 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

^cS. int. = small intestine.

TABLE I.40 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Off-site Individual from Nonsevere Accidents at INEL Support Facilities^a

Technology/ Accident ^b	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^c	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR					
IH1C1	1.8×10^{-2}	Thyroid	3.8×10^{-1}	3.8×10^{-1}	7×10^{-6}
IH2N1	3.0×10^{-2}	Thyroid	9.6×10^{-1}	9.6×10^{-1}	1×10^{-5}
IH4N1	1.7×10^{-2}	Thyroid	5.4×10^{-1}	5.4×10^{-1}	7×10^{-6}
IH6D1	2.1	S. int.	2.5	2.4	8×10^{-4}
LWR					
IL1C1	1.8×10^{-2}	Thyroid	3.8×10^{-1}	3.8×10^{-1}	7×10^{-6}
IL2C1	4.8×10^{-3}	Thyroid	8.3×10^{-2}	8.3×10^{-2}	2×10^{-6}
IL4C1	1.1×10^{-2}	Thyroid	2.2×10^{-1}	2.2×10^{-1}	4×10^{-6}
IL6D1	2.1	S. int.	2.5	2.4	8×10^{-4}
MHTGR					
IM1C1	1.8×10^{-2}	Thyroid	3.8×10^{-1}	3.8×10^{-1}	7×10^{-6}
IM2F1	4.6×10^{-2}	Thyroid	1.5	1.5	2×10^{-5}
IM4F1	2.7×10^{-2}	Thyroid	8.8×10^{-1}	8.8×10^{-1}	1×10^{-5}
IM6D1	2.1	S. int.	2.5	2.4	8×10^{-4}

^aLocation of maximally exposed off-site individual: 15,500 m south-southwest for accidents at fuel fabrication and tritium-target processing facilities; 14,400 m south-southwest for accidents at fuel processing and plutonium-target processing facilities.

^bSee Table I.24 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

^cS. int. = small intestine.

TABLE I.41 Estimated Doses by Pathway to the Maximally Exposed Off-site Individual from Nonsevere Accidents at INEL Support Facilities^a

Technology/ Accident ^b	CEDE (rem)		EDE (rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR				
IH1C1	9.4×10^{-4}	1.1×10^{-2}	3.6×10^{-3}	2.4×10^{-3}
IH2N1	2.7×10^{-4}	2.9×10^{-2}	6.4×10^{-5}	2.6×10^{-4}
IH4N1	1.5×10^{-4}	1.6×10^{-2}	3.2×10^{-5}	1.5×10^{-4}
IH6D1	2.8×10^{-2}	2.1	0	0
LWR				
IL1C1	9.4×10^{-4}	1.1×10^{-2}	3.6×10^{-3}	2.4×10^{-3}
IL2C1	2.5×10^{-4}	2.5×10^{-3}	1.7×10^{-3}	4.0×10^{-4}
IL4C1	5.7×10^{-4}	6.5×10^{-3}	2.8×10^{-3}	1.2×10^{-3}
IL6D1	2.8×10^{-2}	2.1	0	0
MHTGR				
IM1C1	9.4×10^{-4}	1.1×10^{-2}	3.6×10^{-3}	2.4×10^{-3}
IM2F1	4.3×10^{-4}	4.6×10^{-2}	2.3×10^{-5}	4.1×10^{-4}
IM4F1	2.5×10^{-4}	2.7×10^{-2}	1.2×10^{-5}	2.4×10^{-4}
IM6D1	2.8×10^{-2}	2.1	0	0

^aLocation of maximally exposed off-site individual: 15,500 m south-southwest for accidents at fuel fabrication and tritium-target processing facilities; 14,400 m south-southwest for accidents at fuel processing and plutonium-target processing facilities.

^bSee Table I.24 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

TABLE I.42 Estimated Population Doses and Health Effects from Nonsevere Accidents at INEL Support Facilities^a

Technology/ Accident ^b	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR					
IH1C1	8.0×10^1	9.4×10^{-1}	5.4×10^2	3×10^{-2}	2×10^{-2}
IH2N1	2.9×10^1	1.3×10^{-1}	7.8×10^2	1×10^{-2}	2×10^{-3}
IH4C2	1.9×10^1	2.5×10^{-1}	2.0×10^2	8×10^{-3}	3×10^{-3}
IH6D1	3.8×10^3	4.2×10^3	4.2×10^3	2	1
LWR					
IL1C1	8.0×10^1	9.4×10^{-1}	5.4×10^2	3×10^{-2}	2×10^{-2}
IL2C1	2.0×10^1	1.7×10^{-1}	1.0×10^2	8×10^{-3}	4×10^{-3}
IL4C1	1.9×10^1	2.5×10^{-1}	2.0×10^2	8×10^{-3}	3×10^{-3}
IL6D1	3.8×10^3	4.2×10^3	4.2×10^3	2	1
MHTGR					
IM1C1	8.0×10^1	9.4×10^{-1}	5.4×10^2	3×10^{-2}	2×10^{-2}
IM2F1	4.4×10^1	2.0×10^{-1}	1.2×10^3	2×10^{-2}	2×10^{-3}
IM4F1	2.6×10^1	1.1×10^{-1}	7.1×10^2	1×10^{-2}	1×10^{-3}
IM6D1	3.8×10^3	4.2×10^3	4.2×10^3	2	1

^aWorst-sector population: 113,000 persons east for accidents at fuel fabrication and tritium-target fabrication facilities; 101,000 persons east for accidents at fuel processing and plutonium-target processing facilities.

^bSee Table I.24 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

TABLE I.43 Estimated Doses by Pathway to the Worst-Sector Population from Nonsevere Accidents at INEL Support Facilities^a

Technology/ Accident ^b	CEDE (person-rem)		EDE (person-rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR				
IH1C1	1.3×10^1	7.8	3.1×10^1	2.8×10^1
IH2N1	4.3	2.0×10^1	1.4	4.0
IH4C2	3.7	3.4	5.4	6.8
IH6D1	5.4×10^2	3.3×10^3	0	0
LWR				
IL1C1	1.3×10^1	7.8	3.1×10^1	2.8×10^1
IL2C1	2.6	1.4	1.2×10^1	4.2
IL4C1	3.7	3.4	5.4	6.8
IL6D1	5.4×10^2	3.3×10^3	0	0
MHTGR				
IM1C1	1.3×10^1	7.8	3.1×10^1	2.8×10^1
IM2F1	6.8	3.0×10^1	5.0×10^{-1}	6.2
IM4F1	4.0	1.8×10^1	2.7×10^{-1}	3.7
IM6D1	5.4×10^2	3.3×10^3	0	0

^aWorst-sector population: 113,000 persons east for accidents at fuel fabrication and tritium-target fabrication facilities; 101,000 persons east for accidents at fuel processing and plutonium-target processing facilities.

^bSee Table I.24 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

TABLE I.44 Estimated Maximum Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Colocated Worker from Nonsevere Accidents at INEL Support Facilities^a

Technology/ Accident ^b	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^c	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR					
IH1C1	3.8×10^{-2}	Thyroid	5.0×10^{-2}	5.0×10^{-2}	2×10^{-5}
IH2C2	4.0×10^{-2}	Thyroid	1.1×10^{-2}	1.1×10^{-2}	2×10^{-5}
IH4C2	3.1×10^{-2}	Thyroid	5.6×10^{-2}	5.6×10^{-2}	1×10^{-5}
IH6D1	7.8×10^{-2}	S. int.	9.2×10^{-2}	8.5×10^{-2}	3×10^{-5}
LWR					
IL1C1	3.8×10^{-2}	Thyroid	5.0×10^{-2}	5.0×10^{-2}	2×10^{-5}
IL2C1	9.5×10^{-3}	Thyroid	1.1×10^{-2}	1.1×10^{-2}	4×10^{-6}
IL4C1	3.1×10^{-2}	Thyroid	5.6×10^{-2}	5.6×10^{-2}	1×10^{-5}
IL6D1	7.8×10^{-2}	S. int.	9.2×10^{-2}	8.5×10^{-2}	3×10^{-5}
MHTGR					
IM1C1	3.8×10^{-2}	Thyroid	5.0×10^{-2}	5.0×10^{-2}	2×10^{-5}
IM2C2	4.0×10^{-2}	Thyroid	1.1×10^{-2}	1.1×10^{-2}	2×10^{-5}
IM4C2	3.1×10^{-2}	Thyroid	5.6×10^{-2}	5.6×10^{-2}	1×10^{-5}
IM6D1	7.8×10^{-2}	S. int.	9.2×10^{-2}	8.5×10^{-2}	3×10^{-5}

^aLocation of maximally exposed colocated worker: 2,900 m south for accidents at fuel fabrication and tritium-target processing facilities; 2,380 m west-northwest for accidents at fuel processing and plutonium-target processing facilities.

^bSee Table I.24 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

^cS. int. = small intestine.

TABLE I.45 Estimated Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Off-Site Individual from Nonsevere Accidents at SRS Support Facilities^a

Technology/ Accident ^b	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^c	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR					
SH1C1	3.7×10^{-1}	Thyroid	7.4	7.4	2×10^{-4}
SH2N1	7.1×10^{-1}	Thyroid	2.3×10^1	2.3×10^1	3×10^{-4}
SH4N1	2.3×10^{-1}	Thyroid	7.5	7.5	9×10^{-5}
SH6D1	1.2	S. int.	1.4	1.3	5×10^{-4}
LWR					
SL1C1	3.7×10^{-1}	Thyroid	7.4	7.4	2×10^{-4}
SL2N2	2.5×10^{-1}	Thyroid	8.0	8.0	1×10^{-4}
SL6D1	1.2	S. int.	1.4	1.3	5×10^{-4}
MHTGR					
SM1C1	3.7×10^{-1}	Thyroid	7.4	7.4	2×10^{-4}
SM2F1	4.2×10^{-2}	Thyroid	1.3	1.3	2×10^{-5}
SM4F1	2.1×10^{-2}	Thyroid	6.7×10^{-1}	6.7×10^{-1}	8×10^{-6}
SM6D1	1.2	S. int.	1.4	1.3	5×10^{-4}

^aLocation of maximally exposed off-site individual: 1,300 m north-northwest for fuel fabrication accidents; 9,390 m west-southwest for plutonium-target processing accidents; and 11,540 m north-northwest for fuel processing and tritium-target processing accidents.

^bSee Table I.25 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

^cS. int. = small intestine.

TABLE I.46 Estimated Doses by Pathway to the Maximally Exposed Off-site Individual from Nonsevere Accidents at SRS Support Facilities^a

Technology/ Accident ^b	CEDE (rem)		EDE (rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR				
SH1C1	1.7×10^{-2}	2.2×10^{-1}	7.9×10^{-2}	5.0×10^{-2}
SH2N1	5.6×10^{-3}	7.0×10^{-1}	4.5×10^{-3}	5.3×10^{-3}
SH4N1	1.8×10^{-3}	2.3×10^{-1}	7.2×10^{-4}	1.7×10^{-3}
SH6D1	2.2×10^{-2}	1.2	0	0
LWR				
SL1C1	1.7×10^{-2}	2.2×10^{-1}	7.9×10^{-2}	5.0×10^{-2}
SL2N2	1.9×10^{-3}	2.4×10^{-1}	1.2×10^{-5}	1.8×10^{-3}
SL6D1	2.2×10^{-2}	1.2	0	0
MHTGR				
SM1C1	1.7×10^{-2}	2.2×10^{-1}	7.9×10^{-2}	5.0×10^{-2}
SM2F1	3.1×10^{-4}	4.1×10^{-2}	1.2×10^{-5}	3.1×10^{-4}
SM4F1	1.6×10^{-4}	2.0×10^{-2}	6.1×10^{-6}	1.5×10^{-4}
SM6D1	2.2×10^{-2}	1.2	0	0

^aLocation of maximally exposed off-site individual: 1,300 m north-northwest for fuel fabrication accidents; 9,390 m west-southwest for plutonium-target processing accidents; and 11,540 m north-northwest for fuel processing and tritium-target processing accidents.

^bSee Table I.25 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

TABLE I.47 Estimated Population Doses and Health Effects from Nonsevere Accidents at SRS Support Facilities^a

Technology/ Accident ^b	Collective Doses (person-rem)			Health Effects	
	CEDE	Gonads CDE	Thyroid CDE	Cancer Deaths	Genetic Effects
HWR					
SH1C1	1.5×10^2	1.8	3.0×10^3	6×10^{-2}	2×10^{-2}
SH2N1	5.0×10^3	2.5×10^1	1.6×10^5	2	4×10^{-2}
SH4N1	2.0×10^3	9.8	6.4×10^4	8×10^{-1}	2×10^{-2}
SH6D1	1.6×10^4	1.7×10^4	1.7×10^4	6	4
LWR					
SL1C1	1.5×10^2	1.8	3.0×10^3	6×10^{-2}	2×10^{-2}
SL2N2	1.7×10^3	8.7	5.5×10^4	7×10^{-1}	1×10^{-2}
SL6D1	1.6×10^4	1.7×10^4	1.7×10^4	6	4
MHTGR					
SM1C1	1.5×10^2	1.8	3.0×10^3	6×10^{-2}	2×10^{-2}
SM2F1	3.0×10^2	1.5	9.4×10^3	1×10^{-1}	2×10^{-2}
SM4F1	1.7×10^2	8.7×10^{-1}	5.5×10^3	7×10^{-2}	1×10^{-3}
SM6D1	1.6×10^4	1.7×10^4	1.7×10^4	6	4

^aWorst-sector population: 254,000 persons west-northwest.

^bSee Table I.25 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

TABLE I.48 Estimated Doses by Pathway to the Worst-Sector Population from Nonsevere Accidents at SRS Support Facilities^a

Technology/ Accident ^b	CEDE (person-rem)		EDE (person-rem)	
	Inhalation	Ingestion	Air Immersion	Ground Surface
HWR				
SH1C1	1.7×10^1	8.0×10^1	2.3×10^1	2.6×10^1
SH2N1	1.5×10^2	4.7×10^3	1.7×10^1	1.1×10^2
SH4N1	6.2×10^1	1.8×10^3	2.9×10^1	4.3×10^1
SH6D1	6.4×10^2	1.4×10^4	0	0
LWR				
SL1C1	1.7×10^1	8.0×10^1	2.3×10^1	2.6×10^1
SL2N2	5.6×10^1	1.6×10^3	4.2×10^{-1}	3.8×10^1
SL6D1	6.4×10^2	1.4×10^4	0	0
MHTGR				
SM1C1	1.7×10^1	8.0×10^1	2.3×10^1	2.6×10^1
SM2F1	9.3	2.8×10^2	4.6×10^{-1}	6.2
SM4F1	5.6	1.6×10^2	2.5×10^{-1}	3.8
SM6D1	6.4×10^2	1.4×10^4	0	0

^aWorst-sector population: 254,000 persons west-northwest.^bSee Table I.25 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

TABLE I.49 Estimated Maximum Doses and Lifetime Fatal Cancer Risk to the Maximally Exposed Colocated Worker from Nonsevere Accidents at SRS Support Facilities^a

Technology/ Accident ^b	Committed Effective Dose Equivalent (rem)	Maximally Exposed Organ ^c	Maximally Exposed Organ CDE (rem)	Thyroid CDE (rem)	Lifetime Fatal Cancer Risk
HWR					
SH1C1	1.8×10^{-1}	Thyroid	4.1×10^{-1}	4.1×10^{-1}	7×10^{-5}
SH2N1	2.5×10^{-2}	Thyroid	6.7×10^{-1}	6.7×10^{-1}	1×10^{-5}
SH4N1	9.1×10^{-3}	Thyroid	2.1×10^{-1}	2.1×10^{-1}	4×10^{-6}
SH6D1	8.6×10^{-2}	S. int.	1.0×10^{-1}	9.3×10^{-2}	3×10^{-5}
LWR					
SL1C1	1.8×10^{-1}	Thyroid	4.1×10^{-1}	4.1×10^{-1}	7×10^{-5}
SL2F1	8.4×10^{-4}	Lung	6.8×10^{-3}	8.9×10^{-6}	3×10^{-7}
SL6D1	8.6×10^{-2}	S. int.	1.0×10^{-1}	9.3×10^{-2}	3×10^{-5}
MHTGR					
SM1C1	1.8×10^{-1}	Thyroid	4.1×10^{-1}	4.1×10^{-1}	7×10^{-5}
SM2F1	1.3×10^{-3}	Thyroid	3.9×10^{-2}	3.9×10^{-2}	5×10^{-7}
SM4F1	5.8×10^{-4}	Thyroid	1.8×10^{-2}	1.8×10^{-2}	2×10^{-7}
SM6D1	8.6×10^{-2}	S. int.	1.0×10^{-1}	9.3×10^{-2}	3×10^{-5}

^aLocation of maximally exposed colocated worker: 540 m north for fuel fabrication accidents; 1,070 m north for fuel processing and tritium-target processing accidents; and 2,680 m east for plutonium-target processing accidents.

^bSee Table I.25 for accident descriptions. Accidents are identified by the third character, as follows: fuel fabrication, 1; fuel processing, 2; plutonium-target processing, 4; and tritium-target processing, 6.

^cS. int. = small intestine.

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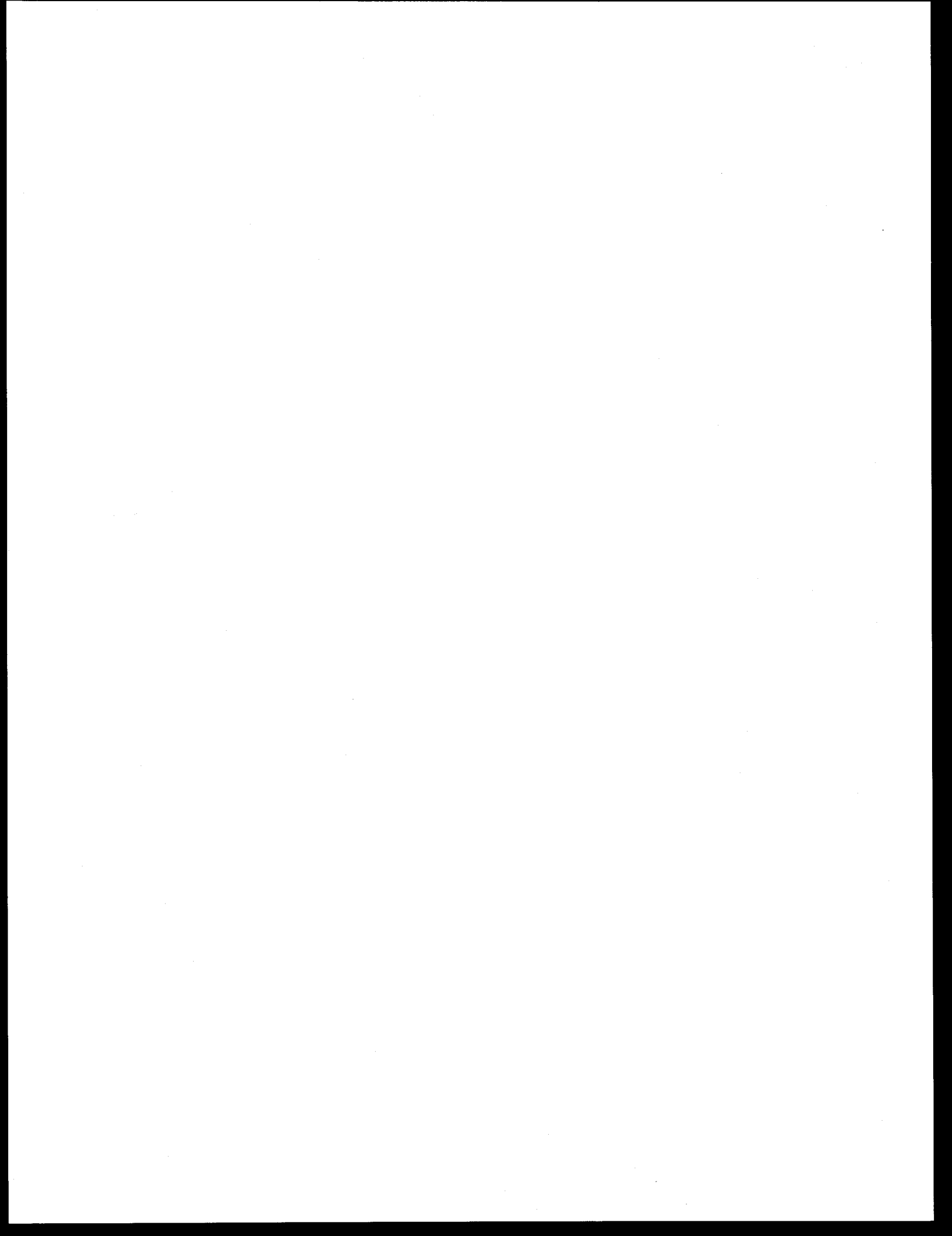
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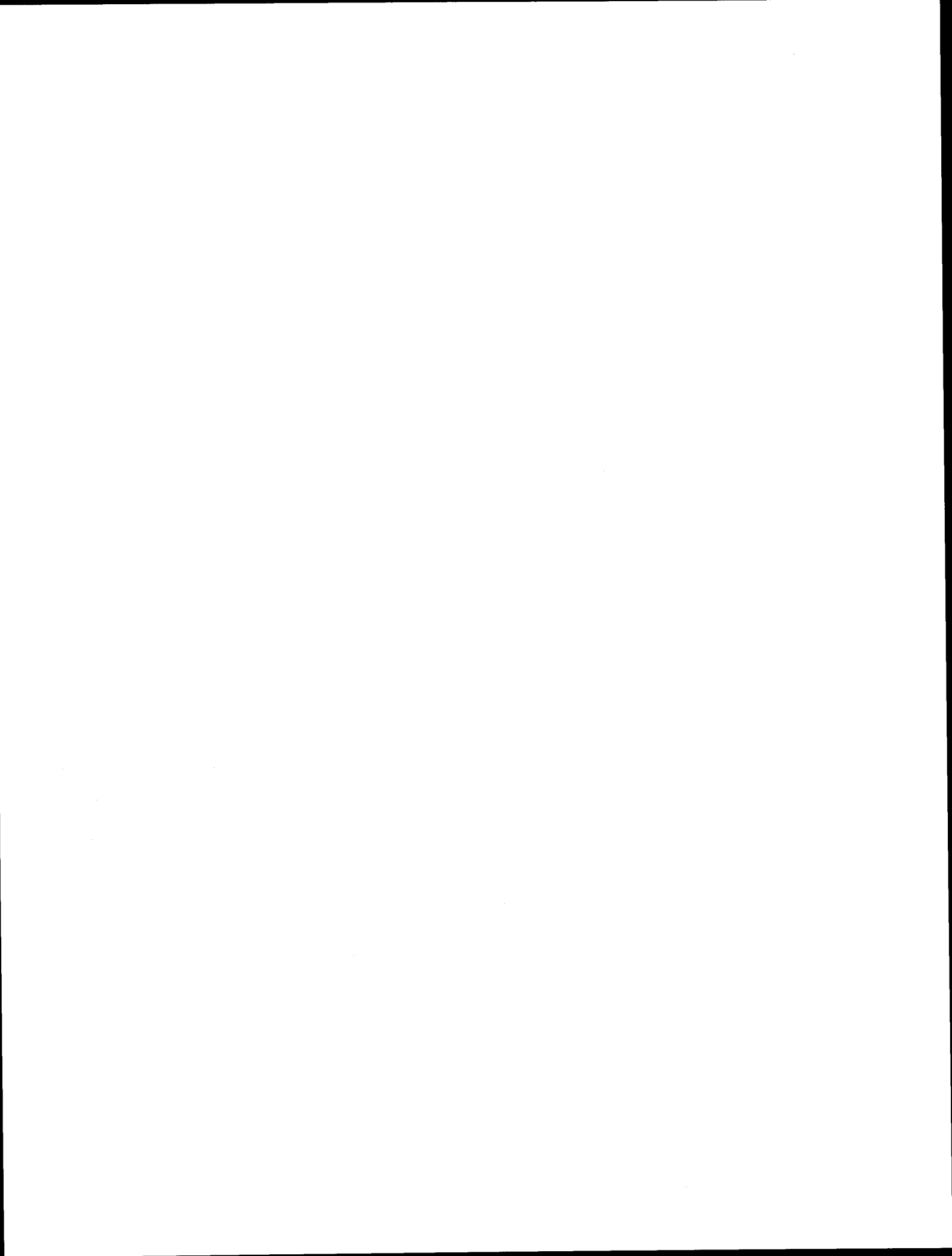
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APPENDIX J:
RADIOLOGICAL IMPACTS OF POSTULATED SEVERE ACCIDENTS



APPENDIX J:

RADIOLOGICAL IMPACTS OF POSTULATED SEVERE ACCIDENTS

J.1 IMPACTS OF SEVERE ACCIDENTS AT REACTORS

Severe reactor accidents are less likely to occur than nonsevere reactor accidents; however, the consequences of severe accidents could be more serious if mitigative actions are not taken. Severe reactor accidents generally involve some degree of physical deterioration of the fuel in the reactor core and degradation of the reactor containment structure's capability to limit the release of radioactive materials to the environment. The frequency of occurrence of any NPR severe accident that would result in a significant release of core radioactive material to the environment is typically on the order of 1×10^{-6} (one chance in one million) or lower per 1 year of reactor operations. The frequency is this low because of the defense-in-depth approach used in the design of (1) the reactor and (2) the radioactive material retention barriers. Radioactive material retention barriers include the fuel and target boundary, the reactor primary system boundary, and the reactor containment boundary. Sections J.1.1 through J.1.4 provide accident descriptions and radioactive source terms, methods of analysis, site-specific data used for calculating potential radiological impacts, and a summary of estimated risks resulting from postulated severe accidents involving reactors.

J.1.1 Accident Descriptions and Source Terms

Probabilistic risk assessment techniques were utilized to analyze NPR severe accidents and to derive the associated overall frequency of occurrence and the magnitude of any radionuclide release. Many accident sequences were considered. Accident sequences having similar fuel damage progression, reactor containment failure characteristics, and atmospheric release parameters were grouped into release categories. The frequency of a release category was determined by considering the probabilities of (1) the initiating event, (2) the failure of plant systems, and (3) the failure of reactor containment systems. The initiating events were categorized as either internal or external to the plant system. For each release category, these release frequencies and atmospheric release parameters constitute important source term information required for the impact and risk assessment.

J.1.1.1 Heavy-Water Reactor

The HWR severe accident analysis (WSRC 1991) quantified source terms (release magnitudes and frequencies) for release categories that would contribute significantly to total risk due to the operation of an HWR. Accidents considered include those that could result from system failures of plant safety features (internal events) or from seismic activity well beyond the design-basis earthquake (external events). Internally initiated events include events such as a loss-of-coolant or loss-of-heat-sink accident coupled with a station blackout or anticipated transient-without-scam accident. Externally initiated events could be caused by flood, wind, or earthquake; an earthquake has been determined to be the dominant external event initiator.

The uranium-aluminum fuel expected to be used in the HWR melts at approximately 670°C. Because of the low melting point, a complete core meltdown of fuel and targets was conservatively assumed in development of the release source terms. Table J.1 lists the core inventory of radionuclides in the HWR core that have the potential to be released to the environment. Release of radionuclides from the primary coolant system to the reactor containment, and subsequently from the reactor containment to the environment, depend upon numerous system responses, decontamination factors, and system failures.

The release categories defined in the HWR severe accident analysis (WSRC 1991) are described in Table J.2. Source term information derived from the accident analysis is presented in Table J.3. Additional source-term parameters such as release time, release duration, release height, and energy released are provided in a separate report (Hong et al. 1991).

J.1.1.2 Light-Water Reactor

The LWR severe accident analysis (WHC 1990) quantified source terms for all release categories that would contribute significantly to total risk due to the operation of an LWR. Accidents considered include those that could result from plant system failures (internal events) or from seismically induced failures (external events). Internally initiated events include those that could lead to fuel damage, i.e., loss-of-coolant and transient events. Loss-of-coolant accidents involve the rupture or failure of the primary system pressure boundary, which includes the reactor coolant system piping, valves, pressure vessel, and interconnecting systems. Such accidents also encompass loss of primary coolant inventory inside and outside the reactor containment structure. Transient events include turbine trip, loss of main feedwater, loss of off-site power, anticipated transients without scram, loss of vital AC or DC power, and loss of ultimate heat sink. Externally initiated events could be caused by flood, wind, or earthquake; an earthquake has been determined to be the dominant external cause of risk. Table J.4 lists the core inventory of important radionuclides in the LWR core that have the potential to be released to the environment. Release of radionuclides to the atmosphere would depend on the degree of core debris cooling, direct reactor containment heating, energetic reaction in the reactor containment, reactor containment bypass, reactor containment isolation response, and reactor containment integrity.

The release categories defined in the LWR severe accident analysis (WHC 1990) are described in Table J.5. Source term information derived from the accident analysis is presented in Table J.6. Additional source-term parameters such as release time, release duration, release height, and energy released are provided in a separate report (Hong et al. 1991).

J.1.1.3 Modular High-Temperature Gas-Cooled Reactor

The MHTGR severe accident analysis (EG&G 1989, 1990a-b) quantified source terms for all release categories that would contribute significantly to total risk due to the operation of an MHTGR. Accidents considered include those that could result from plant system failures (internal events) and seismically induced failures (external events). Internally initiated events include events that could result in the release of a significant amount of core radionuclides, i.e., depressurized reactor cooldown with loss of heat transport systems in the reactor and reactor cavity, air ingress, and moisture ingress. These accident scenarios are also coupled with reactor containment integrity in various conditions, ranging from as-designed reactor specifications to

TABLE J.1 Core Inventory of Important Radionuclides for an HWR
Operating at 3125 MWt in the Tritium Production Mode

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Tritium	7.0×10^7	Iodine-131	7.3×10^7
Cobalt-58	2.2×10^8	Iodine-132	1.1×10^8
Cobalt-60	2.6×10^8	Iodine-133	1.7×10^8
Krypton-85	2.9×10^5	Iodine-134	2.0×10^8
Krypton-85m	3.2×10^7	Iodine-135	1.6×10^8
Krypton-87	6.3×10^7	Xenon-133	1.7×10^8
Krypton-88	8.7×10^7	Xenon-135	1.4×10^7
Rubidium-86	8.3×10^4	Cesium-134	2.5×10^6
Strontium-89	1.2×10^8	Cesium-136	8.3×10^5
Strontium-90	2.4×10^6	Cesium-137	2.3×10^6
Strontium-91	1.5×10^8	Barium-139	1.8×10^8
Strontium-92	1.5×10^8	Barium-140	1.6×10^8
Yttrium-90	2.5×10^6	Lanthanum-140	1.6×10^8
Yttrium-91	1.4×10^8	Lanthanum-141	1.5×10^8
Yttrium-92	1.5×10^8	Lanthanum-142	1.5×10^8
Yttrium-93	1.6×10^8	Cerium-141	1.5×10^8
Zirconium-95	1.5×10^8	Cerium-143	1.5×10^8
Zirconium-97	1.5×10^8	Cerium-144	6.2×10^7
Niobium-95	1.4×10^8	Praseodymium-143	1.5×10^8
Molybdenum-99	1.5×10^8	Neodymium-147	5.7×10^7
Technetium-99m	1.3×10^8	Neptunium-239	4.8×10^7
Ruthenium-103	8.1×10^7	Plutonium-238	5.5×10^4
Ruthenium-105	2.8×10^7	Plutonium-239	3.6×10^2
Ruthenium-106	4.0×10^6	Plutonium-240	3.1×10^2
Rhodium-105	2.4×10^7	Plutonium-241	1.0×10^5
Antimony-127	2.9×10^6	Americium-241	2.4×10^1
Antimony-129	1.6×10^7	Curium-242	6.2×10^3
Tellurium-127	2.8×10^6	Curium-244	9.7×10^{-1}
Tellurium-127m	3.6×10^5		
Tellurium-129	1.6×10^7		
Tellurium-129m	4.3×10^6		
Tellurium-131m	9.2×10^6		
Tellurium-132	1.1×10^8		

Source: WSRC 1991.

TABLE J.2 Description of Release Categories for the HWR Severe Accident Analysis

Release Category	Description for HWR
1	Early reactor containment failure and early failure of containment sprays. Accumulators/borated water storage tanks fail to dump.
2	Early reactor containment failure and early failure of reactor containment sprays. Accumulators/borated water storage tanks succeed in dumping.
3	Reactor containment isolation failure with containment sprays failed.
4	Reactor containment isolation failure with containment sprays functioning.
5	Late reactor containment failure with containment sprays failed.
6	No reactor containment failure but containment sprays fail.
7	Core melt but no reactor containment failure or containment spray failure.
8	Seismically induced core melt causing early reactor containment failure with containment sprays failed.
9	Seismically induced core melt causing late reactor containment failure with containment sprays failed.
10	Seismically induced core melt causing no reactor containment failure but containment sprays fail.

TABLE J.3 Release Specifications Used in HWR Impact Assessments

Release Category ^a	Release Frequency (per reactor-year)	Fraction of Core Inventory Released ^b				
		Noble Gases	I-Cs-Rb	Te	Sr-Ba	Others ^c
1	1×10^{-8}	1	1×10^{-1}	5×10^{-2}	6×10^{-2}	3×10^{-5}
2	1×10^{-7}	1	3×10^{-2}	1×10^{-2}	1×10^{-2}	5×10^{-6}
3	5×10^{-8}	1	3×10^{-2}	1×10^{-2}	1×10^{-2}	5×10^{-6}
4	5×10^{-8}	1	1×10^{-3}	4×10^{-4}	5×10^{-4}	2×10^{-7}
5	5×10^{-8}	1	5×10^{-5}	2×10^{-5}	1×10^{-5}	1×10^{-8}
6	2×10^{-6}	3×10^{-2}	5×10^{-6}	2×10^{-6}	3×10^{-6}	1×10^{-9}
7	5×10^{-6}	3×10^{-2}	3×10^{-8}	1×10^{-8}	1×10^{-8}	5×10^{-12}
8	1×10^{-7}	1	3×10^{-2}	1×10^{-2}	1×10^{-2}	5×10^{-6}
9	5×10^{-8}	1	5×10^{-5}	2×10^{-5}	1×10^{-5}	1×10^{-8}
10	2×10^{-6}	3×10^{-2}	5×10^{-6}	2×10^{-6}	3×10^{-6}	1×10^{-9}

^aSee Table J.2 for descriptions of HWR release categories.

^bThe release fraction for tritium is 1 for all categories.

^cIncludes: Co, Ru, Y, Zr, Nb, Mo, Tc, Rh, Sb, La, Ce, Pr, Nd, Np, Pu, Am, and Cm.

Source: WSRC 1991.

various postulated failures that would increase the leakage of radionuclides to the environment. Externally initiated events could be caused by flood, wind, or earthquake; an earthquake has been determined to be the dominant external cause of risk.

The types of accidents identified in the severe accident analysis would result in inadequate heat removal or chemical attack, such that the fuel and target particle coatings would allow increased radionuclide release. Particle coatings are the first barrier to the release of fission products and tritium from an MHTGR to the environment. The magnitude of releases from fuel and target particles was determined to be an exponential function of post-accident temperatures. Therefore, releases at high post-accident temperatures would be expected to be significantly greater than releases occurring at or near normal core operating temperatures. For the purpose of assessment, the amount of release was determined on the basis of a conservative assumption that the core cooling mode is by heat transfer through the reactor core and reactor vessel to the building surrounding the reactor. This mode of heat removal, designated as "conduction cooldown," results in achieving the highest core temperatures during transient events. If the particle coatings are breached, the second barrier to radionuclide release is the primary coolant system pressure boundary. The release of radionuclides from this barrier would occur if a break developed in the primary coolant system. Thereafter, releases to the reactor containment would leak to the environment at a rate dictated by the integrity of the reactor containment boundary. Reactor containment leakage is conservatively assumed to be independent of reactor containment pressure and is assumed to be equivalent to the maximum potential value.

TABLE J.4 Core Inventory of Important Radionuclides for an LWR Operating at 3800 MWt in the Tritium Production Mode

Radionuclide	Inventory (Ci)	Radionuclide	Inventory (Ci)
Tritium	8.5×10^7	Tellurium-127	6.2×10^6
Krypton-85	4.0×10^5	Tellurium-129	2.4×10^7
Krypton-85m	3.8×10^7	Tellurium-131m	1.2×10^7
Krypton-87	7.6×10^7	Iodine-131	9.3×10^7
Krypton-88	1.1×10^8	Iodine-132	1.4×10^8
Rubidium-86	5.1×10^4	Iodine-133	2.1×10^8
Strontium-89	1.4×10^8	Iodine-134	2.4×10^8
Strontium-90	3.1×10^6	Iodine-135	2.0×10^8
Strontium-91	1.7×10^8	Xenon-133	2.0×10^8
Strontium-92	1.8×10^8	Xenon-135	5.2×10^7
Yttrium-90	3.2×10^6	Cesium-134	1.5×10^6
Yttrium-91	1.7×10^8	Cesium-136	1.3×10^6
Yttrium-92	1.8×10^8	Cesium-137	3.3×10^6
Zirconium-95	1.9×10^8	Barium-140	1.9×10^8
Zirconium-97	1.8×10^8	Lanthanum-140	2.1×10^8
Niobium-95	1.8×10^8	Cerium-144	8.2×10^7
Molybdenum-99	1.9×10^8	Neodymium-147	7.1×10^7
Technetium-99m	1.7×10^8	Neptunium-239	9.8×10^8
Ruthenium-103	1.1×10^8	Plutonium-238	9.8×10^3
Ruthenium-105	4.6×10^7	Plutonium-239	1.3×10^4
Ruthenium-106	8.3×10^6	Plutonium-240	6.5×10^3
Rhodium-105	4.3×10^7	Plutonium-241	1.3×10^6
Antimony-127	6.4×10^6	Americium-241	4.0×10^2
Antimony-129	2.4×10^7		

Source: WHC 1990.

The radionuclide release associated with MHTGR severe accidents is assumed to occur in two phases. The first phase would occur at the initiation of the accident when the primary system is breached and would consist of releasing the fission products in the primary coolant (circulating activity) and those being plated out on system surfaces (plateout activity). The second phase of the release would occur between 50 and 80 hours later in the accident when the core heated up and achieved a maximum temperature. The release during the second phase would be significantly larger than that during the first phase.

Source term information derived from the accident analysis is presented in Tables J.7 and J.8 (EG&G 1990a-b). Additional source-term parameters such as release time, release duration, release height, and energy released are provided in a separate report (Hong et al. 1991).

J.1.2 Methods

A schematic of the impact assessment approach is presented in Figure J.1. The impacts were calculated using the MELCOR Accident Consequence Code System (MACCS), Version 1.5

TABLE J.5 Release Categories for the LWR Severe Accident Analysis

Release Category	Description for LWR
RC1A	Reactor containment bypass via an interfacing system's loss-of-coolant accident (LOCA) resulting in rupture of a low-pressure decay heat removal system pipe in the General Services Building (GSB) with the ventilation system fire dampers successfully closed.
RC2	Reactor containment bypass via an interfacing system's LOCA resulting in failure of a low-pressure decay heat removal system seal, thus flooding the decay heat removal system sump in the GSB with the ventilation system fire dampers failed open.
RC2A	Reactor containment bypass via an interfacing system's LOCA resulting in failure of a low-pressure decay heat removal system seal, thus flooding the decay heat removal system sump in the GSB with the ventilation system fire dampers successfully closed.
RC3	Reactor containment isolation failure at high energy with containment sprays failed.
RC3A	Reactor containment isolation failure at high energy with containment sprays functioning.
RC4	Reactor containment isolation failure at low energy with containment sprays failed.
RC4A	Reactor containment isolation failure at low energy with containment sprays functioning.
RC5	Early reactor containment failure with containment sprays failed.
RC5A	Early reactor containment failure with containment sprays functioning.
RC6	Late reactor containment failure with containment sprays failed.
RC8	Reactor containment intact.

TABLE J.6 Release Specifications Used in LWR Impact Assessments

Release Category ^a	Release Frequency (per reactor-year)	Fraction of Core Inventory Released ^b									
		Xe-Kr	I-Cs-Rb	Mo	Sr	Ba	Ce	La ^c	Te	Sb	
RC1A	8 × 10 ⁻⁹	1	8 × 10 ⁻²	4 × 10 ⁻³	2 × 10 ⁻⁴	2 × 10 ⁻³	1 × 10 ⁻⁵	1 × 10 ⁻⁶	2 × 10 ⁻²	2 × 10 ⁻²	
RC2	8 × 10 ⁻⁹	1	2 × 10 ⁻³	2 × 10 ⁻⁴	4 × 10 ⁻⁶	2 × 10 ⁻⁵	2 × 10 ⁻⁷	2 × 10 ⁻⁸	5 × 10 ⁻⁴	3 × 10 ⁻⁴	
RC2A	8 × 10 ⁻⁷	1	4 × 10 ⁻⁴	3 × 10 ⁻⁵	8 × 10 ⁻⁷	4 × 10 ⁻⁶	4 × 10 ⁻⁸	4 × 10 ⁻⁹	1 × 10 ⁻⁴	3 × 10 ⁻⁵	
RC3	2 × 10 ⁻⁸	1	7 × 10 ⁻²	4 × 10 ⁻³	1 × 10 ⁻²	5 × 10 ⁻³	7 × 10 ⁻⁴	6 × 10 ⁻⁵	1 × 10 ⁻¹	2 × 10 ⁻²	
RC3A	2 × 10 ⁻⁷	1	4 × 10 ⁻⁴	3 × 10 ⁻⁵	8 × 10 ⁻⁷	4 × 10 ⁻⁶	4 × 10 ⁻⁸	4 × 10 ⁻⁹	8 × 10 ⁻⁴	2 × 10 ⁻⁴	
RC4	2 × 10 ⁻⁹	1	5 × 10 ⁻²	2 × 10 ⁻³	3 × 10 ⁻²	2 × 10 ⁻²	3 × 10 ⁻³	3 × 10 ⁻⁴	7 × 10 ⁻²	4 × 10 ⁻²	
RC4A	8 × 10 ⁻⁹	1	8 × 10 ⁻⁴	4 × 10 ⁻⁵	2 × 10 ⁻⁶	2 × 10 ⁻⁵	1 × 10 ⁻⁷	1 × 10 ⁻⁸	1 × 10 ⁻⁴	2 × 10 ⁻⁴	
RC5	2 × 10 ⁻⁸	1	3 × 10 ⁻³	2 × 10 ⁻⁴	6 × 10 ⁻⁴	2 × 10 ⁻⁴	3 × 10 ⁻⁵	3 × 10 ⁻⁶	6 × 10 ⁻³	1 × 10 ⁻³	
RC5A	1 × 10 ⁻⁸	1	3 × 10 ⁻⁴	3 × 10 ⁻⁵	7 × 10 ⁻⁷	3 × 10 ⁻⁶	3 × 10 ⁻⁸	3 × 10 ⁻⁹	1 × 10 ⁻⁴	6 × 10 ⁻⁵	
RC6	6 × 10 ⁻⁷	7 × 10 ⁻¹	1 × 10 ⁻³	3 × 10 ⁻⁶	8 × 10 ⁻⁵	1 × 10 ⁻⁴	9 × 10 ⁻⁶	8 × 10 ⁻⁷	7 × 10 ⁻³	9 × 10 ⁻³	
RC8	2 × 10 ⁻⁵	1 × 10 ⁻³	3 × 10 ⁻⁶	2 × 10 ⁻⁷	7 × 10 ⁻⁹	6 × 10 ⁻⁸	6 × 10 ⁻¹⁰	5 × 10 ⁻¹¹	8 × 10 ⁻⁷	9 × 10 ⁻⁷	
RC3-E	6 × 10 ⁻⁹	1	7 × 10 ⁻²	4 × 10 ⁻³	1 × 10 ⁻²	5 × 10 ⁻³	7 × 10 ⁻⁴	6 × 10 ⁻⁵	1 × 10 ⁻¹	2 × 10 ⁻²	
RC3A-E	2 × 10 ⁻⁹	1	4 × 10 ⁻⁴	3 × 10 ⁻⁵	8 × 10 ⁻⁷	4 × 10 ⁻⁶	4 × 10 ⁻⁸	4 × 10 ⁻⁹	8 × 10 ⁻⁴	2 × 10 ⁻⁴	
RC4-E	3 × 10 ⁻⁹	1	5 × 10 ⁻²	2 × 10 ⁻³	3 × 10 ⁻²	2 × 10 ⁻²	3 × 10 ⁻³	3 × 10 ⁻⁴	7 × 10 ⁻²	4 × 10 ⁻²	
RC5-E	5 × 10 ⁻⁹	1	3 × 10 ⁻³	2 × 10 ⁻⁴	6 × 10 ⁻⁴	2 × 10 ⁻⁴	3 × 10 ⁻⁵	3 × 10 ⁻⁶	6 × 10 ⁻³	1 × 10 ⁻³	
RC6-E	2 × 10 ⁻⁷	7 × 10 ⁻¹	1 × 10 ⁻³	3 × 10 ⁻⁶	8 × 10 ⁻⁵	1 × 10 ⁻⁴	9 × 10 ⁻⁶	8 × 10 ⁻⁷	7 × 10 ⁻³	9 × 10 ⁻³	
RC8-E	9 × 10 ⁻⁷	1 × 10 ⁻³	3 × 10 ⁻⁶	2 × 10 ⁻⁷	7 × 10 ⁻⁹	6 × 10 ⁻⁸	6 × 10 ⁻¹⁰	5 × 10 ⁻¹¹	8 × 10 ⁻⁷	9 × 10 ⁻⁷	

^aSee Table J.5 for descriptions of LWR release categories; the designation "-E" added to a release category means that the accident is earthquake induced.

^bRelease fraction for tritium is 1 for all release categories.

^cIncludes La, Y, Zr, Nb, Tc, Ru, Rh, Nd, Np, Pu, and Am.

Source: WHC 1990.

TABLE J.7 Release Specifications Used in MHTGR Impact Assessments

Release Category		Release Activity ^a		Form of Tritium Release ^b	Frequency of Occurrence ^c (per reactor-year)
No.	Description	Off-site	On-site		
1	Depressurized conduction cooldown with reactor cavity cooling system functioning (leakage 100% per day)	A	G	HT (gaseous)	6×10^{-6}
2	Depressurized conduction cooldown without reactor cavity cooling system functioning (leakage 1% per day)	B	F	HT (gaseous)	6×10^{-6}
3	Air ingress (leakage 100% per day)	A	G	HT (gaseous)	2×10^{-6}
4	Moisture ingress (leakage 1% per day)	B	F	HTO (oxide)	2×10^{-6}
5	Double-ended break considering air ingress (leakage changes from 100% to 1% per day)	D	G	HT (gaseous)	8×10^{-8}
6	Double-ended break considering air ingress (leakage changes from 1% to 10% per day)	C	F	HT (gaseous)	8×10^{-8}
7	Depressurized conduction cooldown without reactor cavity cooling system functioning (leakage changes from 100% to 1% per day)	D	G	HT (gaseous)	6×10^{-8}
8	Depressurized conduction cooldown without reactor cavity cooling system functioning (leakage changes from 1% to 10% per day)	C	F	HT (gaseous)	6×10^{-8}
9	Moisture ingress (leakage changes from 100% to 1% per day)	D	G	HTO (oxide)	2×10^{-8}
10	Double-ended break considering air ingress (leakage 100% per day)	E	G	HT (gaseous)	8×10^{-9}
11	Depressurized conduction cooldown without reactor cavity cooling system functioning (leakage 100% per day)	E	G	HT (gaseous)	6×10^{-9}
12	Depressurized conduction cooldown with reactor cavity cooling system functioning -- earthquake-induced accident (leakage 100% per day)	A	G	HT (gaseous)	1×10^{-9}

^aRelease activity categories A through G are given in Table J.8.

^bHT = elemental tritium; HTO = tritiated water. It is assumed that 10% of the elemental tritium released in gaseous form is converted to tritiated water (oxide form).

^cCombined frequency of an accident at any one of the eight modules on the site.

Sources: EG&G 1990a-b; Bowman 1990.

TABLE J.8 Activity of Important Radionuclides Released to the Environment from a Severe Accident for an MHTGR Operating in the Tritium Production Mode

Radionuclide	Release Activity ^a (Ci)						
	A	B	C	D	E	F	G
Tritium	7.3×10^5	2.8×10^6	2.8×10^6	2.8×10^6	2.8×10^6	2.3×10^1	2.3×10^1
Krypton-85	1.4×10^{-2}	1.4×10^{-2}	1.4×10^{-2}	1.4×10^{-2}	1.4×10^{-2}	1.4×10^{-2}	1.4×10^{-2}
Krypton-85m	1.3	1.7×10^{-2}	1.7×10^{-2}	1.3	1.3	1.7×10^{-2}	1.3
Krypton-87	8.9×10^{-1}	9.6×10^{-3}	9.6×10^{-3}	8.9×10^{-1}	8.9×10^{-1}	9.6×10^{-3}	8.9×10^{-1}
Krypton-88	2.7	3.1×10^{-2}	3.1×10^{-2}	2.7	2.7	3.1×10^{-2}	2.7
Rubidium-86	1.6	2.8×10^{-2}	2.8×10^{-2}	1.6	1.6	2.8×10^{-2}	1.6
Strontium-89	1.2×10^3	5.2×10^1	4.9×10^2	5.5×10^1	3.0×10^3	5.4×10^{-2}	3.1
Strontium-90	7.4×10^1	3.1	2.9×10^1	4.4	1.8×10^2	2.3×10^{-2}	1.3
Strontium-91	2.4	6.9×10^{-1}	6.7	6.9×10^{-1}	5.2×10^1	4.8×10^{-7}	3.6×10^{-5}
Yttrium-90	3.4×10^1	1.9	1.8×10^1	1.9	1.1×10^2	5.7×10^{-8}	3.5×10^{-6}
Yttrium-91	1.5×10^3	6.5×10^1	6.0×10^2	6.5×10^1	3.7×10^3	2.3×10^{-6}	1.3×10^{-4}
Zirconium-95	1.7×10^3	7.2×10^1	6.8×10^2	7.2×10^1	4.1×10^3	1.3×10^{-6}	7.7×10^{-5}
Zirconium-97	4.1×10^1	4.9	4.7×10^1	4.9	3.4×10^2	7.1×10^{-7}	4.9×10^{-5}
Niobium-95	1.6×10^3	7.0×10^1	6.6×10^2	7.0×10^1	4.0×10^3	2.7×10^{-6}	1.5×10^{-4}
Molybdenum-99	6.6×10^2	3.6×10^1	3.4×10^2	3.6×10^1	2.2×10^3	1.1×10^{-6}	6.7×10^{-5}
Technetium-99m	6.7×10^{-2}	5.9×10^{-1}	5.8	5.9×10^{-1}	4.7×10^1	7.4×10^{-7}	5.9×10^{-5}
Ruthenium-103	9.6×10^2	4.1×10^1	3.9×10^2	4.1×10^1	2.4×10^3	7.8×10^{-7}	4.4×10^{-5}
Ruthenium-105	6.2×10^{-4}	2.0×10^{-3}	2.0×10^{-2}	2.0×10^{-3}	1.7×10^{-1}	9.6×10^{-8}	8.0×10^{-6}
Ruthenium-106	6.9×10^1	3.0	2.8×10^1	3.0	1.7×10^2	7.1×10^{-8}	4.1×10^{-6}
Rhodium-105	7.0×10^1	4.8	4.5×10^1	4.8	3.1×10^2	5.0×10^{-7}	3.2×10^{-5}
Antimony-127	2.8×10^1	1.4	1.4×10^1	1.4	8.6×10^1	3.8×10^{-8}	2.3×10^{-6}
Antimony-129	5.3×10^{-8}	3.9×10^{-7}	3.9×10^{-6}	4.5×10^{-7}	3.9×10^{-5}	5.3×10^{-10}	5.3×10^{-8}
Tellurium-127	9.9	1.7×10^{-1}	1.7×10^{-1}	9.9	9.9	1.7×10^{-1}	9.9
Tellurium-127m	8.1×10^{-1}	1.4×10^{-2}	1.4×10^{-2}	8.1×10^{-1}	8.1×10^{-1}	1.4×10^{-2}	8.1×10^{-1}
Tellurium-129	7.6×10^{-2}	8.1×10^{-4}	8.1×10^{-4}	7.6×10^{-2}	7.6×10^{-2}	8.1×10^{-4}	7.6×10^{-2}
Tellurium-129m	2.1	3.7×10^{-2}	3.7×10^{-2}	2.1	2.1	3.7×10^{-2}	2.1
Tellurium-131m	1.2	1.9×10^{-2}	1.9×10^{-2}	1.2	1.2	1.9×10^{-2}	1.2
Tellurium-132	8.0×10^1	1.4	1.4	8.0×10^1	8.0×10^1	1.4	8.0×10^1
Iodine-131	2.8×10^1	4.8×10^{-1}	4.8×10^{-1}	2.8×10^1	2.8×10^1	4.8×10^{-1}	2.8×10^1
Iodine-132	2.1×10^1	2.3×10^{-1}	2.3×10^{-1}	2.1×10^1	2.1×10^1	2.3×10^{-1}	2.1×10^1
Iodine-133	1.5×10^1	2.2×10^{-1}	2.2×10^{-1}	1.5×10^1	1.5×10^1	2.2×10^{-1}	1.5×10^1
Iodine-134	5.1×10^{-1}	5.3×10^{-3}	5.3×10^{-3}	5.1×10^{-1}	5.1×10^{-1}	5.3×10^{-3}	5.1×10^{-1}
Iodine-135	5.0	6.4×10^{-2}	6.4×10^{-2}	5.0	5.0	6.4×10^{-2}	5.0
Xenon-133	2.5	2.0×10^{-1}	2.0×10^{-1}	2.5	2.5	2.0×10^{-1}	2.5
Xenon-135	2.1	3.3×10^{-2}	3.3×10^{-2}	2.1	2.1	3.3×10^{-2}	2.1
Cesium-134	2.5×10^2	9.1	8.5×10^1	9.1	5.2×10^2	2.3×10^{-5}	1.3×10^{-3}
Cesium-136	1.7×10^2	6.5	6.1×10^1	6.5	3.8×10^2	1.1×10^{-7}	6.3×10^{-6}
Cesium-137	1.1×10^2	4.0	3.7×10^1	4.0	2.3×10^2	1.1×10^{-4}	6.1×10^{-3}
Barium-140	1.1×10^3	4.8×10^1	4.5×10^2	4.8×10^1	2.8×10^3	9.5×10^{-7}	5.5×10^{-5}
Lanthanum-140	4.3×10^2	2.8×10^1	2.7×10^2	2.8×10^1	1.8×10^3	1.9×10^{-6}	1.2×10^{-4}
Cerium-141	1.5×10^3	6.6×10^1	6.2×10^2	6.6×10^1	3.8×10^3	1.3×10^{-6}	7.2×10^{-5}
Cerium-143	2.4×10^2	1.7×10^1	1.6×10^2	1.7×10^1	1.1×10^3	8.7×10^{-7}	5.6×10^{-5}
Cerium-144	5.4×10^2	2.4×10^1	2.3×10^2	2.4×10^1	1.4×10^3	5.1×10^{-7}	2.9×10^{-5}
Praseodymium-143	1.3×10^3	5.8×10^1	5.4×10^2	5.8×10^1	3.3×10^3	2.3×10^{-6}	1.3×10^{-4}
Neodymium-147	5.1×10^2	2.3×10^1	2.1×10^2	2.3×10^1	1.3×10^3	4.7×10^{-7}	2.7×10^{-5}

^aThe release activity magnitude corresponds to the release caused by the postulated accident at one of the modules.

Sources: EG&G 1990a-b.

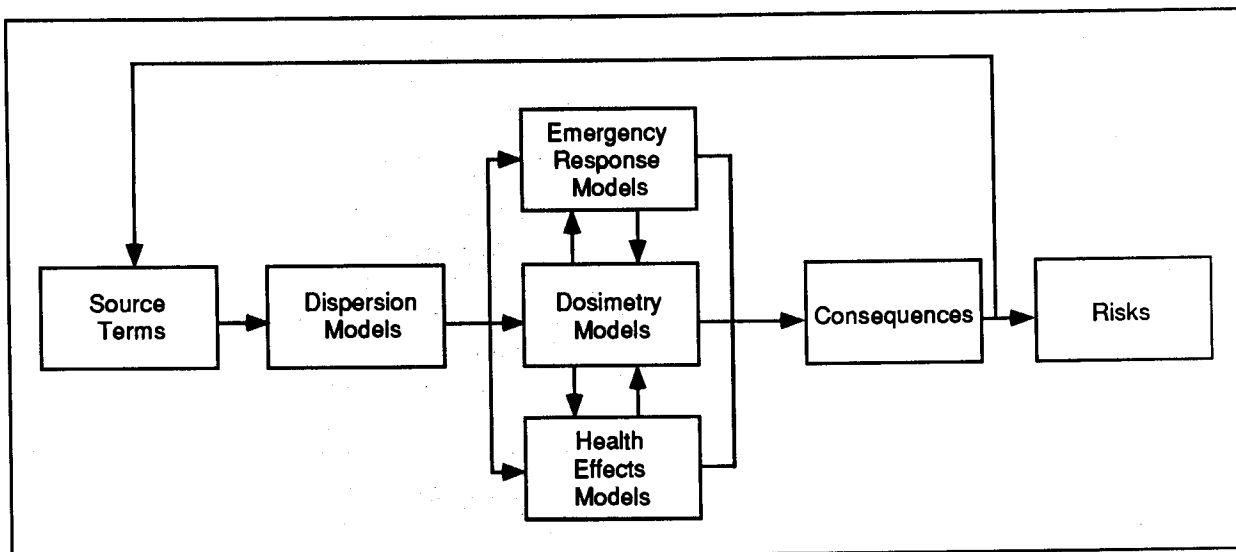


FIGURE J.1 Schematic Outline of Consequence and Risk Assessments for Postulated Severe Accidents

(Jow et al. 1990). The MACCS was developed as a successor to the Calculation of Reactor Accident Consequences (CRAC2) code (Ritchie et al. 1984), which was developed to estimate the consequences of severe reactor accidents for the NRC's reactor safety study (NRC 1975). The MACCS includes models for dispersion, emergency response, dosimetry, and health effects.

The MACCS used the source term data given in Tables J.1, J.3, J.4, J.6, J.7, and J.8 and modeled the dispersion and deposition of radionuclides released from the reactor containment to the atmosphere with a straight-line Gaussian plume. Plume rise and dry and wet deposition were taken into consideration. One year of hourly on-site meteorological data and a weather bin sampling technique were used to represent the dispersion process according to each site's characteristic weather. Downwind concentrations of radionuclides up to a distance of 800 km were calculated for each of 16 directional sectors around the reactor.

Radiation doses to an off-site population were calculated in the dosimetry models using the concentrations of radionuclides obtained from the dispersion models. Dose conversion factors were used to convert the radionuclide concentrations to organ dose equivalents and whole-body effective dose equivalents. Exposure pathways considered in the MACCS for calculating doses received during the early period following an accident were (1) direct radiation from the passing plume and from radioactive material deposited on the ground, (2) inhalation from the plume, (3) deposition on skin, and (4) inhalation of resuspended ground contamination. Long-term exposure pathways considered were (1) ground shine, (2) inhalation of resuspended ground contamination, and (3) ingestion of contaminated food and water. The on-site impact analyses considered exposures only during the early phase of an accident. Air pathway exposures would be dominant contributors to the effects of a severe reactor accident for both off-site and on-site populations. Liquid pathway exposures were also considered because radionuclides can be dispersed by a reactor's liquid effluent streams; however, the radiation doses and health risks associated with such liquid pathways would be several orders of magnitude lower than those associated with air pathways in a severe accident situation.

In calculating potential radiation doses, emergency response actions were accounted for that could take place immediately after an accident and in the long term. Short-term actions included evacuation, sheltering, and relocation. Long-term actions included interdiction of milk and crops, decontamination, and temporary interdiction and condemnation of land and buildings.

Health effects for an off-site population were estimated in the health effects models using the organ dose equivalents obtained from the dosimetry models and the results from recent studies (Evans 1990; Abrahamson et al. 1989). The health effects considered in this study were early fatalities and latent cancer fatalities. The MACCS code does not implement the BEIR V cancer fatality models. The MACCS cancer fatality estimates could be about a factor of two lower than the BEIR V predictions. This factor is not significant considering the overall uncertainty in this type of assessment.

The RISKIN code (Rollstin and Hong 1990) was developed to process the MACCS results, incorporate the frequencies of release categories, and produce complementary cumulative distribution functions (CCDFs) for each of the consequence measures -- i.e., early fatalities, latent cancer fatalities, and population doses. The CCDFs show the relationship between the frequency of a particular type of consequence being equaled or exceeded and the magnitude of the consequence.

Radiological risks can be calculated by combining the frequency (likelihood of occurrence) of accidents and their impacts (consequences). The resultant risk is expressed as a measure of consequence per unit of time (per reactor-year). Such a risk estimate can aid in comparing the radiological risks associated with accidents to the risks associated with other activities. The RISKIN code estimated the average value of risk for a given risk measure by summing the product of (1) the frequency and (2) the average value of the consequence calculated by the MACCS for the release categories. The risk measures considered in this study were societal early fatality risk, societal cancer fatality risk, societal dose risk, and two measures -- early fatalities and cancer fatalities -- for the purpose of comparison to the NPR safety goals (DOE 1989) that are used to demonstrate achievement of the NPR safety objectives.

J.1.3 Site-Specific Data

J.1.3.1 Hanford Site

Environmental parameters specific to Hanford were used for the Hanford assessment, including meteorological data, population distribution data, and agricultural data. The meteorological data consist of 1 year of hourly recordings of wind direction, wind speed, atmospheric stability, and accumulated precipitation. The meteorological data were obtained from measurements on the WNP-1 site at Hanford.

For the projected off-site population within 80 km of the Hanford Site, the data used for the MACCS calculations for severe accidents were consistent with the data used for the GENII calculations for normal and nonsevere accidents (see Appendixes H and I). The projected populations residing between 80 and 800 km were derived from U.S. and Canadian census data (NUS Corporation 1989). The current colocated worker population was used without adjustment

for the year 2020. For calculating impacts, the population distribution was assigned to a geometric grid consisting of 16 angular sectors, with the first sector centered on due north and the others proceeding in a clockwise rotation; an 800-km radius from the site was partitioned into 26 spatial intervals.

Agricultural data were derived from the 1982 U.S. Census of Agriculture and from Statistics Canada (NUS Corporation 1989). The data were compiled separately from the U.S. and Canadian data, and the U.S. and Canadian components were then summed.

The following assumptions were made regarding dose mitigative actions for both on-site and off-site dose calculations. Two sets of parameters were assumed for short-term emergency response actions. The first set assumed that the population within 16 km would be evacuated after a 30-minute delay time with a radial speed of 10 m/s (Mogle 1987) and that the population beyond 16 km, where the projected lifetime whole-body effective dose equivalent from a 7-day occupancy would exceed 50 or 25 rem, would be relocated within 12 or 24 hours, respectively, after plume arrival. It is recognized that for certain accident initiators, such as a severe seismic event, the assumed dose mitigative actions might be adversely affected. The second set of parameters, which assumed twice the delay time and half the evacuation speed, reflects a radiological emergency response situation hampered by a regional earthquake within a 16-km zone. For long-term countermeasures, protective action guidelines proposed by the Food and Drug Administration (1982) and the EPA (1990) were used with the off-site protective measures assumptions provided in NUREG-1150.

J.1.3.2 Idaho National Engineering Laboratory

Environmental parameters specific to INEL were used for the INEL analysis. The meteorological data were obtained from measurements on the INEL site. The methods used to derive the population and agricultural data were the same for INEL as for the Hanford Site (see Section J.1.3.1). The assumptions used for dose mitigative actions at INEL were also the same as those used for Hanford (Section J.1.3.1).

J.1.3.3 Savannah River Site

Environmental parameters specific to SRS were used for the SRS assessment. The meteorological data were obtained from measurements taken at a meteorological tower near the candidate NPR site at SRS. Except for the fact that no Canadian population or agricultural data were required for SRS, the methods used to derive the population and agricultural data were the same for SRS as for the Hanford Site (see Section J.1.3.1).

The assumptions used for dose mitigative actions at SRS are similar to those used for Hanford. Two sets of parameters were assumed. The first set of parameters specifies a delay time of 30 minutes and a radial speed of 3.8 m/s for on-site evacuation; for off-site evacuation, the delay time is 1 hour and the radial speed is 1.12 m/s. The second set of parameters doubles the delay time and reduces the radial speed by a factor of two for both on-site and off-site cases.

J.1.4 Result of Risk Analysis

Estimated risks of cancer fatality and population dose to the general public resulting from postulated NPR severe accidents at the three alternative sites are summarized in Table J.9. Results of various accident analyses of other nuclear plants are provided as reference.

The estimated risks at NPR facilities are lower than those calculated at the reference nuclear plants. This is generally due to a combination of factors such as fuel burnup, remoteness of DOE sites in low population areas, modern safety features, and very strong and large reactor containment.

J.2 IMPACTS OF SEVERE ACCIDENTS AT SUPPORT FACILITIES

The NPR support facilities considered in the severe accident analysis consist of the fuel fabrication facility, the tritium-target processing facility, the tritium-target fabrication facility, and the heavy water processing facility (for HWR only). These facilities are associated with the production of tritium.

A range of severe accidents, including internally initiated process-related events and externally induced events, were considered for the NPR support facilities. The inventories of radionuclides at risk, as well as the potential energy released in a severe event for the support facilities, are generally orders of magnitude smaller than those of the reactor. Thus, the radiological consequences of a severe accident involving support facilities would generally be lower than those expected from a severe reactor accident. Furthermore, severe support facility accidents with the potential to result in a larger release than nonsevere accidents would have to involve multiple component failures and significant structural damage impacting several rooms within NPR facilities that would be designed to meet the Non-Reactor Safety Class 1 Criteria (DOE Order 6430.1A). Any such accidents, whether internally or externally initiated, would have to have sufficiently large amounts of energy to cause significant release of radionuclides to the environment. Events of this magnitude are rather difficult to postulate, and the frequencies of such events are quite uncertain. The approach is thus to present a reasonable upper bound on the consequences of mechanistically possible, although extremely unlikely, severe accidents at the support facilities. It is, therefore, not the intent of this EIS to discuss all possible accident initiators leading to a severe accident at the support facilities.

Sections J.2.1 through J.2.3 provide descriptions of postulated accidents, radio-active source terms, methods of analysis, and a summary of calculated radiological impacts resulting from accidents at the fuel fabrication facility and the tritium-target processing facility at the three candidate sites.

J.2.1 Accident Descriptions and Source Terms

This analysis focuses on presenting a reasonable bound for severe accidents at NPR support facilities; therefore, a range of severe accident initiators was considered. Source terms were also developed for scenarios that can be considered as "bounding."

**TABLE J.9 Estimated Risks Resulting from Postulated
NPR Severe Accidents at the Three Alternative Sites**

Site/Reactor	Cancer Fatality (fatalities/ reactor-year) ^a	Population Dose (person-rem/ reactor-year) ^a
Hanford		
HWR	3×10^{-4}	7×10^{-1}
LWR	5×10^{-4}	1
MHTGR	2×10^{-5}	4×10^{-2}
INEL		
HWR	3×10^{-4}	7×10^{-1}
LWR	6×10^{-4}	2
MHTGR	1×10^{-5}	4×10^{-2}
SRS		
HWR	4×10^{-4}	1
LWR	6×10^{-4}	1
MHTGR	2×10^{-5}	6×10^{-2}
Commercial Plants ^b		
Surry	5×10^{-3}	3×10^1
Sequoyah	2×10^{-2}	8×10^1
Zion	2×10^{-2}	1.5×10^2
Peach Bottom	5×10^{-3}	2.9×10^1
Grand Gulf	9×10^{-4}	6
South Texas	9×10^{-3}	2×10^2
Vogtle	2×10^{-2}	3.1×10^2

^aOne reactor-year for the MHTGR technology includes operations at all eight modules.

^bSources: NRC 1985, 1986, 1989.

A significant amount of radionuclides would be released from a support facility only if multiple component failures and structural damage occurred. Internally initiated events, such as a widespread facility fire and/or explosion, have the potential to involve several rooms, but such events would not result in a major breach of radionuclide confinement. Typically, only small amounts of radionuclides would be released to the environment through building leakage, air reversals, releases from "clean" room exhausts, and so forth. Internally initiated events are, therefore, not likely candidates for severe accidents.

Externally induced events, on the other hand, have the potential for significantly greater energy and are, therefore, potential candidates for severe accidents. For example, a large aircraft crash could cause a major breach in a wall or roof of a facility and create a direct path for release of radionuclides from a process or storage area to the environment. A concurrent or subsequent fuel fire would provide additional mechanisms for volatilization of materials, rupture of confinement barriers, and release of radionuclides to the environment. However, because NPR support facilities would be large, it is very unlikely that the entire building would be at risk from

the structural damage or possible fire that could result from a large aircraft crash. The risk of direct, unfiltered release of radioactive material to the environment would most likely be limited to a few rooms in the building.

Theoretically, for most NPR support facilities, an earthquake of sufficient magnitude to cause major structural damage to a facility probably has the highest potential to involve the release of all radioactive materials within a facility to the environment. Most materials are likely to be stored in such a manner that release is unlikely even from the most severe earthquake. However, all materials in process tanks, vessels, gloveboxes, and so forth could be vulnerable. Also, a widespread fire, due to leaks in process gases such as propane, might occur. For these reasons, a severe earthquake event was assumed to be the initiator for analyzing the impacts to support facilities from severe accidents. The release inventories estimated in Sections J.2.1.1 through J.2.1.3 are considered to be reasonable upper bounds for each of the three reactor technologies (HWR, LWR, and MHTGR).

J.2.1.1 Fuel Fabrication Facility

A number of major process areas in the fuel fabrication facility are considered to be at risk in the event of a severe seismic event -- i.e., the feed powder storage vault, pellet fabrication gloveboxes, fuel pellet storage area, finished pin storage area, assembly area, and finished assembly storage area. In these process areas, uranium is the radioactive material of concern. Two categories of radioactive releases are considered for a severe accident: (1) fresh uranium particulates that are available for dispersion due to damage to the facility and (2) fission product gases caused by criticality events in general areas of the facility.

Releases of Uranium Particulates

For the first release category, uranium particulates are assumed to disperse and be released to the environment. Because of their special design and structural reinforcement, a few areas in the fuel fabrication facility are not likely to be affected and, therefore, no release contributions were assumed. Such areas include the uranium storage vault and the finished fuel assemblies stored in holes of a concrete slab or in shipping containers. In addition, no releases were assumed for the finished fuel pins, in either the storage area or the assembly area. The assembly area is characterized by open areas where there are transfer carts containing Zircaloy tubes loaded with uranium pellets that have end caps welded in place. Because of the integrity of the pin itself, potential releases are deemed possible only from the pellet fabrication gloveboxes and the fuel pellet storage area.

The fabrication glovebox area of the facility is a relatively large room containing glovebox enclosures with equipment for pressing powder into pellets; it is this area where the larger scale potential releases might occur. According to an estimate by Pajunen (1990), the release inventory would contain uranium particulates with activities on the order of 1×10^4 Ci or less. Such a release inventory is too small to warrant consideration for severe accidents. Therefore, source terms from uranium particulate releases were not included in the impact analysis.

Releases of Fission Products

Criticality events cannot be precluded as a secondary result of the primary initiating event because of the potential for powder spills and for rearrangement of materials by explosion and flooding caused by pipe failure. The NRC Regulatory Guide 3.34 (NRC 1979) describes a range of criticality events that have a mean event of 2×10^{17} fissions. However, it has been estimated that solid uranium systems could produce criticalities with a total yield of 3×10^{19} fissions. That value was first suggested as a reasonable upper bound for accidental criticality by Woodcock (1966) and is consistent with a later review by Elder et al. (1986). Thus, the total yield of 3×10^{19} fissions has been adopted as a reasonable upper bound to characterize the potential criticalities associated with a severe accident event in the fuel fabrication facility. For this accident scenario, the fission product gas spectrum given in NRC Regulatory Guide 3.34 was used as the basis for estimating releases from the criticalities. As a matter of conservatism, no credit was given to the reduction in iodine releases commonly anticipated from absorption or plate-out. The estimated release inventory from the criticality scenario is provided in Table J.10.

J.2.1.2 Tritium-Target Processing Facility

The areas and sources considered in this severe accident analysis to characterize the release from the tritium-target processing facility include the irradiated target storage area, irradiated targets in furnace processing, spent target materials in storage, tritium product processing systems, bulk product storage area, and tritium. Two of the material classes are represented by forms of tritium, i.e., elemental tritium and tritiated water. The third material class is characterized as particulate target cladding containing activation products.

In the tritium-target processing facility, the bulk of the tritium content (on the order of 1×10^7 to 2×10^7 Ci) would be located in the irradiated target storage area, in tritium processing equipment, and in the bulk storage area. However, tritium in the irradiated target storage area would be confined in target pins and not easily released during an accident; a release of about 100 Ci of tritium in elemental form was estimated for the accident (Pajunen 1990). The tritium processing equipment, on the other hand, would be located inside a glovebox-type enclosure within the facility. It was assumed that in the event of a severe accident, the entire inventory of tritium would be involved in a fire or explosion resulting from glovebox and equipment failures during the event. Sufficient electrical equipment was assumed to be located throughout the facility to provide ignition sources for fire or explosion during the accident. Therefore, the inventory was assumed to be converted to tritiated water and released from the facility. The release inventory was estimated to be 1×10^7 Ci.

The tritium content of the irradiated targets in furnace processing would be much smaller, and the release was considered to be dominated by the tritium release from the processing equipment and bulk storage area discussed above. The spent targets would contain activation products in the irradiated target claddings. Release of the activation products from the potentially pulverized spent targets would be too small (on the order of 1×10^{-3} Ci) to warrant consideration in the severe accident risk assessment. Based on the above estimates, the tritium release quantity of 1×10^7 Ci in the form of tritiated water was considered to be a reasonable bound for the tritium-target processing facility in a severe accident and was used for estimating risk impacts in this analysis.

J.2.1.3 Other Support Facilities

The tritium-target fabrication facility does not contain radioactive materials during operations and is therefore not at risk during a severe accident with regard to radioactive releases.

The heavy water processing facility would be designed for the NPR to process the existing tritiated heavy water inventory from SRS if the HWR technology is chosen for the candidate site. The inventory of tritiated heavy water at SRS is currently estimated to be 950 t (Bean 1990), with varying levels of tritium contamination (3 to 13 Ci/L). Thus, the total amount of tritium inventory at the facility would be about 1×10^7 Ci or less. Because it is unlikely that all the heavy water would evaporate during the accident, the potential release of tritium at the heavy water processing facility would be only a fraction of the total tritium inventory. Thus, the likely release inventory of tritium would be much less than 1×10^7 Ci, which is the estimated tritium release from the tritium-target processing facility. Therefore, no separate analysis was performed for the heavy water processing facility.

J.2.1.4 Probabilities of Severe Seismic Events

The estimated safe shutdown earthquake criteria are as follows: 0.3 g earthquake for Hanford, 0.35 g earthquake for INEL, and 0.3 g earthquake for SRS. No safety features of the facilities are expected to fail at or under these earthquake criteria. Peak ground acceleration (PGA) criteria for NPR facilities have been developed for the three candidate sites (Lawrence Livermore National Laboratory 1990; Murray and Wesley 1990); the facilities would be designed to meet the Non-Reactor Safety Class 1 Criteria (DOE Order 6430.1A). For a severe accident to occur, the PGA must far exceed the design-basis criteria. Based on the maximum recorded worldwide earthquake ground motion and best expert judgment, Murray and Wesley (1990) estimated upper bounds on the PGAs that are possible at the three candidate sites.

TABLE J.10 Estimated Radionuclide Release Inventories from a Postulated Severe Accident at the Fuel Fabrication Facility^a

Radionuclide	Release Activity (Ci)
Krypton-83m	4.8×10^2
Krypton-85m	4.5×10^2
Krypton-85	4.8×10^{-3}
Krypton-87	3.0×10^3
Krypton-88	2.0×10^3
Krypton-89	1.3×10^5
Xenon-131	2.5×10^{-1}
Xenon-133m	5.4
Xenon-133	8.1×10^1
Xenon-135m	6.6×10^3
Xenon-135	1.1×10^3
Xenon-137	1.5×10^5
Xenon-138	3.9×10^4
Iodine-131	2.6×10^1
Iodine-132	3.3×10^3
Iodine-133	4.8×10^2
Iodine-134	1.4×10^4
Iodine-135	1.4×10^3

^aInventory shown includes fission product gases from a postulated criticality accident with 3×10^{19} fissions. Inventory from uranium particulate releases would be very small (about 1×10^{-4} Ci or less) and therefore are not included in the source terms.

The PGA data were compared with plant fragility data to establish the basis for accident assumptions. The NPR plant fragility data were estimated using engineering judgment from a review of current design criteria and an estimate of the additional conservatism likely to be introduced by the analyst or test engineer (Lawrence Livermore National Laboratory 1990). Using these fragility data, the analysis indicates that, except for a few items of equipment (such as off-site power, circuit breakers, blowers, and fans), the NPR support facility structures and major equipment are likely to survive the upper bound PGAs. However, Murray and Wesley (1990) have suggested that because the estimated fragility data represent only approximated values, it is still possible (although not likely) that major structure/equipment failure could occur under the given PGAs. It was thus conservatively assumed that the estimated upper bound PGAs represent the severe accident initiators that would lead to the releases estimated in Section J.2.1.1 and J.2.1.2 for the NPR support facilities.

J.2.2 Methods and Site-Specific Data

The methods, MACCS computer code, and site-specific data used to calculate the radiological impacts from severe accidents at support facilities are similar to those used for severe accidents at the reactors. The calculated impacts, including population doses and health effects (early fatalities and cancer fatalities), are summarized in tabular form.

J.2.3 Results of Risk Analysis

Estimated radiological impacts from severe accidents at support facilities and estimated probabilities for postulated seismic events are summarized for the three candidate sites in Table J.11. Potential impacts are presented for the tritium-target processing facility and the fuel fabrication facility in terms of both consequence and risk (i.e., consequence times probability of event). For each facility, impacts were assessed for the general public and on-site colocated workers. The postulated severe accidents would not result in any early fatalities, and all individual risks are estimated to be much less than 1×10^{-10} cancer fatalities/yr.

For the tritium-target processing facility, the total estimated population doses (in person-rem) are 6.1×10^3 for Hanford, 7×10^3 for INEL, and 6.4×10^3 for SRS. These correspond to 3 latent cancer fatalities at Hanford, 3 at INEL, and 2 at SRS. For the fuel fabrication facility, the estimated total population doses (in person-rem) are 14 at Hanford, 7 at INEL, and 46 at SRS. These correspond to latent cancer fatalities of 5×10^{-3} at Hanford, 1×10^{-3} at INEL, and 2×10^{-2} at SRS.

The estimated total population dose risks (in person-rem/yr) for the tritium-target processing facility are 6×10^{-5} for Hanford, 7×10^{-5} for INEL, and 2×10^{-3} for SRS. For the fuel fabrication facility, the estimated total population dose risks are 1×10^{-7} for Hanford, 7×10^{-8} for INEL, and 2×10^{-5} for SRS. The corresponding health risks (in cancer fatalities/yr) are 3×10^{-8} at Hanford and INEL and 1×10^{-6} at SRS for the tritium-target processing facility and $< 1 \times 10^{-10}$ at Hanford and INEL and 7×10^{-9} at SRS for the fuel fabrication facility. The higher risk values at SRS are mainly attributable to the higher event probabilities estimated for that site.

TABLE J.11 Summary of Potential Radiological Impacts from a Bounding Severe Seismic Accident at NPP Support Facilities^a

Site and Accident Frequency (per year)	Population Type	Consequences ^b			Risks ^c			
		Population Dose (person-rem)		Cancer Fatalities		Population Dose Risk (person-rem/yr)		Cancer Risk (fatalities/yr)
		TTPF	FFF	TTPF	FFF	TTPF	FFF	
Hanford, 1×10^{-8}	General public	6×10^3	7	3	2×10^{-3}	6×10^5	7×10^8	3×10^{-8}
	Collocated workers	1×10^2	7	5×10^{-2}	3×10^{-3}	1×10^6	7×10^8	5×10^{-10}
	Total	6.1×10^3	1.4×10^1	3	5×10^{-3}	6×10^5	1×10^7	3×10^{-8}
INEL, 1×10^{-8}	General public	7×10^3	7	3	1×10^{-3}	7×10^5	7×10^8	3×10^{-8}
	Collocated workers	6	3×10^{-1}	2×10^{-3}	1×10^{-4}	6×10^8	3×10^9	$<1 \times 10^{-10}$
	Total	7×10^3	7	3	1×10^{-3}	7×10^5	7×10^8	3×10^{-8}
SRS, 4×10^{-7}	General public	6×10^3	6	2	2×10^{-3}	2×10^3	2×10^6	9×10^{-7}
	Collocated workers	4×10^2	4×10^1	1×10^{-1}	2×10^{-2}	1×10^4	2×10^5	6×10^{-9}
	Total	6.4×10^3	4.6×10^1	2	2×10^{-2}	2×10^3	2×10^5	1×10^{-6}
								7×10^{-9}

^aThe support facilities include the tritium-target processing facility (TTPF) and the fuel fabrication facility (FFF).^bIncludes only latent cancer fatalities; the accident would not result in any early fatalities.^cThe individual risks, which are not shown here, were estimated to be much less than 1×10^{-10} fatalities/yr for all facilities at all sites.

J.3 IMPACTS TO IN-PLANT WORKERS

Severe accidents involving either reactors or support facilities could result in very high radiation exposures as well as fatalities to workers located in the same building where the severe accident occurred (in-plant workers). The occurrence and severity of radiological consequences of a severe accident to in-plant workers depends on the accident scenario, the detailed design of the facility (shielding, alarms, and so forth), the ability of workers to respond adequately to the emergency, and the distance between workers and the accident location. Detailed designs for each of the three reactor technologies and their support facilities will be required by DOE to include similar provisions for worker safety at all three of the proposed sites. Thus, the consequences of severe accidents to in-plant workers presented in this analysis are not a site or technology differentiator.

To ensure worker safety, a preliminary safety analysis for the selected reactor technology and site will be prepared to identify and document any potential health, safety, or environmental problems prior to the final design process. The safety analysis process within DOE involves a so-called Safety Analysis Review System (SARS) (as prescribed by DOE Order 5481.1B). The SARS is used to ensure that health, safety, and environmental protection will be as intended prior to operation of the facility. During the final design process, a final safety analysis is performed to formally document the potential hazards and consequences and the methods, measures, or controls to be employed to eliminate or mitigate hazards and consequences. The final safety analysis is then reviewed and the review documented to allow independent evaluation of its adequacy. Elements or topics addressed in the Safety Analysis Report include safety systems and features, industrial safety, radiation protection, events and accidents, and operational safety requirements.

Radiological consequences of internally initiated severe accidents involving core damage at a reactor would depend upon adequacy of emergency response preparedness procedures and training because no credible scenario exists in which in-plant workers would not be given some advance warning. Radiation exposures would result from subsequent cleanup operations, but these would be kept below regulatory requirements by exposure monitoring and administrative procedures.

For externally initiated severe accidents, the bounding case for both reactors and support facilities was assumed to be a severe earthquake beyond the design-basis criteria for the facilities. All NPR reactor and support facility structures will be designed and constructed to survive a "design-basis" earthquake with little damage and no significant radiological consequences. However, in the event of a major earthquake, it is assumed that parts of structures would fail in both reactor and support facility structures, that uncontrolled nuclear criticalities might occur, and that large amounts of radioactivity and/or toxic materials would be released within the structures. In such an event, in-plant workers would be subject to both nonradiological and radiological hazards, the former possibly compounding the latter. For example, damages caused by a major earthquake might involve collapse of walls or ceiling components, which could in themselves cause injuries or fatalities but might also block building exits; steam, gas, water, and electric lines might be broken, again providing both a primary and secondary avenue for injury or fatality. In practice, buildings and their utilities would have been designed with sufficient conservatism to limit such damage, personnel would have been trained adequately in emergency preparedness, and processes involving fissionable material would have been isolated and shielded from areas where personnel would be routinely working. Nevertheless, a severe

earthquake -- greater than design-basis magnitude -- might well result in fatalities as a result of both radiological exposures and nonradiological consequences.

The overall risk (product of the consequence and the frequency of occurrence) to in-plant workers from such possible severe accidents, even assuming a catastrophic consequence, would be very low because the estimated frequency of occurrence is extremely small. The estimated frequency of occurrence for accidents of a type described above are between 10^{-6} and 10^{-9} per operating year of a reactor and/or support facility.

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APPENDIX K:
IMPACTS FROM TRANSPORTATION OF RADIOACTIVE MATERIALS

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IMPACTS FROM TRANSPORTATION OF RADIOACTIVE MATERIALS

The impacts of transporting NPR-related radioactive materials would include the impacts of on-site transport (within the boundaries of each of the three alternative sites) and off-site transport (outside the boundaries of the alternative sites) for both routine and accident transport conditions.

Radioactive materials transported on-site would include the feed, products, and by-products; high-level waste (HLW); transuranic (TRU) waste; and low-level waste (LLW). These materials are necessary for or are generated by operation of an NPR and its support facilities. Except for LLW, which would be disposed of on-site, radioactive materials would be transported off-site. Such materials would include tritium and plutonium metal as products, excess uranium oxides resulting from NPR operation, tritium-contaminated heavy water from the SRS reserve inventory, and HLW and TRU waste. In addition, uranium feed materials would be transported from off-site locations for fabrication into fuel. For an LWR operated in the plutonium mode at Hanford, the fuel would be shipped from an off-site commercial vendor.

Regardless of NPR location, it is assumed that the tritium product would be shipped to SRS, where it would be refined at an existing facility. It is assumed that the plutonium metal produced at any of the three alternative sites would be transported to DOE's Rocky Flats Plant. If the HWR technology is selected for either Hanford or INEL, SRS would be the shipping origin of heavy water containing tritium. In that case, the existing tritiated heavy-water inventory at SRS would be shipped to either Hanford or INEL for processing and reuse.

The proposed NPR sites would use on-site facilities for processing fuel and waste. The HLW and TRU waste would temporarily be stored on-site and eventually disposed of off-site at national repositories. Transport of waste to either the HLW repository (for HLW) or the Waste Isolation Pilot Plant (WIPP) (for TRU waste) is described and assessed in the national transportation plans and NEPA documentation (DOE 1986, 1990a) for those facilities. The results of those assessments are used here for estimating the risks associated with transport of HLW and TRU waste generated by NPR operations.

The enriched uranium oxide feed materials would be shipped from the Portsmouth plant, which would also be the destination for excess feed materials resulting from NPR operations. The off-site commercial vendor for the LWR fuel needed to operate in the plutonium mode at Hanford has not been identified. Because of the bounding approach discussed in the following paragraphs, impacts from shipping fresh fuel (including feed materials) have been determined to be bounded by those from shipping HLW and TRU waste and were therefore not evaluated separately. Both HLW and TRU waste would be generated from processing the highly radioactive irradiated fuel from NPR operations. They would therefore contain several orders of magnitude more radioactivity than fresh fuel.

Data related to producing and transporting plutonium and tritium products are generally classified. Therefore, the radioactive source term data presented in this appendix regarding these materials represent "bounding case" values intended to provide conservative scenarios for the impact analyses. The values do not represent the actual source terms involved in NPR operations. Such bounding data are provided in the NPR generic design documents for

the HWR technology (WSRC 1991), LWR technology (WHC 1990a, 1990b), and MHTGR technology (EG&G 1989).

The product plutonium and tritium would be transported in Type B containers, certified by NRC or approved by DOE, aboard trailers classified as "safe secure transport" (SST) and operated by DOE couriers. For the purpose of comparing risks among the candidate sites and technologies, all off-site shipments are assumed to be by truck. Depending on site-specific situations, the on-site shipments are assumed to be by truck or rail.

For transport of radioactive materials resulting from on-site or off-site activities, the radiological impacts are assessed collectively for the affected general population and individually for the hypothetical maximally exposed individual (maximum individual). The radiological impacts are expressed as the 50-year committed effective dose equivalent (CEDE) for exposed individuals and for the exposed population as a whole.

For incident-free operations, whether on-site or off-site, effective dose equivalents (EDEs) are calculated for the transport crew members and for the general population near the shipment route. The incident-free maximum individual doses are based on the conservative assumption that the hypothetical maximally exposed individual receives doses that are cumulative over annual shipments. The RADTRAN III computer code (Madsen et al. 1986) is used to calculate the incident-free doses.

For on-site transport, the GENII computer code (Napier et al. 1988) is used to calculate population dose equivalents under accident conditions. Site-specific parameters (e.g., site population density and meteorological data) are inputted into the code. For off-site shipments, the RADTRAN III computer code is used to calculate the collective impacts to the affected population. Parameters appropriate for the shipment routes are inputted into the code. The GENII computer code is used to assess the maximum-individual impacts from an accident for both on-site and off-site operations.

For accident conditions, the CEDEs are weighted by the accident probabilities to yield dose risks, an approach that is consistent with the methodology suggested by the existing RADTRAN computer code. To represent the bounding estimates of the impacts to individuals, the maximum-individual doses are assessed in terms of radiological consequences (i.e., CEDE). Such calculations are also based on the GENII methodology.

The impacts are further expressed as health risks in terms of cancer fatalities in exposed individuals and genetic effects in the offspring of exposed individuals. The current health risk conversion factors are suggested by EPA in 40 CFR Part 61 (EPA 1989): 4.0×10^{-4} fatal cancer cases per person-rem and 2.6×10^{-4} genetic effects per person-rem. All risks are evaluated for the three reactor technologies at each of the three alternative sites. Estimated risks from incident-free conditions are presented in Section K.1 and those from accident conditions in Section K.2.

Also estimated are risks from nonradiological causes associated with the transport of NPR-related radioactive materials for both on-site and off-site activities. The nonradiological risks considered are the fatalities occurring under traffic accident conditions. In estimating these risks, the risk factors derived by Cashwell et al. (1986) are used. The nonradiological risks are presented in Section K.3. The uncertainties in these analyses are discussed in Section K.4.

This analysis provides a technical assessment of the radiological risk associated with on-site and off-site transport of NPR-related radioactive materials. Awareness of public concerns is partly responsible for the recognition of "secondary factors" in the DOT routing guidelines (such as HM-164 as codified in 49 CFR Parts 171, 172, 173, and 177) and for the conservatism of this risk analysis (i.e., the strong tendency toward overestimating risk).

K.1 RISKS FROM INCIDENT-FREE TRANSPORT

K.1.1 Method

The RADTRAN III computer code (Madsen 1986) has been used to calculate the radiological risks from incident-free transport. This risk analysis model was developed by Sandia National Laboratories to calculate the population radiological risks associated with transport of radioactive materials by a variety of modes, including truck, rail, air, ship, and barge.

Included in the RADTRAN incident-free risk calculations for highway transport are models describing (1) doses to persons (e.g., residents) adjacent to the transport link (off-link doses), (2) doses to persons (e.g., passengers) sharing the transport link (on-link doses), (3) doses to persons at stops (e.g., refueling stops), and (4) doses to transportation crew members.

The population dose of a shipment program is estimated by summing the doses from each of the four population groups mentioned above, that is:

$$D = N_s L_s \sum_i D_i \quad (\text{K.1})$$

where:

D = dose (person-rem/yr),

N_s = number of annual shipments of a particular radioactive package (shipments/yr),

L_s = distance of shipment (km), and

D_i = dose received by i -th population group (person-rem/km-shipment).

Associated with the calculation of D_i are parameters such as the radiation field strength, source-receptor distance, exposure time, vehicle speed, stop time, and population density. The RADTRAN code users' manual contains derivations of the equations and descriptions of these parameters.

The single greatest limitation facing users of RADTRAN, or any computer code of this type, is the scarcity of data for certain input parameters. This difficulty is usually overcome by using conservative estimates of these parameters. The resulting risks tend to be overestimates (Neuhauser and Reardon 1986) but appropriate for use as bounding estimates in environmental evaluations. The sensitivity of RADTRAN risk calculations to variations in parameters was analyzed for a sample truck transport case (Neuhauser and Reardon 1986). The parameters

having the greatest effect on the incident-free risk calculation for truck transport were found to be, in decreasing order of importance: (1) source-receptor distance; (2) the factor K_0 , which accounts for the shape of the package; (3) distance traveled; (4) number of persons exposed while stopped; (5) stop time; (6) shipments per year; (7) distance from source to transportation crew; and (8) number of transportation crew members. Each of these factors is either deterministic (i.e., has a known and fixed value for the problem being analyzed) or capable of being appropriately bounded by conservative assumptions (i.e., assumptions that result in overestimation of doses).

K.1.2 Assumptions

Doses received during incident-free transport are primarily attributable to radiation (e.g., gamma rays or neutrons that can escape to the outside by penetrating the container shielding). Nonpenetrating radiation (e.g., beta or alpha particles) does not contribute to radiation exposure during incident-free transport because it is confined within the primary inner containment.

The magnitude of the risk from external exposure to penetrating radiation depends primarily on the external dose rate of the shipping container, the surrounding population densities, and the distance between the container and the population. For this EIS, the conservative NRC regulatory dose-rate limit of 10 mrem/h at 2 m from all Type B package surfaces (10 CFR 71) was assumed for all by-products containing gamma- or neutron-emitting radionuclides. Package external radiation field strengths are usually expressed in terms of dose rate (in mrem/h) at 1 m from the package surface, a value referred to as the transport index (TI). Appropriate TI values for various packages were taken from NEPA documents (DOE 1986, 1990a, 1990b) or, in the case of on-site transport, the TI values were based on site practice and experience.

Three population density zones (i.e., rural, suburban, and urban) were used for the in-transit off-site transport analyses. These zones respectively correspond to mean population densities of 6, 719, and 3,861 persons/km². For on-site transport, site-specific population densities were used. The population groups that could be exposed to radiation while the shipments were in transit would include truck drivers, individuals who handle the radioactive shipping containers during shipment, and the general public (e.g., bystanders at truck stops, persons living or working along a route, and travelers in the vicinity). In all cases, the radioactive inventory is assumed to be that associated with operating in the plutonium mode, which was determined to be the bounding case.

Within the RADTRAN methodology, calculating the maximum individual dose to a member of the public has not been an objective. Such maximum doses are subject to various exposure scenarios and are discussed in detail in a NEPA document (DOE 1986). For all scenarios studied in that document, it was determined that maximum-individual exposures would not exceed 0.005 rem/yr. For this EIS, this dose was used to represent the maximum individual dose to the public during transport of NPR-related radioactive materials.

K.1.2.1 On-site Transport

Site-specific conditions and the related input values used for the on-site transport analyses are summarized in the following paragraphs. As can be seen in the following

discussions for each site, variations in reactor technology as well as in waste processing method result in large differences in annual shipment requirements and therefore in total miles of on-site shipment. Such differences result from fuel enrichment requirements and the specific design of the reactors. For instance, shipping irradiated driver fuels for MHTGR involves shipping large amounts of graphite blocks containing fuel rods. The result is more shipments on-site. In addition, site-specific conditions result in differences in the estimated mileage. For example, the reactor and facilities at INEL are more clustered than those at either Hanford or SRS. The result is a shorter shipping distance.

Hanford Site

The Hanford Site covers an area of about 1,500 km². It has no permanent public residents. The NPR-related radioactive materials that would be transported at Hanford can generally be categorized as (1) irradiated driver fuels, (2) irradiated target rods, (3) LLW, and (4) TRU waste (both remote- and contact-handled). Other radioactive materials (e.g., HLW) would be stored at tank farms before being solidified and shipped to an off-site national repository. Product materials (i.e., plutonium and tritium) would be stored at separation facilities prior to shipment off-site. The characteristics of on-site shipment of these materials are listed in Table K.1. In the table, the population densities estimated for the routes mostly represent those of the workers at the origin and destination facilities. The estimated total shipping mileage is dominated by transport of LLW (i.e., 3,800 m³/yr for HWR to 7,800 m³/yr for MHTGR) to the on-site disposal facility.

Idaho National Engineering Laboratory

For the INEL analysis, members of the public are considered to be all persons sharing the transport link. No on-site, off-link doses to members of the public are postulated because there are no permanent public residents on-site and NPR-related facilities are some distance from routes used by the public. The NPR-related radioactive materials to be transported at INEL can be categorized as (1) irradiated driver fuels, (2) irradiated target rods, (3) LLW, and (4) TRU waste (i.e., both remote- and contact-handled). The LLW includes liquid waste transported by tank trucks to the Idaho Chemical Processing Plant (ICPP). The HLW would be calcined and held at the ICPP tank farm prior to off-site shipment to the national repository. Plutonium and tritium would be stored at the separation facilities before being shipped off-site. The characteristics of on-site shipment of these materials are listed in Table K.2. In the table, the population densities estimated for the routes mostly represent those of the workers at the origin and destination facilities. The estimated total shipping mileage is dominated by transport of LLW (i.e., 2,200 m³/yr for HWR to 5,300 m³/yr for MHTGR) to the on-site disposal facility.

Savannah River Site

For the NPR dose assessment for SRS, it was assumed that any new facilities required for current operations at SRS are also fully operational. The transportation routes for the NPR were therefore assumed to be similar to those currently used. The average population densities on each of the routes were determined from on-site worker population statistics specific to the affected areas.

TABLE K.1 Characteristics of On-site Shipments of Radioactive Materials at Hanford^a

Material	Origin ^b	Destination ^b	Transport Mode	Package Type	Transport Index	Speed of Shipment (km/h)	Average Route Pop. Density (persons/km ²)	Total Travel Distance (km/yr)		
								HWR	LWR	MHTGR
Irradiated driver fuels	NPR	SFRC	Rail	B	14	24	109 ^c	1,230	4,880	30,800
Irradiated targets	NPR	SFRC	Rail	B	14	24	109 ^c	2,800	d	28,800
LLW	TPF, SFPC, NPR, FTFF	On-site burial	Truck	A	10	56	109	44,000	41,000	91,000
TRU waste										
Contact-handled	SFPC	WRAP	Truck	B	0.15	56	109	180	530	270
Remote-handled	SFPC	WRAP	Truck	B	14	56	109	220	610	390

^aInventory is based on operating in the plutonium mode, which was determined to be the bounding case.^bFTFF is Fuel and Target Fabrication Facility, SFPC is Spent Fuel Processing Complex, TPF is Target Processing Facility, and WRAP is Waste Receiving and Packaging Facility.^cFor MHTGR, the average population density is 98 persons/km².^dFor LWR, the targets are in the driver fuel rods.

Source: Evans 1990.

TABLE K.2 Characteristics of On-site Shipments of Radioactive Materials at INEL^a

Material	Origin ^b	Destination ^b	Transport Mode	Package Type	Transport Index	Speed of Shipment (km/h)	Average Route Pop. Density (persons/km ²)	Total Travel Distance (km/yr)		
								HWR	LWR	MHTGR
Irradiated driver fuels	NPR	ICPP	Truck	B	14	24	229	448	780	5,640
Irradiated targets	NPR	PTPF	Truck	B	14	24	229	1,024	c	5,275
LLW	NPR, FFF, PTFE, ICPP, PTFE	ICPP, WERF, RWMC	Tank truck	A	10	24	229	40	40	7
			Truck	A	10	24	229	270	122	444
			Truck	A	10	24	229	1,430	1,450	3,475
TRU waste	ICPP	TSA	Truck	B	0.15	24	229	1,325	150	2,250
Contact-handled	ICPP	TSA	Truck	B	14	24	229	150	125	775

^aInventory is based on operating in the plutonium mode, which was determined to be the bounding case.

^bFFF is Fuel Fabrication Facility, ICPP is Idaho Chemical Processing Plant, PTFE is Plutonium-Target Fabrication Facility, PTFE is Plutonium-Target Processing Facility, RWMC is Radioactive Waste Management Complex, TSA is Transuranic Storage Area, and WERF is Waste Experimental Reduction Facility.

^cFor LWR, the targets are in the driver fuel rods.

Source: Maheras 1990.

The NPR-related radioactive materials that would be transported at SRS can generally be categorized as (1) irradiated driver fuels, (2) irradiated target rods, (3) LLW, and (4) TRU waste (contact-handled only). Other radioactive materials (e.g., HLW) would be transferred by encapsulated pipe to the S Area for temporary storage before being solidified and shipped off-site to the national repository. Product materials (i.e., plutonium and tritium) would be stored at separation facilities before shipment off-site. The characteristics of on-site shipment of these materials are listed in Table K.3. In the table, the population densities estimated for the routes mostly represent those of the workers at the origin and destination facilities. The estimated total shipping mileage is dominated by transport of LLW (i.e., 1,100 m³/yr for HWR to 12,600 m³/yr for MHTGR) to the on-site disposal facility.

K.1.2.2 Off-site Transportation

Shipment of tritium or tritiated heavy water during incident-free operations would not cause external exposure to surrounding personnel because beta radiation (the only radiation emitted by tritium) cannot penetrate shipping containers. The risks of off-site transport would therefore be mainly the risks from shipping (1) plutonium from the NPR sites to the Rocky Flats Plant in Colorado, (2) HLW to the proposed national repository at Yucca Mountain in Nevada, and (3) TRU waste to the WIPP site near Carlsbad, New Mexico.

The representative Interstate highway routes from each potential origin to the potential destination for products (i.e., plutonium and tritium) and tritiated heavy water were from the EIS for the Special Isotope Separation Project (DOE 1988). The origins and destinations considered for that project are similar to those considered for the NPR. Risk information regarding transport of HLW and TRU waste was taken from existing NEPA documents (DOE 1986, 1990a). Table K.4 lists the distances of those routes and gives fractions of travel in rural, suburban, and urban population density zones for product materials and tritiated heavy water. Similar data for HLW and TRU waste are given in Table K.5.

All HLW and TRU waste from NPR operations were assumed to be temporarily stored on-site before being shipped to the proposed repository site at Yucca Mountain or to the WIPP site near Carlsbad for permanent disposal. For assessing risks associated with off-site transport of HLW and TRU waste, the incident-free risk factors developed in NEPA documentation for HLW (DOE 1986) and TRU waste (DOE 1990a) were used. The risk factors that apply to the three NPR sites are presented in Table K.6. The dose factors were calculated with the RADTRAN III computer code.

K.1.3 Risk Impacts from Incident-Free Transport

K.1.3.1 On-site Transport

Table K.7 summarizes the radiological risk from on-site, incident-free transport of NPR-related radioactive materials at Hanford, INEL, and SRS.

TABLE K.3 Characteristics of On-site Shipments of Radioactive Materials at SRS^a

Material	Origin ^b	Destination ^b	Transport Mode	Package Type	Transport Index	Speed of Shipment (km/h)	Average Route Pop. Density (persons/km ²)	Total Travel Distance (km/yr)		
								HWR	LWR	MHTGR
Irradiated driver fuels	100 Area	200 Areas	Rail	B	14	27	120	350	1,200	9,000
Irradiated targets	100 Area	200 Areas	Rail	B	14	24	120	810	c	8,300
LLW	100, 200 Areas	100 Area	Truck	A	10	56	120	66,000	110,000	75,000
TRU waste Contact-handled Remote-handled ^d	200 Areas 3/700 Area	G Area	Truck	B	0.15	56	410	370	130	500

^aInventory is based on operating in the plutonium mode, which was determined to be the bounding case.

^bAreas represent site designations for various facilities.

^cFor LWR, the targets are in the driver fuel rods.

^dNot applicable.

Source: McGough 1990a.

TABLE K.4 Characteristics of Off-site Radioactive Shipments for NPR Alternative Sites for Product Materials and Tritiated Heavy Water

Material	Origin/ Destination	Bounding Case Inventory		Estimated Shipments per Year ^a	Distance per Shipment (km)	Type of Travel (%)		
		Technology	Ci			Rural	Suburban	Urban
Tritium	Hanford/SRS INEL/SRS	All	9.8×10^7	50	4,539	75	23	2
		All	9.8×10^7	50	3,797	75	24	1
Tritiated heavy water	SRS/Hanford SRS/INEL	HWR	1.4×10^7	53 ^b	4,539	75	23	2
		HWR	1.4×10^7	53 ^b	3,797	75	24	1
Plutonium metal product ^c	Hanford/Rocky Flats	HWR	5.8×10^5	10	1,718	83	15	2
		LWR	1.8×10^6	28				
		MHTGR	2.3×10^6	22				
	INEL/Rocky Flats	HWR	5.8×10^5	10	1,113	87	12	1
		LWR	1.8×10^6	28				
		MHTGR	2.3×10^6	22				
	SRS/Rocky Flats	HWR	5.8×10^5	10	2,672	73	26	1
		LWR	1.8×10^6	28				
		MHTGR	2.3×10^6	22				

^aFor tritium and plutonium shipments, it was assumed that 10 casks would be carried per SST shipment.^bOne-time shipment. It was assumed that heavy water would be shipped by truck; each truckload would have a capacity of 80 55-gal drums.^cThe quantities of plutonium reflect the bounding inventories as reported in NPR generic design documents for the different reactor technologies (WSRC 1991; WHC 1990a, 1990b; EG&G 1989).

Source: Based on data in DOE 1988.

TABLE K.5 Characteristics of Off-site Radioactive Shipments for NPR Alternative Sites for HLW and TRU Waste

Material	Origin/ Destination	Bounding Case Inventory		Estimated Shipments per Year ^a	Distance per Shipment (km)	Type of Travel (%)		
		Technology	(m ³ /yr)			Rural	Suburban	Urban
HLW	Hanford/Repository	HWR	106	177	1,840	82	18	0
		LWR	120	200				
		MHTGR	38	64				
	INEL/Repository	HWR	20	34	1,184	80	19	1
		LWR	65	109				
		MHTGR	24	40				
	SRS/Repository	HWR	15	25	3,760	82	17	1
		LWR	29	49				
		MHTGR	29	49				
TRU waste ^b	Hanford/WIPP	HWR	87	11	3,061	85.7	13.4	0.9
		LWR	250	30				
		MHTGR	130	16				
	INEL/WIPP	HWR	114	13	2,434	85.0	13.8	1.2
		LWR	19	3				
		MHTGR	201	24				
	SRS/WIPP	HLW	107	13	2,536	74.3	25.1	0.6
		LWR	107	13				
		MHTGR	145	17				

^aThe volume of an HLW canister was assumed to be 0.6 m³.

^bWaste consists of contact- and remote-handled TRU waste.

Sources: Inventory data are from Appendix M of this document; waste shipment data are from existing NEPA documentation (DOE 1986, 1990a).

TABLE K.6 Incident-Free Radiological Risk Factors for Off-site Transport of HLW and TRU Waste^a

Material	Origin/Destination	Maximum-Individual Dose (rem/shipment)	Population Dose (person-rem/shipment)
HLW	Hanford/Repository	2.3×10^{-2}	3.1×10^{-1}
	INEL/Repository	1.5×10^{-2}	1.9×10^{-1}
	SRS/Repository	4.8×10^{-2}	6.1×10^{-1}
TRU waste ^b			
CH TRU	Hanford/WIPP	1.0×10^{-4}	6.0×10^{-2}
	INEL/WIPP	2.3×10^{-4}	7.0×10^{-2}
	SRS/WIPP	3.1×10^{-4}	2.1×10^{-1}
RH TRU	Hanford/WIPP	4.0×10^{-4}	5.0×10^{-1}
	INEL/WIPP	2.4×10^{-5}	1.8×10^{-1}
	SRS/WIPP	c	c

^aDose includes occupational and nonoccupational exposures.

^bCH TRU is contact-handled TRU waste; RH TRU is remote-handled TRU waste.

^cThere is no RH TRU at SRS.

Sources: Data are derived from existing NEPA documentation (DOE 1986, 1990a).

Hanford Site

For the Hanford Site, on-site shipments of NPR-related radioactive materials result in a total collective dose of (in person-rem/yr) 1.5 for HWR, 1.2 for LWR, and 3.4 for MHTGR. These doses correspond to annual health risks of (in cancer fatalities) 6×10^{-4} for HWR, 5×10^{-4} for LWR, and 0.001 for MHTGR.

The annual doses to maximally exposed individuals (i.e., members of the truck crews) were calculated to be (in rem/yr) 0.21 for HWR, 0.15 for LWR, and 0.33 for MHTGR. These maximum-individual doses correspond to annual cancer risks of 8×10^{-5} for HWR, 6×10^{-5} for LWR, and 1×10^{-4} for MHTGR.

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The total collective dose from on-site transport of NPR-related radioactive materials at INEL is estimated to be (in person-rem/yr) 0.11 for HWR, 0.075 for LWR, and 0.54 for MHTGR. The estimated annual health risks from these doses are (in cancer fatalities) 4×10^{-5} for HWR, 3×10^{-5} for LWR, and 2×10^{-4} for MHTGR. The estimated maximum-individual doses to crew

TABLE K.7 Estimated Radiological Risks from On-site, Incident-Free Transport Operations

Site	Technology	Maximum- Individual Dose ^a (rem/yr)	Maximum- Individual Health Risk (fatalities/yr)	Population Dose (person-rem/yr)	Population Health Risk (fatalities/yr)	Population Genetic Risk (cases/yr)
Hanford	HWR	2.1×10^{-1}	8×10^{-5}	1.5	6×10^{-4}	4×10^{-4}
	LWR	1.5×10^{-1}	6×10^{-5}	1.2	5×10^{-4}	3×10^{-4}
	MHTGR	3.3×10^{-1}	1×10^{-4}	3.4	1×10^{-3}	8×10^{-4}
INEL	HWR	2.4×10^{-3}	1×10^{-6}	1.1×10^{-1}	4×10^{-5}	3×10^{-5}
	LWR	2.2×10^{-3}	9×10^{-7}	7.5×10^{-2}	3×10^{-5}	2×10^{-5}
	MHTGR	1.0×10^{-2}	4×10^{-6}	5.4×10^{-1}	2×10^{-4}	1×10^{-4}
SRS	HWR	4.7×10^{-1}	2×10^{-4}	5.2	2×10^{-3}	1×10^{-3}
	LWR	7.7×10^{-1}	3×10^{-4}	8.5	3×10^{-3}	2×10^{-3}
	MHTGR	5.4×10^{-1}	2×10^{-4}	6.0	2×10^{-3}	2×10^{-3}

^aMaximum-individual dose applies to transportation crew members.

members are (in rem/yr) 0.0024 for HWR, 0.0022 for LWR, and 0.01 for MHTGR. These maximum-individual doses correspond to annual cancer risks of 1×10^{-6} for HWR, 9×10^{-7} for LWR, and 4×10^{-6} for MHTGR.

Savannah River Site

On-site transport of NPR-related radioactive materials at SRS results in total collective doses of (in person-rem/yr) 5.2 for HWR, 8.5 for LWR, and 6.5 for MHTGR. These doses correspond to an estimated annual health risk of (in cancer fatalities) 0.002 for HWR, 0.003 for LWR, and 0.002 for MHTGR. The estimated maximum-individual doses to crew members are (in rem/yr) 0.47 for HWR, 0.77 for LWR, and 0.54 for MHTGR. These maximum-individual doses correspond to annual cancer risks of 2×10^{-4} for HWR, 2×10^{-4} for LWR, and 3×10^{-4} for MHTGR.

K.1.3.2 Off-site Transportation

Because tritium is a beta emitter, no radiological risk is anticipated from off-site, incident-free transport of tritium and tritiated heavy water (Section K.1.2). The estimated radiological risks for off-site, incident-free transport of plutonium for the three alternative NPR sites (calculated with the RADTRAN III computer code) are summarized in Table K.8. The radiological risks associated with off-site transport of HLW and TRU waste are given in Table K.9. These estimated risks were derived from the inventory data in Table K.5 and the risk factors in Table K.6.

The estimated doses for either the maximally exposed individuals or the general population are the same, regardless of the technology. The maximally exposed individual was identified as a crew member. These maximum-individual doses were estimated on the assumption that a crew can work no more than 250 days per year. In practice, all radiation doses would be maintained in accordance with the as low as reasonably achievable principle, as specified in DOE orders 5480.11 for occupational workers (e.g., crew members) and 5400.5 for the general public.

Hanford Site

The estimated annual total population doses for off-site shipments of NPR-related radioactive materials from Hanford are (in person-rem) 58 for HWR, 71 for LWR, and 25 for MHTGR. The corresponding annual health risks are (in cancer fatalities) 0.02 for HWR, 0.03 for LWR, and 0.01 for MHTGR. The corresponding estimated annual genetic risks are (in cases) 0.02 for HWR and LWR and 0.007 for MHTGR. More than 80% of the estimated risks results from transport of HLW to the national repository.

The maximum annual individual doses are received by truck crew members transporting HLW. The estimated doses are 1.5 rem for all technologies. The corresponding maximum annual cancer risk is 6×10^{-4} .

TABLE K.8 Estimated Radiological Risks from Off-site, Incident-Free Transport of Product Materials and Tritiated Heavy Water from NPR Operations

Material	Origin/ Destination	Technology	Maximum- Individual Dose ^a (rem/yr)	Maximum- Individual Health Risk (fatalities/yr)	Population Dose (person- rem/yr)	Population Health Risk (fatalities/yr)	Population Genetic Risk (cases/yr)
Tritium	Hanford/SRS INEL/SRS	All	0	0	0	0	0
		All	0	0	0	0	0
Tritiated heavy water ^b	SRS/Hanford SRS/INEL	HWR	0	0	0	0	0
		HWR	0	0	0	0	0
Plutonium metal product	Hanford/ Rocky Flats	HWR	1.4×10^{-2}	6×10^{-6}	3.7×10^{-1}	1×10^{-4}	1×10^{-4}
		LWR	4.0×10^{-2}	2×10^{-5}	1.04	4×10^{-4}	3×10^{-4}
		MHTGR	3.1×10^{-2}	1×10^{-5}	8.2×10^{-1}	3×10^{-4}	2×10^{-4}
	INEL/ Rocky Flats	HWR	9.0×10^{-3}	4×10^{-6}	2.3×10^{-1}	9×10^{-5}	6×10^{-5}
		LWR	2.5×10^{-3}	1×10^{-6}	6.5×10^{-1}	3×10^{-4}	2×10^{-4}
		MHTGR	1.9×10^{-3}	8×10^{-7}	5.1×10^{-1}	2×10^{-4}	1×10^{-4}
	SRS/ Rocky Flats	HWR	2.4×10^{-3}	1×10^{-6}	6.1×10^{-1}	2×10^{-4}	2×10^{-4}
		LWR	6.7×10^{-3}	3×10^{-6}	1.69	7×10^{-4}	4×10^{-4}
		MHTGR	5.2×10^{-3}	2×10^{-6}	1.33	5×10^{-4}	3×10^{-4}

^aMaximally exposed individual is a crew member.

^bHeavy water is needed only for the HWR technology.

TABLE K.9 Estimated Radiological Risks from Off-site, Incident-Free Transport of HLW and TRU Waste from NPR Operations

Material	Origin/ Destination	Technology	Maximum- Individual Dose ^a (rem/yr)	Maximum- Individual Health Risk (fatalities/yr)	Population Dose (person- rem/yr)	Population Health Risk (fatalities/yr)	Population Genetic Risk (cases/yr)
HLW	Hanford/Repository	HWR	1.5	6×10^{-4}	55	2×10^{-2}	1×10^{-2}
		LWR	1.5	6×10^{-4}	62	2×10^{-2}	2×10^{-2}
		MHTGR	1.5	6×10^{-4}	20	8×10^{-3}	5×10^{-3}
	INEL/Repository	HWR	5.1×10^{-1}	2×10^{-4}	6.5	3×10^{-3}	2×10^{-3}
		LWR	1.5	6×10^{-4}	21	8×10^{-3}	5×10^{-3}
		MHTGR	6.0×10^{-1}	2×10^{-4}	7.6	3×10^{-3}	2×10^{-3}
	SRS/Repository	HWR	1.2	5×10^{-4}	15	6×10^{-3}	4×10^{-3}
		LWR	1.2	5×10^{-4}	30	1×10^{-2}	8×10^{-3}
		MHTGR	1.2	5×10^{-4}	30	1×10^{-2}	8×10^{-3}
TRU waste	Hanford/WIPP	HWR	2.6×10^{-4}	1×10^{-7}	2.9	1×10^{-3}	8×10^{-4}
		LWR	6.9×10^{-4}	3×10^{-7}	7.5	3×10^{-3}	2×10^{-3}
		MHTGR	3.7×10^{-4}	1×10^{-7}	4.0	2×10^{-3}	1×10^{-3}
	INEL/WIPP	HWR	2.8×10^{-4}	1×10^{-7}	1.0	4×10^{-4}	3×10^{-4}
		LWR	4.8×10^{-5}	2×10^{-8}	0.32	1×10^{-4}	8×10^{-5}
		MHTGR	4.1×10^{-4}	2×10^{-7}	2.5	1×10^{-3}	7×10^{-4}
	SRS/WIPP	HWR	4.0×10^{-4}	2×10^{-7}	2.7	1×10^{-3}	7×10^{-4}
		LWR	4.0×10^{-4}	2×10^{-7}	2.7	1×10^{-3}	7×10^{-4}
		MHTGR	5.3×10^{-4}	2×10^{-7}	3.6	1×10^{-3}	9×10^{-4}

^aMaximally exposed individual is a crew member.

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The estimated annual population doses for off-site shipments of NPR-related radioactive materials from INEL are (in person-rem) 7.7 for HWR, 2.2 for LWR, and 11 for MHTGR. The corresponding annual health risks are (in cancer fatalities) 0.003 for HWR, 0.0009 for LWR, and 0.004 for MHTGR. The corresponding annual genetic risks are (in cases) 0.002 for HWR, 0.0006 for LWR, and 0.003 for MHTGR. About 70% or more of the estimated risks results from transport of HLW to the national repository.

The maximum annual individual doses are received by truck crew members transporting HLW. These doses are (in rem) 0.51 for HWR, 1.5 for LWR, and 0.60 for MHTGR. These doses correspond to annual cancer risks of 2×10^{-4} for HWR, 6×10^{-4} for LWR, and 2×10^{-4} for MHTGR.

Savannah River Site

The estimated annual population doses for off-site shipments of NPR-related radioactive materials from SRS are (in person-rem) 18 for HWR, 34 for LWR, and 35 for MHTGR. The corresponding annual health risks are (in cancer fatalities) 0.007 for HWR and 0.01 for LWR and MHTGR. The corresponding annual genetic risks are (in cases) 0.005 for HWR and 0.009 for LWR and MHTGR. More than 80% of the estimated risks results from transport of HLW to the national repository.

The maximum annual individual doses are received by truck crew members transporting HLW. The estimated doses are 1.2 rem for all technologies. The corresponding maximum annual cancer risk is 5×10^{-4} .

K.2 RISKS FROM TRANSPORT UNDER ACCIDENT CONDITIONS

K.2.1 Method

Accident risk is generically defined as the consequences of an accident multiplied by the probability of the occurrence of that accident. In actuality, there are any number of accident sequences, each of which has an associated probability. For this analysis of the radiological impacts from NPR-related transportation accidents, the GENII computer code (Napier et al. 1988) was used for on-site transportation accidents and the RADTRAN III computer code (Madsen et al. 1986) for off-site transportation accidents. For calculating the risks to maximally exposed individuals, the GENII computer code was used. Site-specific data, such as meteorological information and population densities, were entered as appropriate.

For accident conditions, the population dose risk for a particular shipment campaign is calculated by:

$$D_R = N_s L_s P_{tr} \sum_{i=1} \left(\sum_{j=1} P_j D_{ij} \right) \quad (K.2)$$

where:

DR = population dose risk resulting from accident-related exposure
(person-rem/yr),

N_s = number of annual shipments of a particular radioactive package,

L_s = distance of shipment (km),

P_{tr} = frequency of traffic accidents per unit distance traveled
(accidents/km),

P_j = conditional probability of occurrence for accident severity category j
among all traffic accidents, and

D_{ij} = dose resulting from accident severity category j through environmental pathway i .

The parameters used in Eq. K.2 to calculate accident population dose risk are discussed below.

Accident severity categories for all potential traffic accidents are described in NUREG-0170 (NRC 1977). Severity is described as a function of the magnitudes of the mechanical forces (impact) and thermal forces (fire) to which a package may be subjected during an accident. Because all accidents can be described in these terms, severity is independent of the accident scenario. In other words, any sequence of events that results in an accident in which a transport package is subjected to forces within a certain range of values is assigned to the accident severity category associated with that range.

The severity category matrix represents a set of scenarios defined by a combination of mechanical and thermal forces. A conditional probability (P_j) is assigned in each category. For example, Category I accidents are the least severe but most frequent, whereas Category VIII accidents are very severe but very infrequent. To determine the expected frequency of each severity category, the conditional probability (P) in each category is multiplied by the baseline accident rate (P_{tr}). Each population density zone has a distinct baseline accident rate and distribution of accident severities related to differences in average vehicle velocity, traffic density, and other factors, including location (i.e., rural, suburban, or urban area).

Radiological consequences (D_{ij}) are calculated by assigning release fractions to each category for each chemically and physically distinct radioisotope. The release fraction is defined as the fraction of the radioisotope group in the package that could be released from the package in a given severity of accident. Release fractions vary according to package type. Most solid radionuclides are difficult to release in particulate form and are therefore relatively nondispersible. Gaseous radionuclides are relatively easy to release if the container is compromised. The population doses are calculated for the following pathways:

- Doses from external exposure to the contaminated ground,
- Doses from external exposure to the radioactive cloud,

- Doses from inhalation of airborne contaminants, and
- Doses from ingestion of foods grown on contaminated ground.

For the on-site analysis, the GENII computer code was used to calculate population dose equivalents. A description of the GENII code is provided in Appendix H. Site-specific information, such as site population density and meteorological data, was used. For the off-site analysis with RADTRAN, appropriate parameters along the shipment routes were used. Doses due to ingestion of water are not calculated since detailed water distributions along transportation routes are very speculative, and doses are expected to be extremely small (DOE 1986).

For the hypothetical maximally exposed individual, the GENII code was used to calculate the radiological consequences of the accident corresponding to the highest possible severity category. To assess the radiological consequences for individuals, the bounding estimate of the impacts to individuals under various transportation accident scenarios was represented. For the on-site transportation analysis, two locations were used for assessing the impact to the maximally exposed individual. One location was for on-site personnel and the other for members of the general public located at the site boundary.

K.2.2 Assumptions

For accidents involving tritium releases, absorption through the skin followed by internal exposure was considered in addition to environmental pathways. The radiological hazard associated with the internal pathway depends strongly on the concentration of tritium as water vapor in the atmosphere. Elemental tritium produces doses that are much smaller (by a factor of about 6×10^{-4}) than those produced by tritium in the form of water vapor.

The additional exposure pathway of tritium absorption through the skin was approximated by estimating tritium's fraction of total water intake, as suggested in NUREG/CR-3332 (NRC 1983). This value was entered into the RADTRAN dosimetric model to calculate the dose. Because of the difficulty in determining the fractional concentration of tritium in water (NCRP 1979) in terms of the tritium equilibrium concentration in the atmosphere, it was assumed that the equilibrium concentration (in 100% water vapor) is readily established at the moment of release. This very conservative approach results in a calculated dose that is usually an overestimate.

The accident conditions for the various severity categories are described in NUREG-0170 (NRC 1977). Eight accident categories are designated for Type A, Type B, and low-specific-activity (LSA) strong-type packages for various radioactive materials shipped on-site or off-site. Conditional probabilities (P_i in Eq. K.2) were developed for the severity categories for truck shipments (Table K.10). The generic release fractions for deriving dose (D_{ij} in Eq. K.2) (Table K.11) for the severity categories are also estimated in NUREG-0170 (NRC 1977). Those specific to spent fuel and HLW are documented in Cashwell et al. (1986). For on-site transport, accidents of very high severity are not expected because traffic is controlled such that vehicle speeds are lower than those on public highways and railways. For this reason, severity categories beyond category VI (those generally with frequencies of 1×10^{-6} /yr or lower) are not considered in the on-site transportation analysis. Although postaccident remedial activities (e.g.,

**TABLE K.10 Fractional Occurrences for Truck Accidents
by Accident Severity Category and Population Density
Zone**

Severity Category	Fractional Occurrence	Fractional Occurrence by Population Density Zone		
		Low	Medium	High
I	5.5×10^{-1}	1×10^{-1}	1×10^{-1}	8×10^{-1}
II	3.6×10^{-1}	1×10^{-1}	1×10^{-1}	8×10^{-1}
III	7.0×10^{-2}	3×10^{-1}	4×10^{-1}	3×10^{-1}
IV	1.6×10^{-2}	3×10^{-1}	4×10^{-1}	3×10^{-1}
V	2.8×10^{-3}	5×10^{-1}	3×10^{-1}	2×10^{-1}
VI	1.1×10^{-3}	7×10^{-1}	2×10^{-1}	1×10^{-1}
VII	8.5×10^{-5}	8×10^{-1}	1×10^{-1}	1×10^{-1}
VIII	1.5×10^{-5}	9×10^{-1}	5×10^{-2}	5×10^{-2}

Source: NRC 1977.

**TABLE K.11 Estimated Release Fractions for Shipping Containers
under Various Accident Severity Categories**

Severity Category	LSA Drum	Type A	Type B		Integral Container Vehicle
			No Plutonium	1985 Plutonium	
I	0	0	0	0	0
II	0.01	0.01	0	0	0
III	0.1	0.1	0.01	0	0
IV	1.0	1.0	0.1	0	0
V	1.0	1.0	1.0	0	0
VI	1.0	1.0	1.0	0	0
VII	1.0	1.0	1.0	0.01	0
VIII	1.0	1.0	1.0	0.1	0.1

Source: NRC 1977.

evacuation or ground cleanup) would reduce the consequences of an accident, these activities were not given credit in the dose assessment. The estimated impacts would certainly be substantially reduced by accident remedial actions taken in accordance with Federal guidelines (EPA 1990).

For this analysis, it was assumed that all transport casks associated with NPR-related activities would be in compliance with appropriate DOT and NRC regulations. This assumption applied in both the on- and off-site transportation analyses. For transport of plutonium, the existing package is one of the few NRC-certified packages for which a large amount of data exists on its response to the higher severity category accidents. Therefore, the release fractions used in this analysis were based on those data (Bonzon 1977; Fischer et al. 1987).

For this analysis, it was assumed that the product tritium and plutonium would be shipped off-site in SST trailers. The SST trailer was designed to act as an impact-force-reduction barrier and also to provide thermal protection. The SST trailer must be transported at the posted highway speed limit and accompanied by armed escorts. The SST trailers are essentially mobile vaults with built-in deterrent devices and special electronically coded locks set in a vault-type door. They are operated by specially trained DOE personnel to provide safe and secure transport. The responses of an SST trailer under severe accident conditions are assumed to be similar to those described for the integral container vehicle in NUREG-0170 (NRC 1977). In this instance, only the most severe accident category (severity category VIII) could potentially cause any releases. Such events are extremely rare. The risks associated with transporting HLW or TRU waste have been assessed generally in existing NEPA documents. The results of these studies were used to estimate the incremental risks for off-site shipment of NPR-related HLW and TRU waste.

To provide bounding consequences, the release fraction for tritium was assumed to be 100% if the container's integrity was compromised. Moreover, for product tritium and plutonium shipments, a conservative assumption was made for the bounding case inventory per cask. The assumed inventory was based on the maximum capacity (190,000 Ci for tritium and 4.4 kg for plutonium) of NRC-certified Type B containers provided by SRS. For on-site transport of TRU waste, the release fractions developed for WIPP (DOE 1990a) were assumed. These release fractions are based on NRC-certified Type B packages (contact- and remote-handled casks).

K.2.2.1 On-site Transport

Hanford Site

Transportation modes, package types, and annual on-site distances for the Hanford Site are given in Table K.1. All site-specific parameters (e.g., population and meteorological data) are similar to those used in this EIS for the reactor analysis.

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Transportation modes, package types, and annual on-site distances for INEL are given in Table K.2. All site-specific parameters (e.g., population and meteorological data) are similar to those used in this EIS for the reactor analysis.

Savannah River Site

Transportation modes, package types, and annual on-site distances for SRS are given in Table K.3. All site-specific parameters (e.g., population and meteorological data) are similar to those used in this EIS for the reactor analysis. For the NPR dose assessment, it was assumed that all existing support facilities and any new facilities required for current operations at SRS are also fully operational. The transportation routes assumed for this NPR analysis are therefore similar to those currently used.

K.2.2.2 Off-site Transport

Traffic accident rates for the off-site transportation analysis for product materials and tritiated heavy water were taken from the EIS for the Special Isotope Separation Project (DOE 1988). Data for the off-site transportation analysis in that document were derived from the DOT accident data base for the entire commercial shipping industry (i.e., accidents on Interstate highways involving at least one commercial tractor trailer, regardless of payload). The accident data are given in terms of millions of vehicle-kilometers traveled. Such national average rates are considered appropriate for relatively long distance routes that traverse several states.

Table K.12 presents various accident rates (P_{tr} in Eq. K.2) based on state-averaged data for tractor trailers on Interstate highways. Along NPR off-site shipment routes, the accident rate ranges from 2.1×10^{-7} to 4.0×10^{-7} accidents/km. This limited variability in accident rates supports the use of national average data for the NPR off-site shipment analyses.

Current DOT regulations allow shipment of tritiated water between exempt facilities (i.e., DOE facilities) in concentrations up to 5 Ci/L as LSA material, provided the shipments are made via exclusive-use transport in strong, tight containers. Historically, SRS has used special 55-gal, leak-tight stainless steel drums to satisfy the requirement for strong, tight containers. Such LSA shipments require no special permits other than the customary shipping manifest, bill of lading, and appropriate placarding. If a single shipment contains 30,000 Ci or more of tritium in the tritiated heavy water, the shipment is classified as a "highway route controlled" quantity, and special provisions apply. These provisions include advance selection of the most direct routes on the Interstate highway system.

The DOE regulations are being revised to conform to the Radioactive Materials Transportation Guidelines of the International Atomic Energy Agency (McGough 1990b). This revision, which is expected to be in effect in the near future, allows shipment of tritiated heavy water in concentrations up to 20 Ci/L under the stipulations described above. For this analysis, NPR-related shipment of tritiated heavy water is assumed to be in compliance with this revised regulation.

The inventory of tritiated heavy water at SRS is estimated to be 950 t (Bean 1990), with the level of tritium contamination varying from 3 to 13 Ci/L. Transport of the heavy-water inventory would be needed if the HWR technology were selected at either Hanford or INEL. If each shipment carries 80 drums, the total truckload would be 18 t; therefore, about 53 shipments would be required to transport this inventory from SRS. Assuming an upper-bound concentration of 13 Ci/L, a truckload would contain about 2×10^5 Ci of tritium.

TABLE K.12 State-Averaged Accident Rates, 1984

Routes/States	Accident Rate ^a (10 ⁻⁷ accidents/km)
Along routes between	
Hanford and SRS	3.7
INEL and SRS	4.0
Hanford and Rocky Flats	3.5
INEL and Rocky Flats	4.0
SRS and Rocky Flats	3.2
All states along all NPR routes	3.2
All states	3.1

^aAccidents involving tractor trailers.

Source: DOE 1988.

As discussed in Section K.1.2, radiological risks derived from off-site transportation risk information were derived from analyses performed in the NEPA documentation for HLW (DOE 1986) and TRU waste (DOE 1990a). Risk factors from transportation accidents during shipment to the waste disposal sites are presented in Table K.13. These data are also based on RADTRAN calculations.

K.2.3 Risk Impacts under Accident Conditions

K.2.3.1 On-site Accidents

The radiological dose risks and health risks from on-site transportation accidents involving NPR-related radioactive materials were calculated for the hypothetical maximally exposed individual (Table K.14) and for the general population (Table K.15). Two cases were analyzed for the maximally exposed individual. The first individual is an on-site worker located 100 m from the accident. The second individual is a member of the general public residing at the site boundary.

Hanford Site

The estimated annual population dose risks from postulated on-site accidents at the Hanford Site are (in person-rem) 0.0025 for HWR, 0.011 for LWR, and 0.0045 for MHTGR. These values correspond to annual health risks of (in cancer fatalities) 1×10^{-6} for HWR, 4×10^{-6} for LWR, and 2×10^{-6} for MHTGR. Annual genetic effects are (in cases) 2×10^{-7} for HWR, 2×10^{-6} for LWR, and 3×10^{-7} for MHTGR.

TABLE K.13 Radiological Risk Factors for Off-site Transportation Accidents Involving HLW and TRU Waste

Material	Origin/Destination	Population Dose Risk (person-rem/shipment)
HLW	Hanford/Repository	1.8×10^{-4}
	INEL/Repository	1.4×10^{-4}
	SRS/Repository	3.2×10^{-4}
TRU waste ^a		
CH TRU	Hanford/WIPP	9.9×10^{-4}
	INEL/WIPP	7.9×10^{-4}
	SRS/WIPP	4.2×10^{-2}
RH TRU	Hanford/WIPP	4.3×10^{-5}
	INEL/WIPP	1.6×10^{-3}
	SRS/WIPP	b

^aCH TRU is contact-handled TRU waste; RH TRU is remote-handled TRU waste.

^bThere is no RH TRU at SRS.

Sources: Data are derived from existing NEPA documentation (DOE 1986, 1990a).

The doses to the hypothetical maximally exposed individuals are 4.7 rem to the on-site worker and 0.14 rem to a member of the general public. The corresponding lifetime cancer risks are 0.002 and 6×10^{-5} . These estimated risks are applicable to all technologies. The estimated accident frequency is on the order of 1×10^{-6} per year from shipment of irradiated driver fuels or target assemblies.

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The estimated annual population dose risks from postulated on-site accidents at INEL are (in person-rem) 5.3×10^{-4} for HWR, 5.4×10^{-4} for LWR, and 8.5×10^{-4} for MHTGR. These values correspond to annual health risks of (in cancer fatalities) 2×10^{-7} for HWR and LWR and 3×10^{-7} for MHTGR. Annual genetic effects are (in cases) 1×10^{-8} for HWR and LWR and 4×10^{-8} for MHTGR.

The doses to the hypothetical maximally exposed individuals are the same for all technologies, that is, 21 rem to the on-site worker and 0.01 rem to a member of the general public at the site boundary. The corresponding lifetime cancer risks are 8×10^{-4} and 4×10^{-6} . The estimated accident frequency is on the order of 1×10^{-6} per year for on-site shipping of an LLW liquid waste tank.

TABLE K.14 Estimated Radiological Dose and Health Risk for the Maximally Exposed Individual from On-site Transportation Accidents

Site	Technology	Maximum-Individual Dose ^a (rem)			Maximum-Individual Health Risk (lifetime fatal cancer risk)
		CEDE	Maximum Organ ^b	Thyroid	
Hanford	HWR	4.7	3.5×10^1	1.9×10^{-1}	2×10^{-3}
		1.4×10^{-1}	3.4×10^{-1}	1.3×10^{-1}	6×10^{-5}
	LWR	4.7	3.5×10^1	1.9×10^{-1}	2×10^{-3}
		1.4×10^{-1}	3.4×10^{-1}	1.3×10^{-1}	6×10^{-5}
	MHTGR	4.7	3.5×10^1	1.9×10^{-1}	2×10^{-3}
		1.4×10^{-1}	3.4×10^{-1}	1.3×10^{-1}	6×10^{-5}
INEL	HWR	2.1×10^1	1.8×10^2	2.8×10^{-2}	8×10^{-4}
		1.0×10^{-2}	8.7×10^{-2}	9.1×10^{-4}	4×10^{-6}
	LWR	2.1×10^1	1.8×10^2	2.8×10^{-2}	8×10^{-4}
		1.0×10^{-2}	8.7×10^{-2}	9.1×10^{-4}	4×10^{-6}
	MHTGR	2.1×10^1	1.8×10^2	2.8×10^{-2}	8×10^{-4}
		1.0×10^{-2}	8.7×10^{-2}	9.1×10^{-4}	4×10^{-6}
SRS	HWR	2.6×10^{-1}	2.0	1.1×10^{-2}	1×10^{-4}
		4.3×10^{-2}	1.3×10^{-1}	3.6×10^{-2}	2×10^{-5}
	LWR	2.6×10^{-1}	2.0	1.1×10^{-2}	1×10^{-4}
		4.3×10^{-2}	1.3×10^{-1}	3.6×10^{-2}	2×10^{-5}
	MHTGR	2.6×10^{-1}	2.0	1.1×10^{-2}	1×10^{-4}
		4.3×10^{-2}	1.3×10^{-1}	3.6×10^{-2}	2×10^{-5}

^aThe first value represents the dose to an on-site worker at 100 m from the accident, and the second value represents the dose to a member of the general public at the site boundary.

^bMaximum organ denotes the organ receiving the highest dose.

Savannah River Site

The estimated annual population dose risks from postulated on-site accidents at SRS are (in person-rem) 0.0015 for HWR, 0.091 for LWR, and 0.25 for MHTGR. These values correspond to annual health risks of (in cancer fatalities) 6×10^{-7} for HWR, 4×10^{-5} for LWR, and 1×10^{-4} for MHTGR. Annual genetic effects are (in cases) 4×10^{-7} for HWR, 2×10^{-5} for LWR, and 6×10^{-5} for MHTGR.

The doses to the hypothetical maximally exposed individuals are 0.26 rem to the on-site worker and 0.043 to a member of the general public. The corresponding lifetime cancer risks are 1×10^{-4} and 2×10^{-5} . These estimated risks are applicable to all technologies. The estimated accident frequency is on the order of 1×10^{-5} to 1×10^{-6} per year from shipment of irradiated driver fuels or target assemblies.

TABLE K.15 Estimated Radiological Dose Risk and Health Risk for the General Population from On-site Transportation Accidents

Site	Technology	Population Dose Risk (person-rem/yr)			Population Health Risk (cases/yr)	
		CEDE	Gonads	Thyroid	Cancer Deaths	Genetic Effects
Hanford	HWR	2.5×10^{-3}	7.8×10^{-4}	4.9×10^{-4}	1×10^{-6}	2×10^{-7}
	LWR	1.1×10^{-2}	6.7×10^{-3}	5.6×10^{-3}	4×10^{-6}	2×10^{-6}
	MHTGR	4.5×10^{-3}	1.3×10^{-3}	7.9×10^{-4}	2×10^{-6}	3×10^{-7}
INEL	HWR	5.3×10^{-4}	5.3×10^{-5}	2.4×10^{-5}	2×10^{-7}	1×10^{-8}
	LWR	5.4×10^{-4}	5.5×10^{-5}	1.0×10^{-5}	2×10^{-7}	1×10^{-8}
	MHTGR	8.5×10^{-4}	1.4×10^{-4}	6.8×10^{-5}	3×10^{-7}	4×10^{-8}
SRS	HWR	1.5×10^{-3}	1.5×10^{-3}	1.5×10^{-3}	6×10^{-7}	4×10^{-7}
	LWR	9.1×10^{-2}	9.0×10^{-2}	8.1×10^{-2}	4×10^{-5}	2×10^{-5}
	MHTGR	2.5×10^{-1}	2.4×10^{-1}	2.1×10^{-1}	1×10^{-4}	6×10^{-5}

K.2.3.2 Off-site Accidents

The radiological impacts of off-site transportation accidents were calculated for tritiated heavy water and product tritium and plutonium. The RADTRAN III computer code was used to calculate population dose risks, and the GENII computer code was used to estimate doses to the maximally exposed individual. The analysis was based on the upper-bound core inventory and very conservative assumptions. Doses were calculated for the collective, affected off-site population as well as for the maximally exposed individual. The population doses were weighted for the annual probabilities of occurrence for all accident severity classes to yield the yearly estimated dose risk. For the maximum-individual doses, the consequences of the event were calculated. The calculated doses were multiplied by the health risk factors to estimate the lifetime fatal cancer risks for the maximally exposed individual (Table K.16) and the cancer fatalities and genetic effects for the exposed population (Table K.17). Radiological impacts were also estimated for the off-site shipment of HLW and TRU waste to their respective national repository locations. The resultant impacts presented in Table K.18 are based on risk factors established in existing NEPA documentation.

Hanford Site

Tritium Product. The tritium produced at Hanford would be shipped to SRS for processing. The estimated annual population dose risks are 1.6×10^{-4} person-rem for all technologies. The corresponding annual health risks are 6×10^{-8} cancer fatalities and 4×10^{-8} genetic effects. The estimated maximum-individual dose to a member of the general public is 24 rem, which corresponds to an estimated lifetime fatal cancer risk of 0.01. This estimated

TABLE K.16 Estimated Radiological Dose and Health Risk for the Maximally Exposed Individual from Off-site Transportation Accidents

Material	Origin/ Destination	Technology	Maximum-Individual Dose (rem)			Maximum- Individual Health Risk (lifetime fatal cancer risk)
			CEDE	Maximum Organ ^a	Thyroid	
Tritium	Hanford/SRS INEL/SRS	All	2.2×10^1	2.6×10^1	2.4×10^1	9×10^{-3}
		All	2.2×10^1	2.6×10^1	2.4×10^1	9×10^{-3}
Tritiated heavy water	SRS/Hanford ^b SRS/INEL ^b	HWR	2.4×10^1	2.7×10^1	2.4×10^1	1×10^{-2}
		HWR	2.4×10^1	2.7×10^1	2.4×10^1	1×10^{-2}
Plutonium metal product	Hanford/ Rocky Flats	HWR	2.9×10^{-1}	5.3	2.4×10^{-6}	1×10^{-4}
		LWR	1.0×10^{-1}	1.9	6.0×10^{-7}	4×10^{-5}
		MHTGR	1.3×10^{-1}	2.4	6.5×10^{-7}	5×10^{-5}
	INEL/Rocky Flats	HWR	2.9×10^{-1}	5.3	2.4×10^{-6}	1×10^{-4}
		LWR	1.0×10^{-1}	1.9	6.0×10^{-7}	4×10^{-5}
		MHTGR	1.3×10^{-1}	2.4	6.5×10^{-7}	5×10^{-5}
	SRS/Rocky Flats	HWR	2.9×10^{-1}	5.3	2.4×10^{-6}	1×10^{-4}
		LWR	1.0×10^{-1}	1.9	6.0×10^{-7}	4×10^{-5}
		MHTGR	1.3×10^{-1}	2.4	6.5×10^{-7}	5×10^{-5}

^aMaximum organ denotes the organ receiving the highest dose. The maximum organ would be the small intestine for tritium and tritiated heavy water and bone surfaces for plutonium metal.

^bFor the HWR technology selected at the destination site.

maximum-individual risk applies to all three reactor technologies. The estimated annual probability of occurrence for the hypothetical event used for assessing the maximum-individual dose is about 1×10^{-6} .

Tritiated Heavy Water. Risks from tritiated heavy water would apply only if the HWR technology were adopted for Hanford. In this case, tritiated heavy water would be transported from the existing inventory at SRS to Hanford. The estimated annual dose to the general population would be 0.15 person-rem, which corresponds to an estimated annual health risk of 6×10^{-5} cancer fatalities and 4×10^{-5} genetic effects. The estimated maximum-individual dose is a whole-body dose of 24 rem, which corresponds to an estimated lifetime cancer risk of 0.01. The estimated annual probability of occurrence of the hypothetical event used to calculate the maximum-individual dose (shipment of the entire inventory of heavy water) is about 8×10^{-4} .

TABLE K.17 Estimated Radiological Dose and Health Risk for the General Population from Off-site Transportation Accidents

Material	Origin/ Destination	Technology	Population Dose Risk (person-rem)		Population Health Risk (cases/yr)	
			CEDE	Gonads	Cancer Deaths	Genetic Effects
Tritium	Hanford/SRS INEL/SRS	All	1.6×10^{-4}	1.8×10^{-4}	6×10^{-8}	5×10^{-8}
		All	1.4×10^{-4}	1.6×10^{-4}	6×10^{-8}	4×10^{-8}
Tritiated heavy water	SRS/Hanford ^a SRS/INEL ^a	HWR	1.5×10^{-1}	1.6×10^{-1}	6×10^{-5}	4×10^{-5}
		HWR	1.3×10^{-1}	1.4×10^{-1}	5×10^{-5}	4×10^{-5}
Plutonium metal product	Hanford/ Rocky Flats	HWR	1.2×10^{-6}	2.6×10^{-7}	5×10^{-10}	7×10^{-11}
		LWR	1.2×10^{-6}	2.8×10^{-7}	5×10^{-10}	7×10^{-11}
		MHTGR	1.1×10^{-6}	2.7×10^{-7}	4×10^{-10}	7×10^{-11}
	INEL/Rocky Flats	HWR	7.7×10^{-7}	1.7×10^{-7}	3×10^{-10}	4×10^{-11}
		LWR	7.8×10^{-7}	1.9×10^{-7}	3×10^{-10}	5×10^{-11}
		MHTGR	7.6×10^{-7}	1.8×10^{-7}	3×10^{-10}	5×10^{-11}
	SRS/Rocky Flats	HWR	2.2×10^{-6}	4.8×10^{-7}	9×10^{-10}	1×10^{-10}
		LWR	2.2×10^{-6}	5.1×10^{-7}	9×10^{-10}	1×10^{-10}
		MHTGR	2.1×10^{-6}	5.1×10^{-7}	8×10^{-10}	1×10^{-10}

^aFor the HWR technology selected at the destination site.

Plutonium Metal Product. The plutonium metal product would be shipped to the Rocky Flats Plant for processing. The estimated population doses from an accident during off-site transport are (in person-rem) 1.2×10^{-6} for HWR and LWR and 1.1×10^{-6} for MHTGR. The corresponding annual health risks are (in cancer fatalities) 5×10^{-10} for HWR and LWR and 4×10^{-10} for MHTGR. The genetic effects for all technologies are (in cases) 7×10^{-11} . The estimated maximum-individual doses are (in rem) 0.29 for HWR, 0.10 for LWR, and 0.13 for MHTGR. These doses correspond to estimated lifetime fatal cancer risks of 1×10^{-4} for HWR, 4×10^{-5} for LWR, and 5×10^{-5} for MHTGR. The estimated annual probability of occurrence of the accident used to assess the maximum individual dose is about 3×10^{-7} .

HLW and TRU Waste. The HLW would be shipped to the national repository for permanent disposal. The estimated annual population dose risks from all accidents are (in person-rem) 0.032 for HWR, 0.036 for LWR, and 0.012 for MHTGR. These values correspond to annual health risks of (in cancer fatalities) 1×10^{-5} for HWR and LWR and 5×10^{-5} for MHTGR. Maximum-individual doses from truck accidents are not available in the existing NEPA documentation (DOE 1986), although the dose for a severe rail accident is estimated to be 10.3 rem.

TABLE K.18 Estimated Radiological Risk from Off-site Transportation Accidents Involving HLW and TRU Waste^a

Material	Origin/Destination	Technology	Maximum-Individual Dose (rem)	Maximum-Individual Health Risk (fatalities)	Population Dose Risk, CEDE (person-rem/yr)	Population Health Risk (fatalities/yr)
HLW	Hanford/Repository	HWR	b	-	3.2×10^{-2}	1×10^{-5}
		LWR	b	-	3.6×10^{-2}	1×10^{-5}
		MHTGR	b	-	1.2×10^{-2}	5×10^{-6}
	INEL/Repository	HWR	b	-	4.8×10^{-3}	2×10^{-6}
		LWR	b	-	1.5×10^{-2}	6×10^{-6}
		MHTGR	b	-	5.6×10^{-3}	2×10^{-6}
	SRS/Repository	HWR	b	-	8.0×10^{-3}	3×10^{-6}
		LWR	b	-	1.6×10^{-2}	6×10^{-6}
		MHTGR	b	-	1.6×10^{-2}	6×10^{-6}
TRU waste	Hanford/WIPP	HWR	1.6×10^{-1}	6×10^{-5}	6.2×10^{-3}	2×10^{-6}
		LWR	1.6×10^{-1}	6×10^{-5}	1.7×10^{-2}	7×10^{-6}
		MHTGR	1.6×10^{-1}	6×10^{-5}	9.2×10^{-3}	4×10^{-6}
	INEL/WIPP	HWR	1.8×10^{-1}	7×10^{-5}	1.1×10^{-2}	4×10^{-6}
		LWR	1.8×10^{-1}	7×10^{-5}	3.2×10^{-3}	1×10^{-6}
		MHTGR	1.8×10^{-1}	7×10^{-5}	2.5×10^{-2}	1×10^{-5}
	SRS/WIPP	HWR	1.6×10^{-1}	6×10^{-5}	5.5×10^{-1}	2×10^{-4}
		LWR	1.6×10^{-1}	6×10^{-5}	5.5×10^{-1}	2×10^{-4}
		MHTGR	1.6×10^{-1}	6×10^{-5}	7.1×10^{-1}	3×10^{-4}

^aThe risk values are estimated from data in Tables K.5 and K.13. They include latent fatal cancers only. Genetic effects are not estimated separately in the NEPA documents cited in Table K.13.

^bThe maximum-individual dose for a truck shipment is not estimated in the existing NEPA documentation (DOE 1986).

The TRU waste would be shipped to WIPP for permanent disposal. The estimated annual population dose risks from all accidents are (in person-rem) 0.0062 for HWR, 0.017 for LWR, and 0.0092 for MHTGR. These dose risks correspond to annual health risks of (in cancer fatalities) 2×10^{-6} for HWR, 7×10^{-6} for LWR, and 4×10^{-6} for MHTGR. The estimated maximum-individual dose is 0.16 rem for all technologies. The corresponding accident frequency is estimated to be about 1×10^{-6} /yr. This maximum-individual dose corresponds to an increased lifetime fatal cancer risk of 6×10^{-5} . Genetic effects data are not available in existing NEPA documents for shipments of either waste. They are therefore not presented here.

Idaho National Engineering Laboratory

Tritium Product. The tritium produced at INEL would be shipped to SRS for processing. The estimated annual population dose risks are 1.4×10^{-4} person-rem for all technologies. The corresponding annual health risks are 6×10^{-8} cancer fatalities and 4×10^{-8} genetic effects. The estimated maximum-individual dose to a member of the general public is 22 rem, which corresponds to an estimated lifetime fatal cancer risk of 0.001. This estimated maximum-individual risk applies to all three reactor technologies. The estimated annual probability of occurrence for the hypothetical event used to calculate the maximum-individual dose is 2×10^{-6} or less.

Tritiated Heavy Water. Risks from tritiated heavy water would apply only if the HWR technology were adopted for INEL. In that case, tritiated heavy water would be transported from the existing inventory at SRS to INEL. The estimated annual dose to the general population is 0.13 person-rem, which corresponds to an estimated annual health risk of 5×10^{-5} cancer fatalities and 4×10^{-5} genetic effects. The estimated maximum-individual dose is a whole-body dose of 24 rem, which corresponds to an estimated lifetime cancer risk of 0.01. The estimated annual probability of occurrence of the hypothetical event used to assess the maximum-individual dose (shipment of the entire inventory of heavy water) is about 7×10^{-4} .

Plutonium Metal Product. The plutonium metal product would be shipped to the Rocky Flats Plant for processing. The estimated annual population doses are (in person-rem) 7.7×10^{-7} for HWR, 7.8×10^{-7} for LWR, and 7.6×10^{-7} for MHTGR. The corresponding annual health risks are (in cancer fatalities) 3×10^{-10} for all technologies. The genetic effects are (in cases) 4×10^{-11} for HWR and 5×10^{-11} for LWR and MHTGR. The estimated maximum-individual doses are (in rem) 0.29 for HWR, 0.10 for LWR, and 0.13 for MHTGR. These doses correspond to estimated lifetime fatal cancer risks of 1×10^{-4} for HWR, 4×10^{-5} for LWR, and 5×10^{-5} for MHTGR. The estimated annual probability of occurrence of the accident used to assess the maximum-individual dose is about 2×10^{-7} .

HLW and TRU Waste. The HLW would be shipped to the national repository for permanent disposal. The estimated annual population dose risks from all accidents are (in person-rem) 0.0048 for HWR, 0.015 for LWR, and 0.0056 for MHTGR. These values correspond to annual health risks of (in cancer fatalities) 2×10^{-6} for HWR, 6×10^{-6} for LWR, and 2×10^{-6} for MHTGR. Maximum-individual doses from truck accidents are not available in the existing

NEPA documentation (DOE 1986), although the dose for a severe rail accident is estimated to be 10.3 rem.

The TRU waste would be shipped to WIPP for permanent disposal. The estimated annual population dose risks from all accidents are (in person-rem) 0.011 for HWR, 0.0032 for LWR, and 0.025 for MHTGR. These dose risks correspond to annual health risks of (in cancer fatalities) 4×10^{-6} for HWR, 1×10^{-6} for LWR, and 1×10^{-5} for MHTGR. The estimated maximum-individual dose is 0.18 rem for all technologies. The corresponding accident frequency is estimated to be about 1×10^{-6} /yr. This maximum-individual dose corresponds to an increased lifetime fatal cancer risk of 7×10^{-5} . Genetic effects data are not available in the existing NEPA documentation for shipments of either waste. They are therefore not presented here.

Savannah River Site

Tritium Product. Tritium product from SRS would be processed on-site; therefore, no off-site risks would be involved.

Tritiated Heavy Water. The tritiated heavy-water inventory would be shipped to either Hanford or INEL, should the HWR technology be chosen for either site. The risks of such transport are discussed above for Hanford or INEL and are therefore not included here.

Plutonium Metal Product. The plutonium metal product would be shipped to the Rocky Flats Plant for processing. The estimated annual population doses are (in person-rem) 2.2×10^{-6} for HWR and LWR and 2.1×10^{-6} for MHTGR. The corresponding annual health risks are (in cancer fatalities) 9.0×10^{-10} for HWR and LWR and 8×10^{-10} for MHTGR. The genetic effects are (in cases) 1×10^{-9} for all technologies. The estimated maximum-individual doses are (in rem) 0.29 for HWR, 0.10 for LWR, and 0.13 for MHTGR. These doses correspond to estimated lifetime fatal cancer risks of 1×10^{-4} for HWR, 4×10^{-5} for LWR, and 5×10^{-5} for MHTGR. The estimated annual probability of occurrence of the accident used to assess the maximum-individual dose is about 4×10^{-7} or less.

HLW and TRU Waste. The HLW would be shipped to the national repository for permanent disposal. The estimated annual population dose risks from all accidents are (in person-rem) 8.0×10^{-3} for HWR and 1.6×10^{-2} for LWR and MHTGR. These values correspond to annual health risks of (in cancer fatalities) 3×10^{-6} for HWR, and 6×10^{-6} for LWR and MHTGR. Maximum-individual doses from truck accidents are not available in the existing NEPA documentation (DOE 1986), although the dose for a severe rail accident is estimated to be 10.3 rem.

The TRU waste would be shipped to WIPP for permanent disposal. The estimated annual population dose risks from all accidents are (in person-rem) 0.55 for HWR and LWR and 0.71 for MHTGR. These dose risks correspond to annual health risks of (in cancer fatalities) 2×10^{-4} for HWR and LWR and 3×10^{-4} for MHTGR. The estimated maximum-individual dose is 0.16 rem for all technologies. The corresponding accident frequency is estimated to be about 1×10^{-6} /yr. This maximum-individual dose corresponds to an increased lifetime fatal cancer risk

of 6×10^{-5} . Genetic effects data are not available in the existing NEPA documentation for shipments of either waste. They are therefore not presented here.

K.3 NONRADIOLOGICAL RISKS FROM TRANSPORTATION ACCIDENTS INVOLVING RADIOACTIVE MATERIALS

The nonradiological impacts discussed in this section are associated with transportation accidents involving NPR-related radioactive materials. These hazards are not related to radioactivity or toxic chemicals and are therefore similar to the everyday risks involved in transporting nonradioactive materials. These risks are the fatalities resulting from traffic accidents. The generic risk factors for U.S. highways are presented in Table K.19. In estimating nonradiological risks, it was assumed that the risk factors apply equally for on- and off-site transport. Because the on-site population densities resemble a rural population density, the risk factors derived for the rural zone were used for the on-site analysis.

The estimated nonradiological risks from transporting NPR-related radioactive materials at the three candidate sites are summarized in the following sections.

K.3.1 Hanford Site

For Hanford, the estimated nonradiological accident risks for on-site transportation are (in annual fatalities) 0.003 for HWR and LWR and 0.009 for MHTGR. For off-site transportation, these risks are 0.005 for HWR, 0.04 for LWR, and 0.02 for MHTGR.

K.3.2 Idaho National Engineering Laboratory

For INEL, the estimated nonradiological accident risks for on-site transportation are (in annual fatalities) 4×10^{-4} for HWR and LWR and 0.002 for MHTGR. For off-site transportation, the risk is 0.03 for all technologies.

K.3.3 Savannah River Site

For SRS, the estimated nonradiological accident risks for on-site transportation are (in annual fatalities) 0.005 for HWR, 0.01 for LWR, and 0.006 for MHTGR. For off-site transportation, these risks are 0.009 for HWR and 0.02 for LWR and MHTGR.

TABLE K.19 Nonradiological Unit Risk Factors

Population Zone	Accident Fatalities per Kilometer	
	Truck	Rail
Rural	6.8×10^{-8}	2.8×10^{-8}
Suburban	1.7×10^{-8}	2.8×10^{-8}
Urban	9.6×10^{-9}	2.8×10^{-8}

Source: Cashwell et al. 1986.

K.4 DISCUSSION OF UNCERTAINTIES

The methodology used in the radiological transportation analyses has been demonstrated to be conservative regardless of the computer code used (RADTRAN III or GENII). In addition, the case scenarios used and the input assumptions made were designed to emphasize conservatism whenever there was any doubt about potential uncertainties in the parameters or the approach. This conservatism was uniformly applied in the analyses to all three technologies and for all three candidate sites. Thus, the estimated risk values represent reasonable upper bounds.

Some uncertainties in this bounding approach can be identified. For example, in the incident-free assessment, the impacts are generally dominated by LLW impacts for on-site transportation and by HLW impacts for off-site transportation. The impacts are directly proportional to the estimated waste volumes, which are generally bounded by operations in the plutonium mode. Variations in waste volumes have been shown to deviate between the plutonium and tritium modes by a factor of about 4 for LLW and about 10 for HLW. In addition, variations in on-site practice may also create uncertainties for the analysis. Other factors that might introduce uncertainties into the analysis include vehicle speeds, loading capacities, route population densities, total shipping mileages, and distances between driver crew and waste package.

In the accident analysis, in addition to the uncertainty associated with the estimated waste volumes (discussed above), the largest uncertainty may be associated with the accident statistics (i.e., accident frequency per kilometer traveled) used for the on-site analysis. These accident data are based on a rather scant accident data base or were extrapolated from generic off-site accident data. The derived accident data generally fall in the range of 1×10^{-6} to 1×10^{-7} per kilometer traveled and can deviate by more than a factor 10. Because the candidate sites have effective control over on-site traffic, accident data derived from generic information usually are conservative. Other factors (e.g., accident release fractions) are based on conservative estimates, which is in keeping with currently available data for which uncertainties have yet to be determined.

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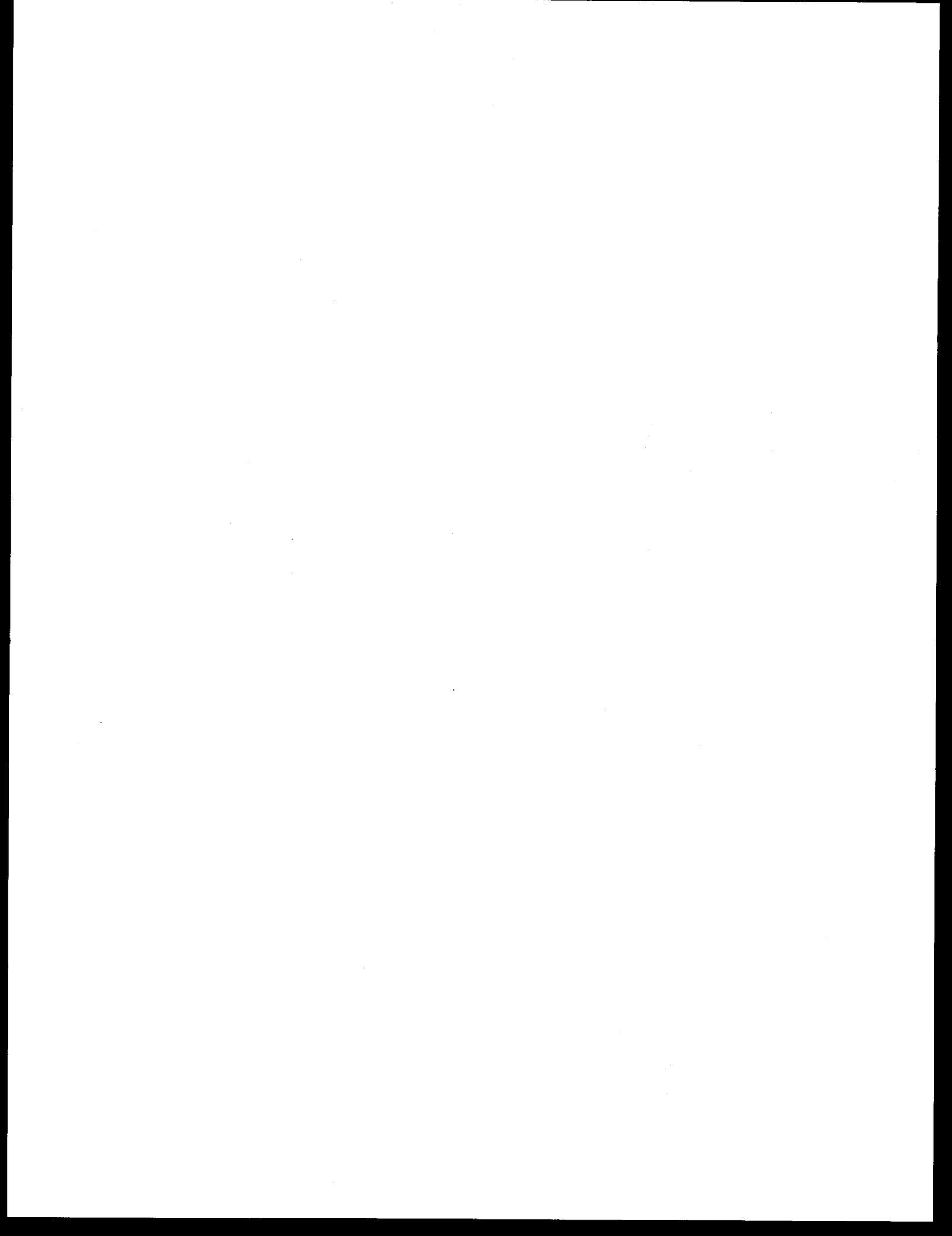
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APPENDIX L:
METHODOLOGY FOR NONRADIOACTIVE HAZARDOUS
MATERIAL ACCIDENT ASSESSMENT

APPENDIX L:

METHODOLOGY FOR NONRADIOACTIVE HAZARDOUS MATERIAL ACCIDENT ASSESSMENT

L.1 INTRODUCTION

The purpose of assessing potential impacts from accidents involving nonradioactive hazardous materials is to characterize the source and magnitude of the human health risks associated with accidental releases of these materials to the atmosphere. Impacts of accidents involving nonradioactive materials are bounded by the effects of chemical spills during storage or transport. These events do not constitute a significant hydrologic threat because of engineered barriers, catchments, and other mitigation contingency plans (Tomasko and Tilbrook 1990).

The storage and use of hazardous materials during NPR operations would be typical of the practices in other chemical or industrial facilities; therefore, the potential impacts from NPR operations would be comparable to those in other U.S. industries. Details of the nonradioactive hazardous material accident assessment methodology are presented in a separate report (NUS 1990).

Because the designs of many process elements for each NPR alternative are not yet finalized, a probabilistic risk assessment is not possible. Thus, a quantitative hazard assessment was conducted to characterize the areal extent of acute health impacts above a given threshold level, as defined in Section L.3. Scenarios developed for this hazard assessment were based on the assumption that a nonradioactive hazardous material storage vessel or transportation container would catastrophically fail, instantaneously releasing its entire contents. Release scenarios for each alternative NPR technology and site combination were evaluated for storage and transportation accidents on the site, as well as for transportation accidents off the site. Because the shipping container size would be the same for a given chemical for either on-site or off-site transportation, the areal extent of the impact zones would be identical. Thus, the impacts projected for transportation accidents apply both to on-site and off-site incidents. Inventories of high-volume nonradioactive hazardous chemicals used in support of the normal operations of each NPR technology are summarized in Tables L.1 through L.9.

L.2 SELECTION OF INDICATOR CHEMICALS

Because numerous hazardous chemicals would be used for the three NPR technologies and associated support facilities, a two-stage screening procedure was applied to reduce the number of chemicals considered in this assessment without affecting the integrity of the analytical results. Preliminary screening of chemicals for further analysis was based on (1) air emissions potential, (2) hazard potential, and (3) storage and/or transport quantities.

The estimate of the air emissions potential of a substance due to a container failure was based on (1) the physicochemical properties of the substance (e.g., physical state, vapor pressure, boiling point), (2) maximum quantity of the chemical stored in a single tank, and (3) conditions (e.g., temperature, pressure) under which the substance would be stored or handled. Liquid releases with low vapor pressures and/or high boiling points have a relatively low air emissions

potential, while gaseous releases have the highest air emissions potential. Hazard potential was assessed using data on acute exposure toxicology, flammability, and explosivity of the chemicals. The potential release quantities of each chemical were assumed to equal the maximum capacities of the on-site storage vessels and transportation vessels used for that chemical.

L.2.1 Preliminary List

During the preliminary screening process, nine chemicals were identified as warranting further evaluation. Those associated with toxic releases were ammonia, chlorine, hydrogen chloride, nitric acid, sulfuric acid, and 1,1,1-trichloroethane; those associated with explosive and/or flammable releases were acetylene, hydrogen, and propylene. Those nine chemicals were further evaluated to determine which would, for this analysis, represent the potential upperbound impacts from catastrophic failure of storage or transportation containment vessels.

Chlorine and ammonia were selected for the final list on the basis of storage quantities, air emissions potential, and toxicological properties. Although nitric acid and sulfuric acid would all be stored on-site and shipped in large quantities, nitric acid and sulfuric acid have normal boiling points well above their storage temperature. Therefore, these chemicals were not selected for the final list.

Chlorinated hydrocarbons were considered because of the potential for phosgene generation during combustion. Only 1,1,1-trichloroethane, with a boiling point of 74°C, would be used in large quantities. In the event of a fire, the high heat would produce a very buoyant plume, and the potential ground-level impacts of phosgene (formed as a combustion by-product) from such an elevated plume would be significantly lower than from dense gas releases of chlorine or ammonia. Therefore, 1,1,1-trichloroethane was not included in the final list of chemicals.

Fires and explosions have the potential to rupture other nearby storage vessels. However, this hazard assessment is limited to an evaluation of the direct human health effects from the initial accident event. Some candidate flammable and explosive releases also have the potential to be dangerous at distances downwind of the release point (i.e., movement of the cloud from the source area before ignition). These specialized cases have been evaluated by comparing dispersion modeling results for various downwind distances with chemical-specific flammable and/or explosive limits.

Acetylene, hydrogen, and propylene were evaluated as potential fire and explosion hazards. Ignition of an instantaneous gas release could result in a fireball, with possible direct flame and thermal radiation impacts, or an explosion, with possible blast-wave impacts. On the basis of the heat of combustion (Lees 1980) and storage/shipping quantities of these gases, hydrogen was selected to represent the fire/explosion hazard because of its higher explosive energy potential.

L.2.2 Final List

The final list of indicator chemicals consists of ammonia, chlorine, and hydrogen. These chemicals are assumed to represent worst-case nonradiological impacts under the scenarios analyzed. The NPR nonradioactive hazardous chemicals inventory is summarized in Table L.10.

TABLE L.1 Chemical Use Information for the Hanford HWR and Support Facilities^a

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Reactor	Sodium hydroxide, 50%	Liquid	T	1,000 gal	Truck	T	1,000 gal
	Chlorine	Liquid	C	1 ton	Truck	C	1 ton
	Hydrogen	Gas	C	16 ft ³	Truck	C	16 ft ³
	Deuterium	Gas	(2,400 psig) C	16 ft ³	Truck	C	16ft ³
	Sodium hypochlorite	Liquid	(2,400 psig) D	15 gal	Truck	D	15 gal
Fuel Fabrication Tritium	Sulfuric acid, 98%	Liquid	T	10,000 gal	Truck	T	10,000 gal
	Nitric acid, 50%	Liquid	D	55 gal	Truck	D	55 gal
	Degreaser	Liquid	D	55 gal	Truck	D	55 gal
	Tin-oil lubricant	Liquid	D _d	55 gal _d	Truck	D _d	55 gal _d
	Tin, powder	Solid			Truck		
Plutonium	Nitric acid, 50%	Liquid	T	20,000 gal	Truck	T	4,000 gal
	Degreaser	Liquid	D	55 gal	Truck	D	55 gal
	Tin-oil lubricant	Liquid	D _d	55 gal _d	Truck	D _d	55 gal _d
	Tin, powder	Solid			Truck		
	Sodium hydroxide, 50%	Liquid	T	20,000 gal	Truck	T	4,000 gal
Target Fabrication ^e Fuel Processing Tritium	Nitric acid	Liquid	T	2,000 gal	Truck	T	1,000 gal
	Fresh, 50%	Liquid	T	1,000 gal	NA	NA	NA
	Recovered, 50%	Liquid	D	55 gal	Truck	D	55 gal
	HAN, 18%	Liquid	D	55 gal	Truck	D	55 gal
	Hydrazine, 35%	Liquid	D	55 gal	Truck	D	55 gal

TABLE L.1 (Cont'd)

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Fuel processing (Cont'd)	Sodium hydroxide, 50%	Liquid	T	5,000 gal	Truck	T	4,000 gal
	Extractant	Liquid	T	1,000 gal	NA	NA	NA
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	NPH	Liquid	D	55 gal	Truck	D	55 gal
	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb
Plutonium	Nitric acid						
	Fresh, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Recovered, 50%	Liquid	T	5,000 gal	NA	NA	NA
	HAN, 18%	Liquid	T	5,400 gal	Truck	T	4,000 gal
	Hydrazine, 35%	Liquid	D	55 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	30,000 gal	Rail	T	16,000 gal
	Extractant	Liquid	5,000 gal	NA	NA	NA	NA
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	NPH	Liquid	D	55 gal	Truck	D	55 gal
	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb
	Sodium nitrite	Solid	B	100 lb	Truck	B	100 lb
	Sodium nitrate	Solid	B	100 lb	Truck	B	100 lb
Target Processing ^f							

^aAbbreviations used in this table include standard abbreviations for units of measure plus the following: ANN = aluminum nitrate nonahydrate; B = bags or boxes; C = cylinders; D = drums; Extractant = process solvent, mix of NPH and tributyl phosphate; HAN = hydroxylamine nitrate; NA = not applicable; NPH = normal paraffin hydrocarbon (similar to n-dodecane); T = tanks.

^bPercentages are weight % in aqueous solutions.

^cRated values (rather than converted metric values) are given for vessel sizes.

TABLE L.1 (Cont'd)

^dAlthough information on the storage and transportation of powdered tin is not available, it is solid and, thus, has a low release potential. In addition, it is not as hazardous as the other materials listed. The impacts for a release of this material would be bounded by an analysis of the other materials in the list.

^eChemicals associated with target fabrication for tritium production in the HWR are combined with the materials for fuel fabrication for the tritium production case. The values presented for fuel fabrication reflect this. Chemicals associated with target fabrication for plutonium production in the HWR are combined with the materials for fuel fabrication for the plutonium production case. The values presented for fuel fabrication reflect this.

^fThe only large-volume materials associated with tritium target processing are nitrogen and argon, both of which are assumed to be nonhazardous if released. Plutonium targets are processed in the same facility as the fuel for plutonium production in the HWR. The materials for plutonium target processing are included in that section.

Sources: Bean 1990 (reactor); Heacock 1990 (support facilities).

TABLE L.2 Chemical Use Information for the Hanford LWR and Support Facilities^a

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Reactor	Ammonia, 28%	Liquid	T	250 gal	Truck	D	55 gal
	Hydrazine, 35%	Liquid	T	250 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	5,500 gal	Truck	T	5,000 gal
	Sulfuric acid, 98%	Liquid	T	270,000 lb	Truck	T ^d	15,000 gal
	Chlorine	Liquid	C ^d	1 ton	Truck	^d	1 ton
	Hydrogen	Gas	(2,400 psig)	24 kg ^d	Truck		32,000 gal ^e
Fuel Fabrication Tritium	Hydrogen	Gas	C	16 ft ³	Truck	C	16 ft ³
			(2,400 psig)				
Target Fabrication							
Tritium	Hydrogen ^f	Gas	C	16 ft ³	Truck	C	16 ft ³
Fuel Processing Tritium	Ammonia	Liquid	T	5,000 gal	Truck	T	4,000 gal
			(114 psig)				
	Nitric acid						
	Fresh, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Recovered, 50%	Liquid	T	5,000 gal	NA	NA	NA
	HAN, 18%	Liquid	T	5,400 gal	Truck	T	4,000 gal
	Hydrazine, 35%	Liquid	D	55 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	20,000 gal	Rail	T	16,000 gal

TABLE L.2 (Cont'd)

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Fuel processing (Cont'd)	ANN, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Extractant, 50%	Liquid	T	1,000 gal	NA	NA	NA
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	Sodium nitrite	Solid	B	100 lb	Truck	B	100 lb
	NPH	Liquid	D	55 gal	Truck	D	55 gal
Plutonium	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb
	Ammonia	Liquid	T	5,000 gal	Truck	T	4,000 gal
	Nitric acid	Liquid	(114 psig)				
	Fresh, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Recovered, 50%	Liquid	T	5,000 gal	NA	NA	NA
	HAN, 18%	Liquid	T	5,400 gal	Truck	T	4,000 gal
	Hydrazine, 35%	Liquid	D	55 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	20,000 gal	Rail	T	16,000 gal
	ANN, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Extractant	Liquid	T	5,000 gal	NA	NA	NA
Target Processing ^g	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	n-Dodecane	Liquid	D	55 gal	Truck	D	55 gal
	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb
	Sodium nitrite	Solid	B	100 lb	Truck	B	100 lb

^aAbbreviations used in this table include standard abbreviations for units of measure plus the following: ANN = aluminum nitrate nonahydrate; C = cylinders; D = drums; Extractant = process solvent, mix of NPH and tributyl phosphate; HAN = hydroxylamine nitrate; NA = not applicable; NPH = normal paraffin hydrocarbon (similar to n-dodecane); sft³ = standard cubic feet; T = tanks.

^bPercentages are weight % in aqueous solutions.

^cRated values (rather than converted metric values) are given for vessel sizes.

TABLE L.2 (Cont'd)

^d2-ft x 20.5-ft seamless cylinders.^e55-ft delivery tube truck.^fFrom common system with fuel fabrication.^gThe only large-volume materials used for tritium target processing are nitrogen and argon, both of which are assumed to be nonhazardous if released. Chemicals associated with plutonium target processing in the LWR are combined with materials for fuel processing for the tritium production case. The values presented for fuel processing reflect this.

Source: Heacock 1990.

TABLE L.3 Chemical Use Information for the Hanford MHTGR and Support Facilities^a

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Reactor	Sulfuric acid	Liquid	T	10,000 gal	Truck	T	4,000 gal
	BETZ 2020 ^d	Liquid	T	7,000 gal	Truck	T	4,000 gal
	Diesel	Liquid	T	15,000 gal	Truck	T	10,000 gal
Fuel Fabrication Tritium	Methyl chloroform	Liquid	Can	2 gal	Truck	Can	2 gal
	Acetylene	Gas	C (250 psig)	20 gal	Truck	C	20 gal
	Propylene	Liquid	C (137 psig)	1,000 gal	Truck	C	200 gal
	Hydrogen	Gas	C (2,400 psig)	64 ft ³	Truck	C	64 ft ³
	Sodium hydroxide, 50% Methyltrichlorosilane Ammonia	Liquid Liquid Liquid	T D T (114 psig)	2,000 gal 55 gal 5,000 gal	Truck Truck Truck	T D T	2,000 gal 55 gal 4,000 gal
Target Fabrication Tritium	Nitric acid, 65% Ammonia	Liquid Liquid	T T (114 psig)	5,000 gal 6,000 gal	Truck Rail	T T	4,000 gal 16,000 gal
	Sodium nitrite Propylene	Solid Liquid	B C (137 psig)	100 lb 5,000 gal	Truck Truck	B T	100 lb 4,000 gal
	Hydrogen	Liquid	C (2400 psig)	64 ft ³	Truck	C	64 ft ³
	Acetylene	Gas	T (250 psig)	10,000 gal	Rail	T	16,000 gal

TABLE L.3 (Cont'd)

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Plutonium	Nitric acid, 65%	Liquid	T (250 psig)	5,000 gal	Truck	T	4,000 gal
	Acetylene	Gas	T (250 psig)	20,000 gal	Rail	T	16,000 gal
	Propylene	Liquid	T (137 psig)	20,000 gal	Rail	T	16,000 gal
	Hydrogen	Liquid	T	500,000 gal	Rail	T	16,000 gal
	Methyl chloroform	Liquid	T	10,000 gal	Truck	T	4,000 gal
	Methyltrichlorosilane	Liquid	T	5,000 gal	Truck	T	4,000 gal
	Ammonia	Liquid	T (114 psig)	30,000 gal	Rail	T	16,000 gal
Fuel Processing Tritium	Tributyl phosphate	Liquid	T	5,000 gal	Truck	T	4,000 gal
	Oxygen	Liquid	T	5,000 gal	Truck	T	4,000 gal
	Nitric acid	Liquid	T	55 gal	Truck	T	55 gal
	Fresh, 50%	Liquid	T	500 gal	NA	NA	NA
	Recovered, 50%	Liquid	D	55 gal	Truck	D	55 gal
	HAN, 18%	Liquid	D	55 gal	Truck	D	55 gal
	Hydrazine, 35%	Liquid	D	55 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	1,000 gal	NA	NA	NA
	Extractant	Liquid	D	55 gal	Truck	D	55 gal
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
Plutonium ^c	NPH	Liquid	D	55 gal	Truck	D	55 gal
	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb

TABLE L.3 (Cont'd)

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Trans- portation Mode	Trans- portation Vessel Type	Trans- portation Vessel Size ^c
Target Processing Tritium ^f							
Plutonium	Oxygen		TBD				
	Nitric acid						
	Fresh, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Recovered, 50%	Liquid	T	5,000 gal	NA	NA	NA
	HAN, 18%	Liquid	T	5,400 gal	Truck	T	4,000 gal
	Hydrazine, 35%	Liquid	D	55 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	20,000 gal	Rail	T	16,000 gal
	Extractant	Liquid	T	5,000 gal	NA	NA	NA
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	NPH	Liquid	D	55 gal	Truck	D	55 gal
	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb

^aAbbreviations used in this table include standard abbreviations for units of measure plus the following: ANN = aluminum nitrate nonahydrate; B = bags or boxes; C = cylinders; D = drums; Extractant = process solvent, mix of NPH and tributyl phosphate; HAN = hydroxylamine nitrate; NA = not applicable; NPH = normal paraffin hydrocarbon (similar to n-dodecane); sft³ = standard cubic feet; sm³ = standard cubic meters; T = tanks; TBD = to be determined.

^bPercentages are weight % in aqueous solutions.

^cRated values (rather than converted metric values) are given for vessel sizes.

^dNontoxic commercial water treatment product made by Betz Laboratory, Inc.

^eFuel processing material requirements included as part of plutonium target processing below.

^fThe only large-volume materials are nitrogen and argon, both of which are assumed to be nonhazardous if released.

Sources: Bulmahn 1990 (reactor); Heacock 1990 (support facilities).

TABLE L.4 Chemical Use Information for the INEL HWR and Support Facilities^a

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Reactor	Sodium hydroxide, 50%	Liquid	T	1,000 gal	Truck	T	1,000 gal
	Chlorine	Liquid	C	1 ton	Truck	C	1 ton
	Hydrogen	Gas	C	16 ft ³	Truck	C	16 ft ³
	Deuterium	Gas	(2,400 psig) C	16 ft	Truck	C	16 ft ³
Fuel Fabrication Tritium	Sodium hypochlorite	Liquid	(2,400 psig) D	15 gal	Truck	D	15 gal
	Sulfuric acid, 98%	Liquid	T	10,000 gal	Truck	T	10,000 gal
	Nitric acid, 50%	Liquid	D	55 gal	Truck	D	55 gal
	Degreaser	Liquid	D	55 gal	Truck	D	55 gal
Plutonium	Tin-oil lubricant	Liquid	D _d	55 gal _d	Truck	D _d	55 gal _d
	Tin, powder	Solid	T	20,000 gal	Truck	T	4,000 gal
	Nitric acid, 50%	Liquid	D	55 gal	Truck	D	55 gal
	Degreaser	Liquid	D _d	55 gal _d	Truck	D _d	55 gal _d
Target Fabrication ^e	Tin-oil lubricant	Liquid	T	20,000 gal	Truck	T	4,000 gal
	Tin, powder	Solid	D	55 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	20,000 gal	Truck	T	4,000 gal
	Nitric acid	Liquid	T	2,000 gal	Truck	T	1,000 gal
Fuel Processing Tritium	Fresh, 50%	Liquid	T	1,000 gal	NA	NA	NA
	Recovered, 50%	Liquid	T	55 gal	Truck	D	55 gal
	HAN, 18%	Liquid	D	55 gal	Truck	D	55 gal
	Hydrazine, 35%	Liquid	D	5,000 gal	Truck	T	4,000 gal
	Sodium hydroxide, 50%	Liquid	T	1,000 gal	NA	NA	NA
	Extractant	Liquid	T	55 gal	Truck	D	55 gal
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	NPH	Liquid	D	55 gal	Truck	D	55 gal

TABLE L.4 (Cont'd)

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Fuel processing (Cont'd)	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb
Plutonium	Nitric acid Fresh, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Recovered, 50% HAN, 18%	Liquid	T	5,000 gal	NA	NA	NA
	Hydrazine, 35%	Liquid	T	5,400 gal	Truck	T	4,000 gal
	Sodium hydroxide, 50%	Liquid	D	55 gal	Truck	D	55 gal
	Extractant	Liquid	T	30,000 gal	Rail	T	16,000 gal
	Tributyl phosphate	Liquid	T	5,000 gal	NA	NA	NA
	NPH	Liquid	D	55 gal	Truck	D	55 gal
	Potassium permanganate	Liquid	D	55 gal	Truck	D	55 gal
	Sodium nitrite	Solid	B	100 lb	Truck	B	100 lb
	Sodium nitrate	Solid	B	100 lb	Truck	B	100 lb
Target Processing ^f							

^aAbbreviations used in this table include standard abbreviations for units of measure plus the following: ANN = aluminum nitrate nonahydrate; B = bags or boxes; C = cylinders; D = drums; Extractant = process solvent, mix of NPH and tributyl phosphate; HAN = hydroxylamine nitrate; NA = not applicable; NPH = normal paraffin hydrocarbon (similar to n-dodecane); sft³ = standard cubic feet; T = tanks.

^bPercentages are weight % in aqueous solutions.

^cRated values (rather than converted metric values) are given for vessel sizes.

^dAlthough information on the storage and transportation of powdered tin is not available, it is solid and, thus, has a low release potential. In addition, it is not as hazardous as the other materials listed. The impacts for a release of this material would be bounded by an analysis of the other materials in the list.

TABLE L.4 (Cont'd)

^eChemicals associated with target fabrication for tritium production in the HWR are combined with the materials for fuel fabrication for the tritium production case. The values presented for fuel fabrication reflect this. Chemicals associated with target fabrication for plutonium production in the HWR are combined with the materials for fuel fabrication for the plutonium production case. The values presented for fuel fabrication reflect this.

^fThe only large-volume materials used for tritium target processing are nitrogen and argon, both of which are assumed to be nonhazardous if released. Plutonium targets are processed in the same facility as the fuel for plutonium production in the HWR. The materials for target processing are included in that section.

Sources: Bean 1990 (reactor); Heacock 1990 (support facilities).

TABLE L.5 Chemical Use Information for the INEL LWR and Support Facilities^a

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Reactor	Ammonia, 28%	Liquid	T	250 gal	Truck	D	55 gal
	Hydrazine, 35%	Liquid	T	250 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	5,500 gal	Truck	T	5,000 gal
	Sulfuric acid, 98%	Liquid	T	270,000 lb	Truck	T ^d	15,000 gal
	Chlorine	Liquid	C ^d	1 ton	Truck	^d	1 ton
	Hydrogen	Gas	(2,400 psig)	24 kg ^d	Truck		32,000 gal ^e
Fuel Fabrication Tritium	Hydrogen	Gas	C	16 ft ³	Truck	C	16 ft ³
Target Fabrication Tritium	Hydrogen ^f	Gas	C	16 ft ³	Truck	C	16 ft ³
Fuel Processing Tritium	Ammonia	Liquid	T	5,000 gal	Truck	T	4,000 gal
	Nitric acid		(114 psig)				
	Fresh, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Recovered, 50%	Liquid	T	5,000 gal	NA	NA	NA
	HAN, 18%	Liquid	T	5,400 gal	Truck	T	4,000 gal
	Hydrazine, 35%	Liquid	D	55 gal	Truck	D	55 gal

TABLE L.5 (Cont'd)

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Fuel processing (Cont'd)	Sodium hydroxide, 50%	Liquid	T	20,000 gal	Rail	T	16,000 gal
	ANN, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Extractant, 50%	Liquid	T	1,000 gal	NA	NA	NA
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	NPH	Liquid	D	55 gal	Truck	D	55 gal
	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb
	Sodium nitrite	Solid	B	100 lb	Truck	B	100 lb
Plutonium	Ammonia	Liquid	T	5,000 gal	Truck	T	4,000 gal
	Nitric acid	Liquid	(114 psig)				
	Fresh, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Recovered, 50%	Liquid	T	5,000 gal	NA	NA	NA
	HAN, 18%	Liquid	T	5,400 gal	Truck	T	4,000 gal
	Hydrazine, 35%	Liquid	D	55 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	20,000 gal	Rail	T	16,000 gal
	ANN, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Extractant	Liquid	T	5,000 gal	NA	NA	NA
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	n-Dodecane	Liquid	D	55 gal	Truck	D	55 gal
	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb
	Sodium nitrite	Solid	B	100 lb	Truck	B	100 lb
Target Processing ^g							

^aAbbreviations used in this table include standard abbreviations for units of measure plus the following: ANN = aluminum nitrate nonahydrate; B = bags or boxes; C = cylinders; D = drums; Extractant = process solvent, mix of NPH and tributyl phosphate; HAN = hydroxylamine nitrate; NA = not applicable; NPH = normal paraffin hydrocarbon (similar to n-dodecane); sft³ = standard cubic feet; T = tanks.

^bPercentages are weight % in aqueous solutions.

TABLE L.5 (Cont'd)

^cRated values (rather than converted metric values) are given for vessel sizes.

^d2-ft x 20.5-ft seamless cylinders.

^e55-ft delivery tube truck.

^fFrom common system with fuel fabrication.

^gThe only large-volume materials used for tritium target processing are nitrogen and argon, both of which are assumed to be nonhazardous if released. Chemicals associated with plutonium target processing in the LWR are combined with materials for fuel processing for the tritium production case. The values presented for fuel processing reflect this.

Source: Heacock 1990.

TABLE L.6 Chemical Use Information for the INEL MHTGR and Support Facilities^a

Process Unit	Chemical	Physical State	Storage Vessel Type	Storage Vessel Size ^b	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^b
Reactor	Sulfuric acid	Liquid	T	10,000 gal	Truck	T	4,000 gal
	BETZ 2020 ^d	Liquid	T	7,000 gal	Truck	T	4,000 gal
	Diesel	Liquid	T	15,000 gal	Truck	T	10,000 gal
Fuel Fabrication Tritium	Acetylene	Gas	C (250 psig)	250 sft ³	Truck	C	250 sft ³
	Methyl chloroform	Liquid	Can	10 kg	Truck	Can	10 kg
	Propylene	Liquid	C (137 psig)	2,500 sft ³	Truck	C	2,500 sft ³
	Hydrogen	Gas	C (2,400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
	Argon	Gas	C (2,400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
	Sodium hydroxide	Liquid	T (2,400 psig)	2,000 gal	Truck	Can	55 gal
Target Fabrication Tritium	Nitrogen	Gas	C (2,400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
	Methyltrichlorosilane	Liquid	D (114 psig)	55 gal	Truck	D	55 gal
	Ammonia	Liquid	T (114 psig)	25 m ³	Truck	T ^c	4,000 gal
	Lithium nitrate	Solid	Can	10 kg	Truck	Can	10 kg
	Nitric acid	Liquid	T (114 psig)	5,000 gal	Truck	T	4,000 gal
	Ammonia	Liquid	T (114 psig)	200 m ³	Truck	T ^c	4,000 gal

TABLE L.6 (Cont'd)

Process Unit	Chemical	Physical State	Storage Vessel Type	Storage Vessel Size ^b	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^b
Target fabrication (Cont'd)	Acetylene	Gas	C (250 psig)	10,000 sft ³	Truck	C	10,000 sft ³
	Propylene	Liquid	T (137 psig)	7,000 gal	Truck	C	4,000 gal
	Hydrogen	Gas	C (2,400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
Plutonium	Acetylene	Gas	T (250 psig)	20,000 gal	Truck	T ^c	4,000 gal
	Methyl chloroform	Liquid	Can	10 kg	Truck	Can	10 kg
	Propylene	Liquid	T (137 psig)	20,000 gal	Truck	T	4,000 gal
	Hydrogen	Gas	T (2,400 psig)	500,000 gal	Truck	T	4,000 gal
	Nitric acid	Liquid	T (250 psig)	5,000 gal	Truck	T	4,000 gal
Fuel Processing Tritium	Methyltrichlorosilane	Liquid	D	55 gal	Truck	D	55 gal
	Ammonia	Liquid	T (114 psig)	30,000 gal	Truck	T ^c	4,000 gal
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	Oxygen	Liquid	T (200 psig)	11,000 gal ^e	Truck	T	5,000 gal
	Nitric acid	Liquid	T	30,000 gal ^c	Truck	T	4,000 gal
	Calcium oxide	Solid	B	100 lb	Truck	B	100 lb
	Cement	Solid	B	100 lb	Truck	B	100 lb
	Decon solutions	Liquid	D	55 gal	Truck	D	55 gal
	Sodium carbonate	Liquid	D	55 gal	Truck	D	55 gal
	Kerosine	Liquid	T	7,000 gal	Truck	T	5,000 gal

TABLE L.6 (Cont'd)

Process Unit	Chemical	Physical State	Storage Vessel Type	Storage Vessel Size ^b	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^b
Fuel processing (Cont'd)	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	n-Dodecane	Liquid	D	55 gal	Truck	D	55 gal
	Ferrous sulfamate	Liquid	D	55 gal	Truck	D	55 gal
	Decon solutions	Liquid	D	55 gal	Truck	D	55 gal
Target Processing Tritium	Cement	Solid	B	100 lb	Truck	B	100 lb
	Nitrogen	Liquid	Cryo. D (2,400 psig)	1 kg	Truck	Cryo. D	1 kg
	Hydrogen	Gas	C (2,400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
	Argon	Gas	C (2,400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
Plutonium	Nitric acid	Liquid	T	30,000 gal ^c	Truck	T	4,000 gal
	Silver nitrate	Liquid	D	55 gal	Truck	D	55 gal
	Oxygen	Liquid	T (200 psig)	11,000 gal ^e	Truck	T	5,000 gal
	Ferrous sulfamate	Liquid	D	55 gal	Truck	D	55 gal
	HAN	Liquid	Bo	5 gal	Truck	Bo	5 gal
	Calcium hydroxide	Liquid	D	55 gal	Truck	D	55 gal
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	Cement	Solid	B	100 lb	Truck	B	100 lb
	Sodium carbonate	Liquid	D	55 gal	Truck	D	55 gal
	Decon solutions	Liquid	D	55 gal	Truck	D	55 gal

TABLE L.6 (Cont'd)

^aAbbreviations used in this table include standard abbreviations for units of measure plus the following: B = bags or boxes; Bo = bottles; C = cylinders; Can = sealed safety canisters; Cryo. D = cryogenic drums; D = drums; HAN = hydroxylamine nitrate sf^3 = standard cubic feet; T = tanks.

^bRated values (rather than converted metric values) are given for vessel sizes.

^cThis assumes use of the existing ICPP storage tanks, which have operating volumes of 30,000 gal and 16,000 gal.

^dNontoxic commercial water treatment product made by Betz Laboratory, Inc.

^eThis assumes use of the two existing ICPP storage vessels, which have operating volumes of 11,000 gal each.

Sources: Bulmahn 1990.

TABLE L.7 Chemical Use Information for the SRS HWR and Support Facilities^a

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Reactor	Sodium hydroxide, 50%	Liquid	T	1,000 gal	Truck	T	1,000 gal
	Chlorine	Liquid	C	1 ton	Truck	C	1 ton
	Hydrogen	Gas	C	16 sft ³	Truck	C	16 ft ³
			(2,400 psig)				
Fuel Fabrication	Deuterium	Gas	C	16 ft ³	Truck	C	16 ft ³
			(2,400 psig)				
	Sodium hypochlorite	Liquid	D	15 gal	Truck	D	15 gal
	Sulfuric acid, 98%	Liquid	T	10,000 gal	Truck	T	10,000 gal
Plutonium	Sulfuric acid	Liquid	D	55 gal	Truck	D	55 gal
	TBD						
Target Fabrication	TBD						
	TBD						
Target Processing	TBD						
	TBD						
Plutonium	Ferrous sulfamate	Liquid	T	TBD	Truck	T	TBD
	Nitric acid, 64%	Liquid	T	10,000 gal	Truck	T	10,000 gal
	Oxalic acid	Solid	B	50 lb	Truck	B	50 lb
	Potassium permanganate	Solid	D	110 lb	Truck	D	110 lb
	Sodium hydroxide	Liquid	T	10,000 gal	Rail	T	15,000 gal
	Sodium nitrate	Solid	B	100 lb	Truck	B	100 lbs
	Sulfuric acid	Liquid	D	55 gal	Truck	D	55 gal
	Tributyl phosphate	Liquid	T	10,000 gal	Rail	T	15,000 gal

TABLE L.7 (Cont'd)

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Fuel Processing	Sodium hydroxide	Liquid	T	10,000 gal	Rail	T	15,000 gal
	Ferrous sulfamate	Liquid	T	TBD	Truck	T	TBD
	Manganese nitrate	Solid	D	55 lb	Truck	D	55 lb
	Nitric acid, 51%	Liquid	T	10,000 gal	Truck	T	10,000 gal
	Oxalic acid	Solid	B	50 lb	Truck	B	50 lb
	Sulfuric acid	Liquid	D	55 gal	Truck	D	55 gal
	Tributyl phosphate	Liquid	T	10,000 gal	Rail	T	15,000 gal
	Potassium permanganate	Solid	D	110 lb	Truck	D	110 lb

^aAbbreviations used in this table include standard abbreviations for units of measure plus the following: B = bags or boxes; C = cylinders; D = drums; sft³ = standard cubic feet; T = tanks; TBD = to be determined.

^bPercentages are weight % in aqueous solutions.

^cRated values (rather than converted metric values) are given for vessel sizes.

Source: Bean 1990.

TABLE L.8 Chemical Use Information for the SRS LWR and Support Facilities^a

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Reactor	Ammonia, 18%	Liquid	T	250 gal	Truck	D	55 gal
	Hydrazine, 35%	Liquid	T	250 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	5,500 gal	Truck	T	5,000 gal
	Sulfuric acid, 98%	Liquid	T	270,000 lb	Truck	T	15,000 gal
	Chlorine	Liquid	C ^d	1 ton	Truck	C ^d	1 ton
	Hydrogen	Gas	(2400 psig)	24 kg	Truck		32,000 gal ^e
Fuel Fabrication Tritium	Hydrogen	Gas	C	16 ft ³	Truck	C	16 ft ³
			(2400 psig)				
Target Fabrication Tritium	Hydrogen ^f	Gas	C	16 ft ³	Truck	C	16 ft ³
Fuel Processing Tritium	Ammonia	Liquid	T	5,000 gal	Truck	T	4,000 gal
			(114 psig)				
	Nitric acid	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Fresh, 50%	Liquid	T	5,000 gal	NA	NA	NA
	Recovered, 50%	Liquid	T	5,400 gal	Truck	T	4,000 gal
	HAN, 18%	Liquid	D	55 gal	Truck	D	55 gal
	Hydrazine, 35%	Liquid	T	20,000 gal	Rail	T	16,000 gal
	Sodium hydroxide, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	ANN, 50%	Liquid	T	1,000 gal	NA	NA	NA
	Extractant, 50%	Liquid	D	55 gal	Truck	D	55 gal
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	NPH	Liquid	D				

TABLE L.8 (Cont'd)

Process Unit	Chemical ^b	Physical State	Storage Vessel Type	Storage Vessel Size ^c	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^c
Fuel processing (Cont'd)	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb
	Sodium nitrite	Solid	B	100 lb	Truck	B	100 lb
	NPH	Liquid	D	55 gal	Truck	D	55 gal
Plutonium	Ammonia	Liquid	T	5,000 gal	Truck	T	4,000 gal
	Nitric acid		(114 psig)				
	Fresh, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Recovered, 50%	Liquid	T	5,000 gal	NA	NA	NA
	HAN, 18%	Liquid	T	5,400 gal	Truck	T	4,000 gal
	Hydrazine, 35%	Liquid	D	55 gal	Truck	D	55 gal
	Sodium hydroxide, 50%	Liquid	T	20,000 gal	Rail	T	16,000 gal
	ANN, 50%	Liquid	T	100,000 gal	Rail	T	16,000 gal
	Extractant	Liquid	T	5,000 gal	NA	NA	NA
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	n-Dodecane	Liquid	D	55 gal	Truck	D	55 gal
	Potassium permanganate	Solid	B	100 lb	Truck	B	100 lb
Target Processing ^g	Sodium nitrite	Solid	B	100 lb	Truck	B	100 lb

^aAbbreviations used in this table include standard abbreviations for units of measure plus the following: ANN = aluminum nitrate nonahydrate; C = cylinders; D = drums; Extractant = process solvent, mix of NPH and tributyl phosphate; HAN = hydroxylamine nitrate; NA = not applicable; NPH = normal paraffin hydrocarbon (similar to n-dodecane); sft³ = standard cubic feet; sm³ = standard cubic meters; T = tanks.

^bPercentages are weight % in aqueous solutions.

^cRated values (rather than converted metric values) are given for vessel sizes.

^d2-ft x 20.5-ft seamless cylinders.

TABLE L.8 (Cont'd)

^e55-ft delivery tube truck.

^fFrom common system with fuel fabrication.

^gThe only large-volume materials used for tritium target processing are nitrogen and argon, both of which are assumed to be nonhazardous if released. Chemicals associated with plutonium target processing in the LWR are combined with materials for fuel processing. The values presented for fuel processing reflect this.

Source: Heacock 1990.

TABLE L.9 Chemical Use Information for the SRS MHTGR and Support Facilities^a

Process Unit	Chemical	Physical State	Storage Vessel Type	Storage Vessel Size ^b	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^b
Reactor	Sulfuric acid	Liquid	T	10,000 gal	Truck	T	4,000 gal
	BETZ 2020 ^c	Liquid	T	7,000 gal	Truck	T	4,000 gal
	Diesel	Liquid	T	15,000 gal	Truck	T	10,000 gal
Fuel Fabrication Tritium	Acetylene	Gas	C (250 psig)	250 sft ³	Truck	C	250 sft ³
	Methyl chloroform	Liquid	Can	10 kg	Truck	Can	10 kg
	Propylene	Liquid	C (137 psig)	2,500 sft ³	Truck	C	2,500 sft ³
	Hydrogen	Gas	C (2,400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
	Argon	Gas	C (2,400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
	Sodium hydroxide	Liquid	T (2,400 psig)	2,000 gal	Truck	Can	5 gal
	Nitrogen	Gas	C (2,400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
Plutonium ^e	Methyltrichlorosilane	Liquid	D (114 psig)	55 gal	Truck	D	55 gal
	Ammonia	Liquid	T	25 m ³	Truck	T ^d	4,000 gal
Target Fabrication Tritium	Lithium nitrate	Solid	Can	10 kg	Truck	Can	10 kg
	Nitric acid	Liquid	T	5,000 gal	Truck	T	4,000 gal
	Ammonia	Liquid	T (114 psig)	200 m ³	Truck	T ^d	4,000 gal
	Acetylene	Gas	C (250 psig)	10,000 sft ³	Truck	C	10,000 sft ³

TABLE L.9 (Cont'd)

Process Unit	Chemical	Physical State	Storage Vessel Type	Storage Vessel Size ^b	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^b
Fuel fabrication (Cont'd)	Propylene	Liquid	T (137 psig)	7,000 gal	Truck	C	4,000 gal
	Hydrogen	Gas	C (2400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
Plutonium	Acetylene	Gas	T (250 psig)	20,000 gal	Truck	T ^d	40,000 gal
	Methyl chloroform	Liquid	Can	10 kg	Truck	Can	10 kg
	Propylene	Liquid	T (137 psig)	20,000 gal	Truck	T	4,000 gal
	Hydrogen	Gas	T (2,400 psig)	500,000 gal	Truck	T	4,000 gal
	Nitric acid	Liquid	T	5,000 gal	Truck	T	4,000 gal
	Methyltrichlorosilane	Liquid	D	55 gal	Truck	D	55 gal
	Ammonia	Liquid	T (114 psig)	30,000 gal	Truck	T ^d	4,000 gal
Fuel Processing Tritium	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	Oxygen	Liquid	T (200 psig)	11,000 gal ^f	Truck	T	5,000 gal
	Nitric acid	Liquid	T	30,000 gal ^d	Truck	T	4,000 gal
	Calcium oxide	Solid	B	100 lb	Truck	B	100 lb
	Cement	Solid	B	100 lb	Truck	B	100 lb
	Decon solutions	Liquid	D	55 gal	Truck	D	55 gal
	Sodium carbonate	Liquid	D	55 gal	Truck	D	55 gal
	Kerosine	Liquid	T	7,000 gal	Truck	T	5,000 gal
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	n-Dodecane	Liquid	D	55 gal	Truck	D	55 gal

TABLE L.9 (Cont'd)

Process Unit	Chemical	Physical State	Storage Vessel Type	Storage Vessel Size ^b	Transportation Mode	Transportation Vessel Type	Transportation Vessel Size ^b
Fuel processing (Cont'd)	Ferrous sulfamate	Liquid	D	55 gal	Truck	D	55 gal
	Decon solutions	Liquid	D	55 gal	Truck	D	55 gal
Target Processing Tritium	Cement	Solid	B	100 lb	Truck	B	100 lb
	Nitrogen	Liquid	Cryo. D (2,400 psig)	1 kg	Truck	Cryo. D	1 kg
	Hydrogen	Gas	C	10,000 sft ³	Truck	C	10,000 sft ³
	Argon	Gas	C (2,400 psig)	10,000 sft ³	Truck	C	10,000 sft ³
Plutonium	Nitric acid	Liquid	T	30,000 gal ^d	Truck	T	4,000 gal
	Silver nitrate	Liquid	D	55 gal	Truck	D	55 gal
	Oxygen	Liquid	T (200 psig)	11,000 gal ^f	Truck	T	5,000 gal
	Ferrous sulfamate	Liquid	D	55 gal	Truck	D	55 gal
	HAN	Liquid	Bo	5 gal	Truck	Bo	5 gal
	Calcium hydroxide	Liquid	D	55 gal	Truck	D	55 gal
	Tributyl phosphate	Liquid	D	55 gal	Truck	D	55 gal
	Cement	Solid	B	100 lb	Truck	B	100 lb
	Sodium carbonate	Liquid	D	55 gal	Truck	D	55 gal
	Decon solutions	Liquid	D	55 gal	Truck	D	55 gal

^aAbbreviations used in this table include standard abbreviations for units of measure plus the following: B = bags or boxes; Bo = bottles; C = cylinders; Can = sealed safety canisters; Cryo. D = cryogenic drums; D = drums; HAN = hydroxylamine nitrate; sft³ = standard cubic feet; T = tanks.

^bRated values (rather than converted metric values) are given for vessel sizes.

^cNontoxic commercial waste-treatment product made by Betz Laboratories, Inc.

TABLE L.9 Chemical Use Information for the SRS MHTGR and Support Facilities^a

^dThis assumes use of the existing ICPP storage tanks, which have operating volumes of 30,000 gal and 16,000 gal.

^eChemicals associated with plutonium fuel fabrication in the MHTGR are the same as for tritium fuel fabrication.

^fThis assumes use of the two existing ICPP storage vessels, which have operating volumes of 11,000 gal each.

Source: Bulmahn 1990.

TABLE L.10 Inventory of the NPR Chemicals of Concern^a

Site/ Process Unit	Chemical	Physical State	HWR			LWR			MHTGR		
			Storage Vessel Size ^b	Storage Vessel Type	Trans. Vessel Size ^b	Trans. Vessel Type	Storage Vessel Size ^b	Storage Vessel Type	Storage Vessel Size ^b	Storage Vessel Type	Trans. Vessel Type
Hanford Reactor	Ammonia	Liquid	-	-	-	-	250 gal	T	55 gal	D	-
	Chlorine	Liquid	1 ton	C	1 ton	C	1 ton	C	1 ton	C	-
	Hydrogen	Gas	16 ft ³	C	16 ft ³	C	24 kg	C	32,000 gal	C	-
Support Facilities	Ammonia	Liquid	-	-	-	-	5,000 gal	T	4,000 gal	T	T
	Chlorine	Liquid	-	-	-	-	-	-	-	-	-
	Hydrogen	Gas	-	-	-	-	16 ft ³	C	16 ft ³	C	C
INEL Reactor	Ammonia	Liquid	-	-	-	-	-	-	-	-	-
	Chlorine	Liquid	1 ton	C	1 ton	C	250 gal	T	55 gal	D	-
	Hydrogen	Gas	16 ft ³	C	16 ft ³	C	1 ton	C	1 ton	C	-
Support Facilities	Ammonia	Liquid	-	-	-	-	5,000 gal	T	4,000 gal	T	T
	Chlorine	Liquid	-	-	-	-	-	-	-	-	-
	Hydrogen	Gas	-	-	-	-	16 ft ³	C	16 ft ³	C	T
SRS Reactor	Ammonia	Liquid	-	-	-	-	-	-	-	-	-
	Chlorine	Liquid	1 ton	C	1 ton	C	250 gal	T	55 gal	D	-
	Hydrogen	Gas	16 ft ³	C	16 ft ³	C	24 kg	C	1 ton	C	-

TABLE L.10 (Cont'd)

Site/ Process Unit	Chemical	Physical State	HWR			LWR			MHTGR		
			Storage Vessel Size ^b	Storage Vessel Type	Trans. Vessel Size ^b	Trans. Vessel Type	Storage Vessel Size ^b	Storage Vessel Type	Trans. Vessel Size ^b	Trans. Vessel Type	Trans. Vessel Size ^b
Support Facilities	Ammonia	Liquid	-	-	-	-	5,000 gal	T	4,000 gal	T	4,000 gal
	Chlorine	Liquid	-	-	-	-	-	-	-	-	-
	Hydrogen	Gas	-	-	-	-	16 ft ³	C	16 ft ³	C	4,000 gal

*Abbreviations used in this table include standard abbreviations for units of measure plus the following: "-" = chemical not used in the specific NPR technology; C = cylinders; D = drums; T = tanks.

^bRated values (rather than converted metric values) are given for vessel sizes.

Sources: Hanford: Bean 1990 and Heacock 1990 (HWR); Heacock 1990 (LWR); Bulmahn 1990 and Heacock 1990 (MHTGR).

INEL: Bean 1990 and Heacock 1990 (HWR); Heacock 1990 (LWR); Bulmahn 1990 (MHTGR).

SRS: Bean 1990 (HWR); Heacock 1990 (LWR); Bulmahn 1990 (MHTGR).

Ammonia would be transported and stored as a pressurized liquid at ambient temperatures. If the transportation or storage vessel ruptured, the ammonia would initially flash, resulting in an instantaneous puff release. Some of the released material, however, would remain behind as a liquid pool formed by flash cooling of the ammonia to its normal boiling point. Consequently, the instantaneous release would be followed by a more continuous evaporative release. Although ammonia is lighter than air, entrainment of liquid droplets might increase the average cloud density sufficiently to make it denser than air. Thus, there would be the potential to maintain a ground-level cloud. To account for the worst-case entrainment conditions, an in-cloud liquid mass fraction of 0.5 was assumed.

Worst-case dispersion conditions for such a scenario involve a balance between (1) stable atmospheric conditions and low wind speed, which tend to minimize the dispersion rate of the cloud, and (2) somewhat unstable conditions and higher wind speeds, which may move the cloud a greater distance during the limited period of denser-than-air behavior.

Chlorine would be stored as pressurized liquids at ambient temperature. Instantaneous releases of these chemicals would behave similarly to those of ammonia. That is, the chlorine would initially flash, causing an instantaneous puff release followed by evaporation from the remaining cooled liquid pool. Liquid entrainment might be included in the instantaneous release. Unlike ammonia, however, chlorine is denser than air and would exhibit denser-than-air dispersion characteristics whether or not entrainment occurred. Worst-case dispersion conditions would include high atmospheric stability and low wind speeds.

Hydrogen would be stored as a gas at ambient temperature under high pressure or as a liquid under pressure and below the critical point temperature. On release, the gas would expand and cool. The densest initial cloud would be formed if reversible expansion is assumed. Released liquid would flash, and the remaining liquid would quickly vaporize. Because hydrogen is much lighter than air, the cloud would rapidly rise under calm atmospheric conditions. Worst-case dispersion conditions would include high wind speeds and unstable atmospheric conditions. These meteorological conditions would tend to extend the range within which high ground-level concentrations of hydrogen could occur.

L.3 HEALTH EFFECTS CRITERIA

Potential impacts were evaluated by comparing dispersion modeling results with standard published data. Emergency Response Planning Guidelines (ERPGs) developed by the American Industrial Hygiene Association (AIHA 1989) were used to evaluate the impacts associated with inhalation of ammonia and chlorine. These guidelines are "... intended for use as planning tools for assessing the adequacy of accident prevention and containment measures undertaken for chemical releases, for transportation emergency planning, and for developing community emergency response plans." The ERPGs are estimates of the maximum airborne concentrations above which nearly all individuals exposed for up to 1 hour could experience defined effects. Three levels of ERPGs are used: (1) ERPG-1 defines the level above which individuals could experience mild, transient adverse health effects or perceive a clearly defined objectionable odor; (2) ERPG-2 defines the level above which individuals could experience or develop irreversible or other serious health effects or symptoms that could impair an individual's ability to take protective action; and (3) ERPG-3 defines the level above which individuals could experience or develop life-threatening health effects. The ERPGs for chlorine and ammonia are given in AIHA 1988a and 1988b, respectively. The project impacts exceeding the ERPG-2 and

ERPG-3 levels are considered significant impacts in the discussion given in Sections L.4 and L.5; and Sections 5.2.7, 5.3.7, and 5.4.7.

For hydrogen, the CHEMS-PLUS hazard assessment code (A.D. Little 1988a,b) was used to evaluate thermal radiation (heat) and explosive overpressure hazards. A large release of hydrogen may burn in the form of a fireball. The resultant thermal radiation impacts would depend on the size of the fireball, the distance to the observer, and the observer's orientation. The model assumes that the burn severity would depend on the amount of energy absorbed by the skin after the surface temperature reaches 55°C. The selected criteria represent two levels of thermal radiation: (1) a heat flux in excess of 160 kJ/m² -- fatal for humans due to irreversible skin tissue damage -- and (2) a lesser level of 40 kJ/m² -- pain or mild second-degree burns (A.D. Little 1988a).

Assessment of overpressure impacts from an unconfined vapor cloud explosion are based on a TNT-equivalent model. It is possible to relate the energy released as a TNT-equivalent charge to overpressure at various distances from the explosion. This is accomplished by comparing the combustion energy per unit mass of a vapor cloud with that of TNT, taking into consideration that only 2%-10% of the cloud will be involved. The overpressure data from measurements of TNT explosions then are used to relate overpressure to the distance from the explosion.

The CHEMS-PLUS model was designed for emergency planning purposes. In most cases, the model results will overestimate hazards, but occasional exceptions are possible (A.D. Little 1988a). Three criteria levels have been selected for explosive overpressure impacts: (1) 15.5-29.0 psig -- fatalities among exposed population due to direct blast effects, (2) 2.4-12.2 psig -- eardrum rupture among exposed populations, and (3) greater than 0.3 psig -- some damage to house ceilings and 10% glass breakage (A.D. Little 1988a).

The health effects criteria are summarized in Table L.11. The fire and explosion hazard criteria are summarized in Table L.12.

L.4 MODELING APPROACH

Estimates of air emission rates from catastrophic failure of a storage or transportation vessel were based on guidance from the SPILLS model (Fleisher 1980) and EPA for determining initial source characterization parameters, such as initial cloud radius and height (EPA 1988). The SLAB model (Ermak 1988, 1989) was used for the dispersion analysis. This model simulates gas dispersion of a flash-off source term and pool evaporation. The model can account for the effects of entrainment of liquid droplets, slumping of a dense gas cloud, subsequent cloud heating by the ground and air, entrainment of air at the cloud boundary, and transition to a passive plume. In addition, the model allows input of constant wind speed and stability parameters, and it provides estimates of downwind concentrations as a function of distance. The capability of the atmosphere to disperse pollutants depends, in part, on wind speed and the thermal stability of the atmosphere. Stability classes, ranging from A (extremely unstable) through F (moderately stable) are an indication of the turbulent nature of the atmosphere. Extremely unstable conditions (A stability) enhance diffusion (generally reducing concentrations), while moderately stable conditions (F stability) inhibit diffusion and pollutant dispersion. Also, pollutant dispersion is generally enhanced with increasing wind speed. Thus, worst-case

TABLE L.11 Health Effects Criteria

Level	Effects	Concentration (ppm) above which Defined Effects Could Occur ^a	
		Ammonia	Chlorine
ERPG-1	Odor perception	25	1
ERPG-2	Serious or irreversible effects	200	3
ERPG-3	Life-threatening effects	1,000	20

^aFor nearly all individuals exposed for up to 1 hour.

Sources: AIHA 1988a-b.

meteorological conditions would include low wind speeds and a slightly to moderately stable atmosphere.

Because the dispersion characteristics of each chemical analyzed are quite varied, conditions producing worst-case ground-level concentrations could be different for each chemical and receptor. Thus, the results from the analysis of accidents involving nonradioactive hazardous materials summarized in Sections 5.2.7, 5.3.7, and 5.4.7 represent an *envelope* of impacts under a variety of conditions.

Dispersion modeling of ammonia indicates maximum ground-level concentrations would occur under conditions of maximum liquid droplet entrainment, wind speeds between 1 and 3 m/s, and E to F atmospheric stability. Modeling of chlorine releases, however, indicates that entrainment does not increase the impact distances. The estimated impact distances correspond to a puff of flashed vapor followed by pool evaporation with a low wind speed (1 m/s) and F-stability conditions.

Calculations were performed out to 100 km to ensure identification of the maximum distance at which the exposure criteria could be exceeded. The maximum distance that any of the exposure criteria were exceeded was 24 km for ammonia and 14 km for chlorine. For ammonia and chlorine, the corresponding travel times to reach these distances ranged from 0.5 to 4 hours. Since the SLAB model assumes constant meteorological conditions throughout the model run, the likelihood that these conditions would occur and the extent of their

TABLE L.12 Fireball and Explosive Hazard Criteria

Hazard	Criterion
Thermal radiation	
Injuries	40 kJ/m ²
Fatalities	160 kJ/m ²
Explosive overpressure	
Some building damage	>0.3 psig
Injuries	2.4-12.2 psig
Fatalities	15.5-29.0 psig

Sources: Thermal radiation, A.D. Little 1988a; overpressure, Lees 1980.

Conversion factor: 1 psig = 703.1 kg/m².

persistence were investigated. Meteorological data from the three sites indicate that the modeled conditions occur about 15%-35% of the time, depending on the site. The meteorological data also indicate that these conditions can, with one exception, persist long enough for the toxic plume to reach the maximum distance where exposure criteria would be exceeded.

Travel times required for toxic plumes to reach their critical impact radii for on-site storage and transportation accidents resulting in ammonia releases would range from 1/2 to 7 hours. The conditions modeled for the ammonia release (1 m/s and F stability) were the same as those for the chlorine release. At all three sites these meteorological conditions have been observed to persist for up to 8 hours or more.

Hydrogen releases present a special modeling problem. With large instantaneous releases of explosive gases, high concentrations can occur at large distances from the release point (Bodurtha 1980). Therefore, the possibility of vapor cloud ignition at some distance downwind of the release must be considered. This is especially a problem with heavier-than-air gases, which tend to stratify near the ground and undergo less dispersion. Light gases, such as hydrogen, rise in the atmosphere because of their buoyancy and do not produce large ground-level accumulations of gas at any appreciable distances downwind. Thus, hydrogen vapor cloud ignition may only be possible near the release point.

Dispersion modeling of hydrogen releases showed a rapid rise of the plume. Even at very high wind speeds (20 m/s) and C stability conditions, the positions of highest concentrations rose to more than 200 m within 600 m of the source. In addition, the highest concentrations at the center of the plume were less than 1% at 50 m from the source. Under the conditions modeled, no concentration exceeding the explosive limit were indicated beyond the immediate vicinity of the release point (within 0.1 km).

L.5 SUMMARY RESULTS

The results of the nonradiological hazardous materials accident assessment are summarized in Tables L.13 through L.21. These results indicate that ammonia releases associated with the MHTGR technology represent the greatest hazard at all three sites for both on-site storage and transportation accidents. Significant effects are predicted out as far as 10 km from the point of release.

On a site-by-site basis, potential chemical releases would significantly affect the fewest people at INEL, where the maximum number of individuals exposed to severe health impacts would range from 0 to 2, depending on the wind direction. For Hanford, from 0 to 20 individuals could be affected, and at SRS, from 0 to 40 could be affected.

In summary, the MHTGR technology at the SRS site appears to pose the greatest potential hazard, while the INEL site would provide the greatest protection to the off-site population for any of the technologies.

TABLE L.13 Acute Inhalation Criteria Impact Radii (km) for NPR Accidents at Hanford Involving Nonradioactive Hazardous Materials

Chemicals/ERPG Levels ^a	HWR		LWR		MHTGR	
	On-site Storage	Transportation	On-site Storage	Transportation	On-site Storage	Transportation
Ammonia						
ERPG-1 (25 ppm)	NU ^b	NU	6.8	6.1	23.6	15.4
ERPG-2 (200 ppm)	NU	NU	2.7	2.3	10.3	6.6
ERPG-3 (1,000 ppm)	NU	NU	1.3	1.1	3.6	2.6
Chlorine						
ERPG-1 (1 ppm)	13.8	13.8	13.8	13.8	NU	NU
ERPG-2 (3 ppm)	7.7	7.7	7.7	7.7	NU	NU
ERPG-3 (20 ppm)	2.6	2.6	2.6	2.6	NU	NU

^aERPG-1 = odor perception; ERPG-2 = serious effects; ERPG-3 = life-threatening effects.

^bNU = not used; chemical not used in the technology, thus no impact radii are estimated.

TABLE L.14 Potential Population Exposures for NPR On-site Storage Accidents at Hanford Involving Nonradioactive Hazardous Materials

Chemicals/ERPG Levels ^a	Population Size		
	HWR	LWR	MHTGR
Ammonia			
ERPG-1 (25 ppm)	NU ^b	0-20	40-24,000
ERPG-2 (200 ppm)	NU	0	0-20
ERPG-3 (1,000 ppm)	NU	0	0
Chlorine			
ERPG-1 (1 ppm)	0-200	0-200	NU
ERPG-2 (3 ppm)	0-20	0-20	NU
ERPG-3 (20 ppm)	0	0	NU

^aERPG-1 = odor perception; ERPG-2 = serious effects; ERPG-3 = life-threatening effects.

^bNU = not used; chemical not used in the technology, thus no impacts are estimated.

TABLE L.15 Thermal Radiation and Explosive Overpressure Maximum Impact Radii (km) for NPR Accidents at Hanford Involving Hydrogen Releases

Impact	HWR		LWR		MHTGR	
	On-site Storage	Transportation	On-site Storage	Transportation	On-site Storage	Transportation
Thermal radiation						
Injuries	<0.1	<0.1	<0.1	0.1	1.0	0.2
Fatalities	<0.1	<0.1	<0.1	<0.1	0.5	<0.1
Explosive overpressure						
Some building damage	<0.1	<0.1	0.1	0.4	1.9	0.6
Eardrum rupture and storage vessel damage	<0.1	<0.1	<0.1	0.1	0.4	0.1
Fatalities	<0.1	<0.1	<0.1	<0.1	0.1	<0.1

TABLE L.16 Acute Inhalation Maximum Impact Radii (km) for NPR Accidents at INEL Involving Nonradioactive Hazardous Materials

Chemicals/ERPG Levels ^a	HWR		LWR		MHTGR	
	On-site Storage	Transportation	On-site Storage	Transportation	On-site Storage	Transportation
Ammonia						
ERPG-1 (25 ppm)	NU ^b	NU	6.8	6.1	23.6	6.1
ERPG-2 (200 ppm)	NU	NU	2.7	2.3	10.3	2.3
ERPG-3 (1,000 ppm)	NU	NU	1.3	1.1	3.6	1.1
Chlorine						
ERPG-1 (1 ppm)	13.8	13.8	13.8	13.8	NU	NU
ERPG-2 (3 ppm)	7.7	7.7	7.7	7.7	NU	NU
ERPG-3 (20 ppm)	2.6	2.6	2.6	2.6	NU	NU

^aERPG-1 = odor perception; ERPG-2 = serious effects; ERPG-3 = life-threatening, effects.^bNU = not used; chemical not used in the technology, thus no impact radii are estimated.

**TABLE L.17 Potential Population Exposures for
NPR On-site Storage Accidents at INEL Involving
Nonradioactive Hazardous Materials**

Chemicals/ERPG Levels ^a	Population Size		
	HWR	LWR	MHTGR
Ammonia			
ERPG-1 (25 ppm)	NU ^b	0	0-150
ERPG-2 (200 ppm)	NU	0	0-2
ERPG-3 (1,000 ppm)	NU	0	0
Chlorine			
ERPG-1 (1 ppm)	0-10	0-10	NU
ERPG-2 (3 ppm)	0-10	0-10	NU
ERPG-3 (20 ppm)	0	0	NU

^aERPG-1 = odor perception; ERPG-2 = serious effects;
ERPG-3 = life-threatening effects.

^bNU = not used; chemical not used in the tech-
nology, thus no impacts are estimated.

**TABLE L.18 Thermal Radiation and Explosive Overpressure Maximum Impact Radii (km)
for NPR Accidents at INEL Involving Hydrogen Releases**

Impact	HWR		LWR		MHTGR	
	On-site Storage	Trans- porta- tion	On-site Storage	Trans- porta- tion	On-site Storage	Trans- porta- tion
Thermal radiation						
Injuries	<0.1	<0.1	<0.1	0.1	1.0	<0.1
Fatalities	<0.1	<0.1	<0.1	<0.1	0.5	<0.1
Explosive overpressure						
Some building damage	<0.1	<0.1	0.1	0.4	1.9	0.3
Eardrum rupture and storage vessel damage	0.1	0.1	<0.1	0.1	0.4	<0.1
Fatalities	<0.1	<0.1	<0.1	<0.1	0.1	<0.1

TABLE L.19 Acute Inhalation Maximum Impact Radii (km) for NPR Accidents at SRS Involving Nonradioactive Hazardous Materials

Chemicals/ERPG Levels ^a	HWR		LWR		MHTGR	
	On-site Storage	Transportation	On-site Storage	Transportation	On-site Storage	Transportation
Ammonia						
ERPG-1 (25 ppm)	NU ^b	NU	6.8	6.1	23.6	11.9
ERPG-2 (200 ppm)	NU	NU	2.7	2.3	10.3	3.0
ERPG-3 (1,000 ppm)	NU	NU	1.3	1.1	3.6	1.1
Chlorine						
ERPG-1 (1 ppm)	13.8	13.8	13.8	13.8	NU	NU
ERPG-2 (3 ppm)	7.7	7.7	7.7	7.7	NU	NU
ERPG-3 (20 ppm)	2.6	2.6	2.6	2.6	NU	NU

^aERPG-1 = odor perception; ERPG-2 = serious effects; ERPG-3 = life-threatening effects.

^bNU = not used; chemical not used in the technology, thus no impact radii are estimated.

TABLE L.20 Potential Population Exposures for NPR On-site Storage Accidents at SRS Involving Nonradioactive Hazardous Materials

Chemicals/ERPG Levels ^a	Population Size		
	HWR	LWR	MHTGR
Ammonia			
ERPG-1 (25 ppm)	NU ^b	0	300-7,000
ERPG-2 (200 ppm)	NU	0	0-40
ERPG-3 (1,000 ppm)	NU	0	0
Chlorine			
ERPG-1 (1 ppm)	0-300	0-300	NU
ERPG-2 (3 ppm)	0	0	NU
ERPG-3 (20 ppm)	0	0	NU

^aERPG-1 = odor perception; ERPG-2 = serious effects; ERPG-3 = life-threatening effects.

^bNU = not used; chemical not used in the technology, thus no impacts are estimated.

**TABLE L.21 Thermal Radiation and Explosive Overpressure Maximum Impact Radii (km)
for NPR Accidents at SRS Involving Hydrogen Releases**

Impact	HWR		LWR		MHTGR	
	On-site Storage	Trans- porta- tion	On-site Storage	Trans- porta- tion	On-site Storage	Trans- porta- tion
Thermal radiation						
Injuries	<0.1	<0.1	<0.1	<0.1	1.0	<0.1
Fatalities	<0.1	<0.1	<0.1	<0.1	0.5	<0.1
Explosive overpressure						
Some building damage	<0.1	<0.1	0.1	0.4	1.9	0.3
Eardrum rupture and storage vessel damage	<0.1	<0.1	<0.1	0.1	0.4	<0.1
Fatalities	<0.1	<0.1	<0.1	<0.1	0.1	<0.1

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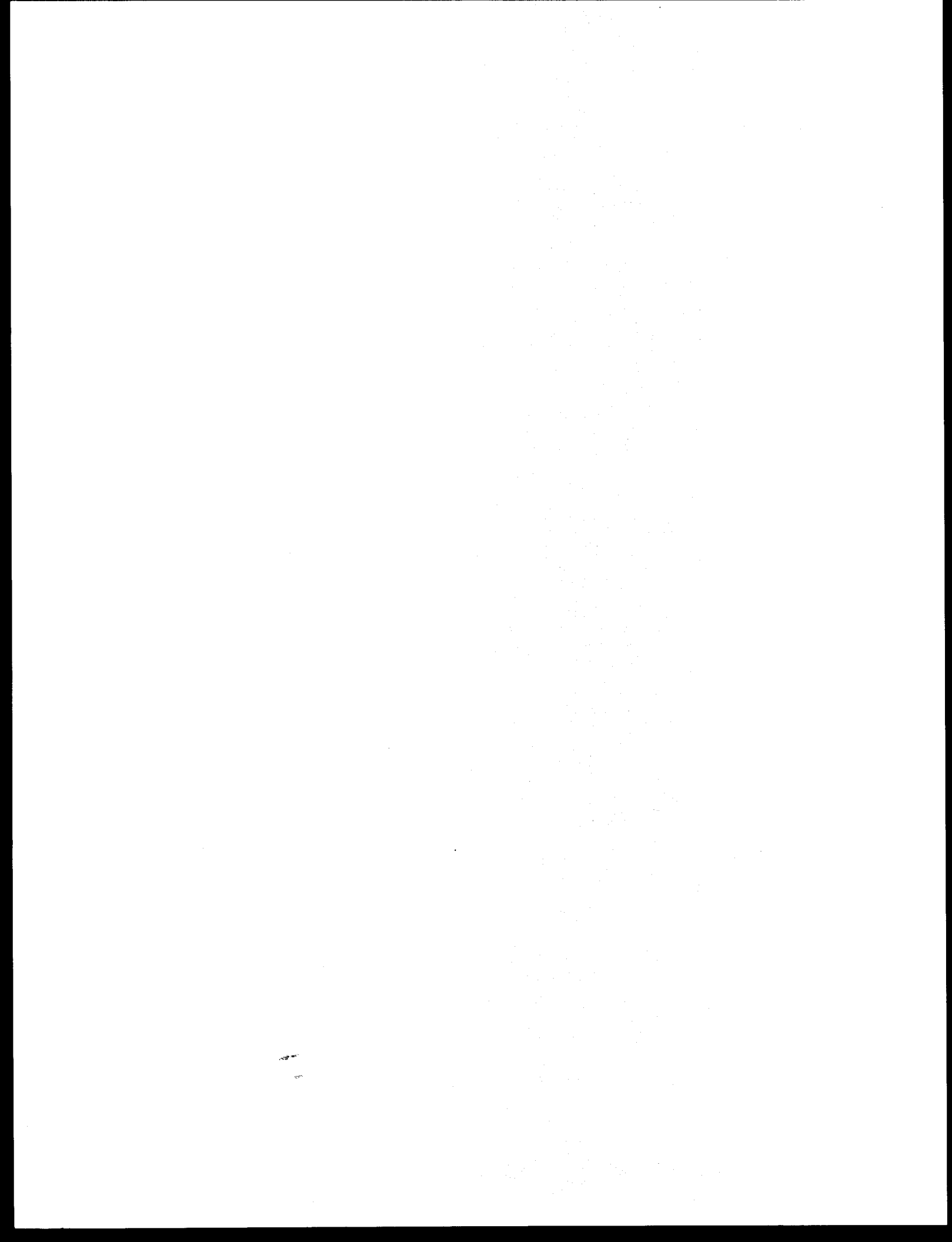
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APPENDIX M:
WASTE MANAGEMENT



APPENDIX M: WASTE MANAGEMENT

M.1 WASTE TYPES

The DOE conducts its environmental management activities at Hanford, INEL, SRS, and other DOE sites pursuant to compliance agreements with state and Federal regulatory authorities. These agreements guide DOE activities at the sites under applicable environmental laws, regulations, and other standards. Compliance with the terms of these negotiated agreements is one of the highest of DOE's priorities.

The DOE's NPR operations will be conducted consistent with commitments DOE has made and will make in these agreements. If it should become necessary, DOE will work with the regulators to incorporate NPR needs into existing agreements or to develop new agreements to ensure continued compliance. Under no circumstances will DOE's performance pursuant to any existing compliance agreement be compromised or diminished as a result of the NPR project. If there are discrepancies between the waste stream volume and waste treatment capacity figures in the DEIS and in existing compliance agreements, or those currently in negotiation, the figures in the compliance agreements will prevail.

The quantities and types of wastes to be managed depend upon whether or not spent fuel is reprocessed and whether or not plutonium is produced. Estimates in this appendix were prepared on the basis that it is reasonably foreseeable that spent fuel could be reprocessed and plutonium produced. If it is decided not to continue those activities, the impacts upon waste management systems from NPR operations would be less than shown here.

Wastes would be generated in gaseous, liquid, and solid states. Appendix M describes management of liquid and solid wastes only. Processing of gaseous wastes is described in Appendix A for the heavy-water reactor (HWR), in Appendix B for the light-water reactor (LWR), and in Appendix C for the modular high-temperature gas-cooled reactor. For this EIS, the term "waste type" applies to one of six basic categories of waste, each of which can be distinguished by (1) the specific chemical and/or physical characteristics of the material and (2) the guidelines that regulate its treatment and disposal. The term "waste stream" simply describes an identified source of waste. It does not pertain to any particular waste category.

Six primary waste types are generated during the operation of nuclear fission reactors and their support facilities. As defined by DOE and EPA, five of these are generated by nuclear material fabrication and processing operations: high-level radioactive waste (HLW), transuranic (TRU) waste, low-level radioactive waste (LLW), mixed waste, and hazardous waste. For this EIS, nonradioactive, nonhazardous waste that does not fall under any of the above five categories is referred to as "other waste." Examples are trash, empty containers, wastewaters, and sewage liquids and solids.

HLW results from the processing of spent nuclear fuel and irradiated target elements.* It contains primarily beta- and gamma-emitting fission products and smaller concentrations of TRU elements (DOE Order 5820.2A, *Radioactive Waste Management*). Converting liquid HLW into a disposable solid requires several processing steps. First, the fission products in the liquids extracted during fuel and target processing must be concentrated in a solid matrix by chemical precipitation and/or evaporation. This solid is then dispersed in a glasslike ceramic matrix that is then formed into a slug (often called a "log") whose volume is approximately 0.5 m³. Processing the waste into this solid matrix is known as vitrification. The purpose of vitrification is to immobilize leachable, highly radioactive species in a waste that is highly resistant to chemical and physical breakdown. Each of the vitrification processes developed at the three candidate sites uses a slightly different technology. Although the composition and volume of the vitrified waste products for a given radioactive waste input differ, the resultant vitrified solids effectively immobilize the HLW.

The vitrified HLW would be disposed of at a Federal repository that would be operative during the active life of the NPR project and that would have sufficient capacity to contain all of the HLW produced. The Federal repository is discussed in detail in two documents: *Mission Plan for the Civilian Radioactive Waste Management Program, Volume 1* (DOE 1985) and the *Environmental Restoration and Waste Management, Five-Year Plan, Fiscal Years 1992-1996* (DOE 1990b).

TRU waste is any solid waste contaminated with alpha-emitting TRU radionuclides (i.e., those with atomic numbers larger than 92) with half-lives longer than 20 years and activities greater than 100 nCi/g at the time of assay (DOE Order 5820.2A). During the processing of spent nuclear fuel, a large portion of the TRU radionuclides generated during fission is dissolved in the liquid HLW stream and becomes incorporated into the final vitrified HLW. Waste containing TRU radionuclides that is processed in this way is actually disposed of in the HLW stream and therefore is not considered as TRU waste for this EIS.

Solid TRU waste typically consists of contaminated materials (e.g., filters, clothing, storage vessels, tools, and other objects) that have come into close contact with waste streams containing TRU radionuclides. The NPR-related TRU waste would be characterized and packaged at each site. It would also be certified at the site as complying with waste acceptance criteria prior to shipment off-site for disposal at the Waste Isolation Pilot Plant (WIPP). This Federal facility is located near Carlsbad, New Mexico, and is designed to store both contact- and remote-handled TRU waste. Waste from ten facilities (including that from the three candidate NPR sites) that have manufactured or processed nuclear materials is destined for disposal at WIPP (DOE 1980; DOE 1990a). The storage capacities at WIPP are 176,000 m³ for contact-handled TRU waste and 7,100 m³ for remote-handled TRU waste. The non-NPR input of TRU waste from the 10 facilities to WIPP through 2013 is projected to be 159,000 m³ of contact-handled waste and 2,700 m³ remote-handled waste. These projections result in unused volumes

*For the purpose of this EIS, it is assumed that spent fuel would ultimately be disposed of in a Federal repository. Spent fuel processing is analyzed in this EIS as part of the original mission of NPR, which included the production of virgin plutonium-239. However, plutonium production is no longer part of the NPR proposed action. The Waste Management PEIS will assess the impacts of wastes from reprocessing of spent fuel from DOE reactors.

of 17,000 m³ and 4,100 m³ for contact- and remote-handled TRU waste, respectively. These unused volumes would be available for NPR-related TRU waste.

The quantities of TRU waste generated by NPR operations at the three candidate sites might vary, depending on such factors as packaging requirement, waste processing protocols, and facility housekeeping procedures. Such factors would be important because the amount of TRU waste generated would be directly related to the number and size of objects (e.g., receptacles, tools, and clothing) that come into contact with TRU radionuclides and thus become contaminated. This aspect also applies to the generation of some solid LLW streams.

For this EIS, NPR-generated TRU waste is not differentiated into contact- and remote-handled waste because of the difficulty in projecting the activities of TRU waste streams until the waste is actually packaged for disposal.

LLW is radioactive waste that cannot be classified as HLW, TRU waste, spent nuclear fuel, or by-product as defined in Section 11e.(2) of the Atomic Energy Act of 1954. It may consist of wash fluids, filters contaminated with radionuclides, equipment or clothing contaminated with radionuclides, or anything that may have become radioactive from contact with radionuclides or nuclear fabrication and fission processes. LLW associated with NPR activities would be disposed of on-site. The waste would be buried in vaults or in other isolated units, depending on the level of radioactivity of the specific waste. Guidelines for storing, treating, and disposing of LLW are detailed in DOE Order 5820.2A.

As with TRU waste, the volume of solid LLW generated by each NPR technology might vary between sites. As previously described for the TRU waste, packaging requirements and facility housekeeping procedures would be important factors affecting the volume of LLW generated. Studies concerning the commercial nuclear power industry indicate a direct correlation between the number of facility workers and the amount of LLW generated (EPRI 1984). Because the number of workers at each NPR facility vary among the sites, LLW generation would likely be different at each site.

Mixed waste is waste that is both "hazardous waste" and "radioactive waste" as previously defined. Currently, mixed waste is stored on-site at Hanford, INEL, and SRS because of the absence of approved treatment methods or off-site disposal facilities. DOE and EPA are responsible for developing guidelines for treating and disposing of mixed waste. The management of mixed waste is described in DOE Order 5400.3.

Hazardous waste is waste that is "hazardous" according to EPA regulations under RCRA (40 CFR 261.3) or regulations of the state in which the wastes are generated or managed. Although hazardous waste can be a liquid, solid, or gas, it is considered as solid waste according to EPA guidelines under Subtitle C of RCRA (40 CFR 260 and 261.2). Waste can be classified as hazardous on the basis of such characteristics as corrosivity, reactivity, ignitability, or toxicity, or by the presence of specific compounds listed as hazardous in 40 CFR 261. Hazardous waste must be disposed of at RCRA-permitted off-site disposal facilities.

Other waste is any nonradioactive, nonhazardous waste. Such waste might include, but is not limited to, solid waste as defined in 40 CFR 261.2 and 261.4. Examples are office trash, packaging and cleaning waste, garbage, and sewage waste. Liquid effluents or solid waste from water treatment or other waste processing facilities and blowdown water from cooling operations might also be considered as other waste. Depending on the specific physical and chemical

characteristics of the waste, other waste is disposed of on-site via landfills, evaporation ponds, ground discharge, or outfall discharge to streams. Appropriate Federal and/or state permits might be required for disposal of other waste.

For this EIS, it is assumed that final disposal of HLW would be at a Federal repository. The capacity of the repository would be sufficient to contain all the HLW generated. It is also assumed that TRU waste would be ultimately disposed of at WIPP. Waste management for NPR activities would be based on current RCRA guidelines and applicable DOE orders. Changes in Federal or state waste management protocols might require modifications to the proposed management of NPR-related waste.

Table M.1 lists the abbreviations used in the figures presented in this appendix. General overviews of the storage, treatment, and disposal of various waste types are presented in Figures M.1, M.2, and M.3 for Hanford, INEL, and SRS, respectively.

M.2 CONSTRUCTION WASTE

The solid and liquid waste generated during construction of NPR facilities would be similar to the waste produced by the construction of any large industrial facility. The waste produced would be similar as to type and disposal protocol and would be independent of site and NPR technology. No radioactive waste would be produced. For this EIS, any waste generated after completion of NPR facilities construction is considered to be an NPR operations waste.

During construction, personal sanitation would be handled by an outside contractor. Because this contractor would also be responsible for disposing of this sewage, existing site wastewater treatment facilities would not be affected. Solid waste generated during construction, with the exception of excavated rock and soil, would include discarded packaging and construction materials (e.g., plasterboard, brick, wood, and scrap steel) and would be considered as other waste to be disposed of on-site.

Hazardous waste generated during construction would consist of such materials as waste adhesives, oils, cleaning fluids, solvents, and coatings. These and other materials such as flammable solvents, coatings, and corrosives would be appropriately packaged and shipped to RCRA-approved off-site disposal facilities. Nonhazardous waste would be properly stabilized and disposed of in site landfills as other waste.

M.3 OPERATIONS WASTE

This section discusses the management of NPR-related operations waste at each of the three candidate sites. This discussion complements the waste management flow diagrams presented in Figures M.4 through M.57. These flow diagrams describe the management of radioactive liquid waste, nonradioactive liquid waste, and solid waste for both the tritium and plutonium production modes for each NPR technology at each of the proposed sites.

In the following sections, only the maximum estimated volume, or the "upper bound," is presented for each waste source. If the upper-bound waste volume is derived from the

weapons-grade plutonium production mode, the designation (P) is used; if the tritium production mode generates the greater waste volume, the designation (T) is used. The designation (TP) is used if the waste volumes for both production modes are similar. In this way, all estimated waste quantities can be considered the maximum expected for a given waste stream. All the estimates of waste volumes presented in this appendix have been rounded and are therefore approximate.

M.3.1 Hanford Operations Waste

M.3.1.1 Waste Produced by the HWR Technology at Hanford

This section describes the upper-bound waste volumes generated by operating an HWR and associated facilities at Hanford. The effects of such operations on Hanford waste management facilities are discussed in Section 5.2.10. Waste management flow diagrams for both the tritium and plutonium production modes are presented in Figures M.4 through M.9.

High-Level Waste (HWR, Hanford)

The driver-fuel and target-processing operations at the Spent Fuel Processing Complex would generate 91 m³/yr (T) of liquid HLW. This waste would be sent to double-shell tanks in the 200-East Area for interim storage before being processed at the Waste Vitrification Plant into two radioactive waste streams. The liquid waste would be routed to the 242-A evaporator for further processing and disposal as LLW. The solid waste containing the fission products would be processed into 15 m³/yr (T) of glasslike ceramic solids at the Waste Vitrification Plant. The corresponding HLW totals for the plutonium production would be 71 m³/yr of liquid HLW and 10 m³/yr of vitrified solids.

Transuranic Waste (HWR, Hanford)

The Spent Fuel Processing Complex would generate 71 m³/yr (P) of TRU waste. An additional 16 m³/yr (P) of TRU waste would be generated during mixed waste sorting operations at the Waste Receiving and Processing Facility. The 87 m³/yr (P) of packaged TRU waste would be certified and stored at the Transuranic Waste Storage and Assay Facility for later off-site disposal at the proposed WIPP facility.

Low-Level Waste (HWR, Hanford)

The HWR facilities would generate two major streams of liquid LLW from NPR operations. The first stream would be 8,100 m³/yr (P) of liquid LLW generated primarily from spent fuel processing at the Spent Fuel Processing Complex. It would be stored in double-shell tanks before being combined with 71 m³/yr (P) of liquid LLW by-product from the Waste Vitrification Plant. The liquid LLW would then be concentrated into a liquid volume of 3,900 m³/yr (P) at the 242-A evaporator in the 200-East Area.

The second liquid LLW stream would be 1,100 m³/yr (P) of tritium-containing process condensate. This waste would be routed directly to storage tanks and combined with the evaporated liquid concentrate from the first stream. This combined liquid waste would be solidified into 6,500 m³/yr (P) of grout at the Grout Treatment Facility and then disposed of in near-surface vaults at the Low-Level Radioactive Waste Disposal Facility.

The 4,300 m³/yr (P) of liquid condensate from the 242-A evaporator would be treated to further remove dissolved low-level radioactive solids. The purification media would be disposed of as LLW. The remaining liquid effluent would contain radioactive materials at levels below regulatory limits and would be released to evaporation ponds, percolated through the soil column, or discharged to the Columbia River.

The HWR facilities would generate 3,600 m³/yr (P) of solid LLW. Including the 6,500 m³/yr (P) of LLW grout produced during liquid LLW processing, the total NPR-related solid LLW stream would be 10,000 m³/yr (P). This waste would be disposed of at the Low-Level Radioactive Waste Disposal Facility.

Mixed Waste and Hazardous Waste (HWR, Hanford)

The HWR facilities would generate approximately 880 m³/yr (TP) of mixed waste that would be sorted at the Waste Receiving and Processing Facility. Sorting would remove approximately 16 m³/yr (P) of TRU waste from the mixed waste stream. A total of 870 m³/yr (T) of non-TRU mixed waste would be stored in tanks or drums in the mixed waste storage buildings. The HWR operations would generate 37 m³/yr (TP) of hazardous waste that would be sent to the Nonradioactive Dangerous Waste Storage Facility for interim storage prior to off-site disposal (WHC 1989a; WHC 1990a).

Other Waste (HWR, Hanford)

The HWR operations would generate 2,100 m³/yr (T) of solid other waste for disposal at the Hanford Central Landfill. Liquid effluents would contain radioactive materials at levels below regulatory limits and would be released to evaporation ponds, percolated through the soil column, or discharged to the Columbia River. The 4300 m³/yr (P) of condensate from the 242-A evaporator would be treated at the Effluent Treatment Facility in the 200-East Area. The solid waste obtained from this facility's treatment media would be disposed of as LLW at the Low-Level Radioactive Waste Disposal Facility. The driver-fuel and target-processing operations and various other waste management processes would discharge 23,000 m³/yr (T) of sanitary waste to tile fields on-site, 33,000 m³/yr (P) of wastewater to solar evaporation ponds, and 170,000 m³/yr (P) of Effluent Treatment Facility discharges to artificial ponds, cribs, or the Columbia River. DOE's Office of Environmental Restoration and Waste Management will review and assess the merits of these different disposal procedures (DOE 1990b). Settled solids from these effluent streams would be collected and disposed of in the site landfill.

Approximately 19,000 m³/yr (TP) of sanitary waste would be sent to the Sewage Treatment Facility in the NPR area. Effluents from both treatment systems would be discharged to the ground or the Columbia River in accordance with applicable rules and regulations. Settled solids would be disposed of in the site landfill.

M.3.1.2 Waste Produced by the LWR Technology at Hanford

This section describes the upper-bound waste volumes generated by operating an LWR and associated facilities at Hanford. The effects of such operations on Hanford waste management facilities are discussed in Section 5.2.10. Waste management flow diagrams for both the tritium and plutonium production modes are presented in Figures M.10 through M.15.

High-Level Waste (LWR, Hanford)

The Spent Fuel Processing Complex would generate $67 \text{ m}^3/\text{yr}$ (P mode is maximum) of liquid HLW. This waste would be sent to double-shell tanks in the 200-East Area for interim storage before being processed at the Waste Vitrification Plant into two radioactive waste streams. The liquid waste would be routed to the 242-A evaporator for further processing and disposal as LLW. The solid waste containing the fission products would be processed into $18 \text{ m}^3/\text{yr}$ (T mode is maximum) of glasslike ceramic solids at the Waste Vitrification Plant.

Transuranic Waste (LWR, Hanford)

The Spent Fuel Processing Complex would generate $232 \text{ m}^3/\text{yr}$ (P) of TRU waste. An additional $16 \text{ m}^3/\text{yr}$ (P) of TRU waste would be generated during mixed waste sorting operations at the Waste Receiving and Processing Facility. The $250 \text{ m}^3/\text{yr}$ (P) of packaged TRU waste would be certified and stored at the Transuranic Waste Storage and Assay Facility for later off-site disposal at the proposed WIPP facility.

Low-Level Waste (LWR, Hanford)

The LWR facilities would generate two major streams of liquid LLW from NPR operations. The first stream would be $4,800 \text{ m}^3/\text{yr}$ (P) of liquid LLW generated primarily by spent fuel processing at the Spent Fuel Processing Complex. It would be stored in double-shell tanks before being combined with liquid LLW by-product from the Waste Vitrification Plant to produce $4,900 \text{ m}^3/\text{yr}$ (P) of liquid LLW. This material would be concentrated into $500 \text{ m}^3/\text{yr}$ (P) of liquid at the 242-A evaporator in the 200-East Area.

The second liquid LLW stream would be $5,000 \text{ m}^3/\text{yr}$ (P) of tritium-containing process condensate. This waste would be routed directly to storage tanks and combined with the $500 \text{ m}^3/\text{yr}$ (P) of 242-A evaporator liquid concentrate. This combined liquid waste would be solidified into $7,000 \text{ m}^3/\text{yr}$ (P) of grout at the Grout Treatment Facility and then disposed of in near-surface vaults at the Low-Level Radioactive Waste Disposal Facility.

The $4,400 \text{ m}^3/\text{yr}$ (P) of liquid condensate from the 242-A evaporator would be treated to further remove dissolved low-level radioactive solids. The purification media would be disposed of as LLW in site landfills. The remaining $4,400 \text{ m}^3/\text{yr}$ (P) of liquid effluent would contain radioactive materials at levels below regulatory limits and would be released to evaporation ponds, percolated through the soil column, or discharged to the Columbia River.

The LWR facilities would generate $3,200\text{--}3,600 \text{ m}^3/\text{yr}$ (TP) of solid LLW. This waste would be packaged at various NPR facilities and buried at the Low-Level Radioactive Waste

Disposal Facility. The total NPR-related solid LLW would be 11,000 m³/yr (P), a total that includes the LLW grout produced during liquid waste processing (WHC 1990a,b).

Mixed Waste and Hazardous Waste (LWR, Hanford)

The LWR facilities would generate approximately 810 m³/yr (TP) of mixed waste that would be sorted at the Waste Receiving and Processing Facility. Sorting would remove 16 m³/yr (P) of TRU waste from the mixed waste stream. A total of 790 m³/yr (TP) of non-TRU mixed waste would be stored in tanks or drums in the mixed waste storage buildings. The LWR operations would generate 290 m³/yr (TP) of hazardous waste that would be sent to the Nonradioactive Dangerous Waste Storage Facility for interim storage prior to off-site disposal.

Other Waste (LWR, Hanford)

The LWR operations would generate 3,100 m³/yr (T) of solid other waste for disposal at the Hanford Central Landfill. An additional 12,000 m³/yr (P) of service effluents would be discharged to site solar evaporation ponds. A total of approximately 33,000 m³/yr (T) of sanitary waste from nonreactor facilities would be discharged to Hanford sanitary tile fields. The reactor-generated sanitary waste, 27,000 m³/yr (TP), would be treated at the Sewage Treatment Facility and the effluent discharged to the ground. Cooling tower blowdown waters, 4,200,000 m³/yr (TP), from LWR operations would be discharged to the Columbia River.

M.3.1.3 Waste Produced by the MHTGR Technology at Hanford

This section describes the upper-bound waste volumes generated by operating an MHTGR and associated facilities at Hanford. The effects of such operations on Hanford waste management facilities are described in Section 5.2.10. Waste management flow diagrams for both the tritium and plutonium production modes are presented in Figures M.16 through M.21.

High-Level Waste (MHTGR, Hanford)

The driver-fuel and target-processing operations at the Spent Fuel Processing Complex would generate 60 m³/yr (P) of liquid HLW. This waste would be sent to double-shell tanks in the 200-East Area for interim storage before being processed at the Waste Vitrification Plant into two radioactive waste streams. The liquid waste, 60 m³/yr (P), would be routed to the 242-A evaporator for further processing and disposal as LLW. The solid waste containing the fission products would be processed into 38 m³/yr (P) of glasslike ceramic solids at the Waste Vitrification Plant. The corresponding HLW totals for the tritium production mode would be 51 m³/yr of liquid HLW and 33 m³/yr of vitrified solids.

Transuranic Waste (MHTGR, Hanford)

The Spent Fuel Processing Complex would generate 114 m³/yr (P) of TRU waste. An additional 16 m³/yr of TRU waste would be generated during mixed waste sorting operations

at the Waste Receiving and Processing Facility. The 130 m³/yr (P) of packaged TRU waste would be certified and stored at the Transuranic Waste Storage and Assay Facility for later off-site disposal at the proposed WIPP facility.

Low-Level Waste (MHTGR, Hanford)

The MHTGR facilities would generate 6,300 m³/yr (P) of liquid LLW. This waste would be concentrated and converted into grout at the Grout Treatment Facility for disposal in vaults. A total of 880 m³/yr (P) of grout would be generated from the liquid LLW stream. Effluent from treatment and concentration of LLW would be discharged for effluent treatment at a rate of 5,700 m³/yr (P).

A second grout system would be included with the reactor area fuel and target fabrication facilities for solidifying sodium chloride solutions containing traces of uranium. These solutions would be regulated by the Washington Department of Ecology. The MHTGR technology would generate 19,000 m³/yr (P) of grout waste containing uranium and sodium chloride. This grout would be disposed of in special on-site vaults. Including all the grouted LLW, the MHTGR technology would generate 28,000 m³/yr (P) of solid LLW.

Mixed Waste and Hazardous Waste (MHTGR, Hanford)

The MHTGR facilities would generate 240 m³/yr (P) of mixed waste that would be sorted at the Waste Receiving and Processing Facility. Sorting would remove 16 m³/yr (P) of TRU waste from the mixed waste stream. A total of 240 m³/yr (P) of non-TRU mixed waste would be stored in tanks and drums in the mixed waste storage buildings. The MHTGR operations would generate 57 m³/yr (P) of hazardous waste that would be sent to the Nonradioactive Dangerous Waste Storage Facility for interim storage prior to off-site disposal.

Other Waste (MHTGR, Hanford)

The MHTGR operations would generate 2,800 m³/yr (P) of solid other waste for disposal at the Hanford Central Landfill. An additional 22,000 m³/yr (P) of service effluents would be discharged to site solar evaporation ponds. A total of 30,000 m³/yr (P) of sanitary waste from nonreactor facilities would be discharged to Hanford sanitary tile fields. The reactor-generated sanitary waste, 26,000 m³/yr (P), would be treated at an on-site NPR sanitary tile field.

M.3.2 INEL Operations Waste

M.3.2.1 Waste Produced by the HWR Technology at INEL

This section describes the upper-bound waste volumes generated by operating an HWR and associated facilities at INEL. The effects of such operations on INEL waste management facilities are discussed in Section 5.3.10. Waste management flow diagrams for both the tritium and plutonium production modes are presented in Figures M.22 through M.27.

High-Level Waste (HWR, INEL)

The Low-Enriched Fuel Processing Facility and the Plutonium-Target Processing Facility would generate 51 m³/yr (P) of first-cycle extraction fluids, which are HLW. Fluid from the first fuel and target extraction, concentrates from secondary extractions, and waste recycle fluid from the New Waste Calcining Facility would be sent as primary liquid HLW to the Idaho Chemical Processing Plant. A total of 562 m³/yr (TP) of dilute, second- and third-cycle, low-level extraction fluids would be sent to the process equipment waste evaporator, where they would be reduced to 8 m³/yr (TP) of concentrated high-activity LLW and then combined with the first-cycle, high-level extraction fluids. A total of 60 m³/yr (P) of liquid HLW would result from this combination. The condensate from the evaporation would be routed to a LLW Effluent Treatment Facility. This waste would then be processed into 7 m³/yr (P) of calcine at the New Waste Calcining Facility before being sent to the Calcined Solids Storage Facility for interim storage. Beginning in about 2011, the NPR input of 7 m³/yr (P) of calcine would be vitrified into 4 m³/yr (P) of a glasslike ceramic at the Waste Immobilization Facility. Final disposal of the ceramic material would be at a Federal repository.

Fuel and target-processing facilities would generate 19 m³/yr (P) of combustible organic and radioactive mixed waste that would be sampled for uranium. If the uranium concentration were lower than a predetermined activity level, the organics would be used as fuel in the New Waste Calcining Facility. If the uranium concentration exceeded the allowable activity limit, further processing would occur at the Idaho Chemical Processing Plant. The product would ultimately be disposed of as LLW.

Transuranic Waste (HWR, INEL)

The HWR technology would generate 190 m³/yr (P) of TRU waste. Included would be contaminated spent clothing, high-efficiency particulate air filters, extraction resins, scrap equipment, and laboratory glassware. Most of this waste, 104 m³/yr (P), would be noncombustible and placed in drums at the individual NPR facilities before being examined at the Stored Waste Examination Pilot Plant. A total of 74 m³/yr (P) of combustible TRU waste would be incinerated and solidified into a volume of 3 m³/yr (P) at a facility with capabilities similar to those of the original Process Reduction Experimental Pilot Plant. All TRU wastes would be sorted, analyzed, and/or certified at the Stored Waste Examination Pilot Plant prior to interim storage at the Transuranic Waste Storage Area prior to ultimate disposal (110 m³/yr [P]) at the proposed WIPP facility. The processing of TRU waste would remove 5% of the waste from the TRU waste category. This 5 m³/yr (P) of material would be routed and handled as LLW.

Low-Level Waste (HWR, INEL)

Liquid LLW would be monitored for radioactivity prior to release to the appropriate liquid waste handling system. High-activity liquid LLW and concentrate from the Process Equipment Waste Evaporator would be processed at the Idaho Chemical Processing Plant. The condensate from the process equipment waste evaporator would contain levels of radioactivity below regulatory limits and could therefore be discharged to an evaporation pond. Treatment media from the evaporator system would be disposed of on-site as LLW.

The HWR facilities would generate 3,600 m³/yr (T) of solid LLW that would be sorted by types (i.e., combustible, compactible, or noncombustible/noncompactible). About 56% (2,000 m³/yr [P]) of this waste would be noncombustible/noncompactible. These materials would be placed in drums and sent directly to the Subsurface Disposal Area. Approximately 1,800 m³/yr (T) of LLW would be processed at the Waste Experimental Reduction Facility, either by incineration or compaction. These waste volume reduction procedures would discharge 190 m³/yr (TP) of processed solid LLW to the Subsurface Disposal Area. An additional 5 m³/yr (P) of LLW by-products from TRU waste processing would also be discharged to the Radioactive Waste Management Complex. A total of 2,200 m³/yr (P) of LLW would be disposed of at the Subsurface Disposal Area.

Mixed Waste and Hazardous Waste (HWR, INEL)

The HWR technology would generate 420 m³/yr (P) of mixed waste that would be stored at the Radioactive Mixed Waste Storage Facility. A total of 43 m³/yr (P) of hazardous waste generated by HWR operations would be sent to the Hazardous Waste Shipping Facility for storage prior to off-site disposal at RCRA-permitted facilities.

Other Waste (HWR, INEL)

The HWR input of solid other waste (e.g., trash) to the Central Facilities Area Landfill would be 7,700 m³/yr (P). A total of 52,000 m³/yr (P) of liquid sanitary waste would be discharged to sanitary tile fields. Blowdown water and other noncontact waters from NPR facilities would total 7,600,000 m³/yr (T) and would be sent to a lined evaporation pond within the reactor complex. The 500,000 m³/yr (P) of liquid effluent from NPR facilities other than the reactor facility would be disposed of via evaporation ponds that are part of existing waste facilities at INEL. A total of 1 m³/yr (TP) of solids from sewage treatment operations would be disposed of in the Central Facilities Area Landfill (Table 5.3.10-3).

M.3.2.2 Waste Produced by the LWR Technology at INEL

This section describes the upper-bound waste volumes generated by operating an LWR and associated facilities at INEL. The effect of such operations on INEL waste management facilities are discussed in Section 5.3.10. Waste management flow diagrams for both the tritium and plutonium production modes are presented in Figures M.28 through M.33.

High-Level Waste (LWR, INEL)

The Low-Enriched Fuel Processing Facility and the Plutonium-Target Processing Facility would generate 67 m³/yr (P) of first-cycle extraction fluids, which are HLW. These fluids would be sent to the Idaho Chemical Processing Plant. A total of 9,900 m³/yr (P) of low-activity decontamination cycle liquids would be sent to the process equipment waste evaporator to be concentrated into a high-activity LLW fraction, 140 m³/yr (P), which would then be combined with other liquid HLW at the Idaho Chemical Processing Plant. The condensate from the evaporator would be routed to a LLW Effluent Treatment Facility. The combined 210 m³/yr (P) of NPR-generated liquid HLW would be converted into 33 m³/yr (P) of calcined solids at the

New Waste Calcining Facility and then sent to the Calcined Solids Storage Facility. Beginning in 2011, the stored materials would be vitrified into 21 m³/yr (P) of glasslike ceramic solids at the Waste Immobilization Facility. Final disposal of the ceramic material would be at a Federal repository.

Transuranic Waste (LWR, INEL)

The LWR technology would generate 280 m³/yr (P) of TRU waste of the type described in the section describing the TRU waste generated by the HWR technology at INEL. Combustible TRU waste, 260 m³/yr (P), would be incinerated and processed into 50 m³/yr (P) of solid TRU waste at the Process Reduction Experimental Pilot Plant. Although the Process Reduction Experimental Pilot Plant will not be operational at NPR start-up, the analysis in the EIS assumes the existence of a facility with similar capabilities. A total of 20 m³/yr (TP) of noncombustible TRU waste would be placed in drums and sent to the Transuranic Waste Storage Area for disposal after examination at the Stored Waste Examination Pilot Plant. All TRU waste would be sorted, analyzed, and/or certified at the Stored Waste Examination Pilot Plant prior to interim storage at the Transuranic Waste Storage Area and ultimate disposal (20 m³/yr [TP]) at the proposed WIPP facility. This processing of TRU waste would remove 50 m³/yr (P) of the waste from the TRU waste category. This material would then be routed and handled as LLW.

Low-Level Waste (LWR, INEL)

The condensate from the process equipment waste evaporator would contain radioactive materials at levels below regulatory limits and would be discharged to an evaporation pond. Treatment media from the evaporator system would be disposed of on-site as LLW.

The LWR facilities would generate 3,100 m³/yr (TP) of solid LLW that would be sorted by type (i.e., combustible, compactible, or noncombustible/noncompactible). A total of 2,300 m³/yr (P) would be noncombustible/noncompactible. These solids would be placed in drums and sent directly to the Subsurface Disposal Area. A total of 900 m³/yr (T) of LLW would be processed at the Waste Experimental Reduction Facility, either by incineration (500 m³/yr) (T) or compaction (400 m³/yr) (T). These waste volume reduction procedures would discharge 100 m³/yr (T) of processed solid LLW. The 50 m³/yr (P) of LLW by-products from TRU waste processing would be combined with the processed solid LLW. A total of 2,400 m³/yr (P) of LLW would be disposed of at the Subsurface Disposal Area.

Mixed Waste and Hazardous Waste (LWR, INEL)

The LWR technology would generate 160 m³/yr (TP) of mixed waste that would be stored at the Radioactive Mixed Waste Storage Facility. LWR operations would generate 1,100 m³/yr (T) of hazardous waste that would be sent to the Hazardous Waste Shipping Facility for storage prior to off-site disposal at RCRA-permitted facilities.

Other Waste (LWR, INEL)

The LWR input of solid other wastes (e.g., trash) to the Central Facilities Area. Landfill would be 9,000 m³/yr (T). A total of 60,000 m³/yr (T) of liquid sanitary waste would be discharged to sanitary tile fields. Blowdown water and other noncontact waters from NPR facilities would total 4,200,00 m³/yr (P) and would be sent to a lined evaporation pond within the reactor complex. The 72,000 m³/yr (P) of liquid effluent from NPR facilities other than the reactor facility would be disposed of via evaporation ponds that are part of existing waste facilities at INEL. A total of 1 m³/yr (TP) of solids from sewage treatment operations would be disposed of in the Central Facilities Area Landfill (Table 5.3.10-3).

M.3.2.3 Waste Produced by the MHTGR Technology at INEL

This section describes the upper-bound waste volumes generated by operating an MHTGR and associated facilities at INEL. The effects of such operations on INEL waste management facilities are discussed in Section 5.3.10. Waste management flow diagrams for both the tritium and plutonium production modes are presented in Figures M.34 through M.39.

High-Level Waste (MHTGR, INEL)

The Low-Enriched Fuel Processing Facility and the Plutonium-Target Processing Facility would generate 310 m³/yr (P) of liquid HLW. This waste would include first-cycle extraction fluids and concentrates from second- and third-cycle extractions. Dilute second- and third-cycle, low-level extraction fluids, 200 m³/yr (P), would first be sent to the process equipment waste evaporator, where it would be reduced to 3 m³/yr (TP) of high-activity LLW that would be combined with other liquid HLW. The combined 310 m³/yr (P) of liquid HLW would be processed into 4 m³/yr (P) of calcine. The calcined HLW would be sent to the Calcined Solids Storage Facility for interim storage. Beginning in 2011, it would be sent to the Waste Immobilization Facility, where it would be vitrified into 3 m³/yr (P) of glasslike ceramic. Final disposal of this ceramic material would be at the Federal repository. Fuel and target-processing facilities would generate 38 m³/yr (P) of combustible organic/radioactive mixed waste, which would be sampled for uranium. If the uranium concentration were less than a predetermined activity level, the organics would be used as fuel in the New Waste Calcining Facility. If the uranium concentration exceeded the allowable activity limit, further processing would occur at the Idaho Chemical Processing Plant. The product material would ultimately be disposed of as LLW.

Transuranic Waste (MHTGR, INEL)

The MHTGR technology would generate 240 m³/yr (P) of TRU waste of the type described in the section describing the TRU waste generated by the HWR technology at INEL. The majority of TRU waste, 210 m³/yr (P), would be noncombustible; approximately 30 m³/yr (P) of combustible TRU waste would be incinerated and processed into 4 m³/yr (P) of solid TRU waste at a facility with capabilities similar to those of the original Process Reduction Experimental Pilot Plant. All TRU waste would be sorted, analyzed, and/or certified at the Stored Waste Examination Pilot Plant prior to interim storage at the Transuranic Waste Storage Area and proposed final disposal (200 m³/m [P]) at WIPP. This processing of TRU waste would

remove 31 m³/yr (P) of the waste from the TRU waste category. This material would then be routed and handled as LLW.

Low-Level Waste (MHTGR, INEL)

The condensate from the process equipment waste evaporator would contain levels of radioactive materials at levels below regulatory limits and would be discharged to an evaporation pond. Because water would not be used as a moderator or coolant in the MHTGR, relatively small amounts of liquid LLW would be generated by the reactor facility.

The Fuel and Tritium-Target Fabrication Facility would generate 150 m³/yr (P) of special sodium chloride waste containing uranium. This waste would be stored on-site. Methods to process and dispose of this waste have not been decided upon.

The MHTGR facilities would generate 7,360 m³/yr (P) of solid LLW. After being sorted by type, 4,900 m³/yr (P) of noncompactible waste would be placed in drums and sent directly to the Subsurface Disposal Area. The remaining waste would be incinerated (1,020 m³/yr [P]) or compacted (1,420 m³/yr [P]) at the Waste Experimental Reduction Facility. The processed LLW would be combined with the noncompactible waste and the 31 m³/yr of LLW resulting from TRU waste processing, for a total of 5,300 m³/yr (P). The LLW would be disposed of at the Subsurface Disposal Area.

Mixed Waste and Hazardous Waste (MHTGR, INEL)

The MHTGR facilities would generate 6 m³/yr (P) of mixed waste that would be stored at the Radioactive Mixed Waste Storage Facility. The 59 m³/yr (P) of hazardous waste generated by the MHTGR technology would be sent to the Hazardous Waste Shipping Facility for storage prior to off-site disposal at RCRA-permitted facilities.

Other Waste (MHTGR, INEL)

A total of 10,000 m³/yr (P) of solid other waste (e.g., trash) from MHTGR operations would be disposed of at the Central Facilities Area Landfill. About 69,000 m³/yr (P) of liquid other waste would be discharged to site sanitary tile fields. Approximately 6.8×10^6 m³/yr (TP) of blowdown water and other noncontact rinse water from NPR facilities would be sent to new site evaporation ponds. The 6.9×10^5 m³/yr (P) of liquid effluent from NPR facilities other than the reactor facility would be disposed of via evaporation ponds that are part of existing waste facilities at INEL. Approximately 1 m³/yr (TP) of solids from sewage treatment operations would be disposed of in the Central Facilities Area Landfill.

M.3.3 SRS Operations Waste

M.3.3.1 Waste Produced by the HWR Technology at SRS

This section describes the upper-bound waste volumes generated by operating an HWR and associated facilities at SRS. The effects of such operations on SRS waste management facilities are discussed in Section 5.4.10. Waste management flow diagrams for both the tritium production and plutonium production modes are presented in Figures M.40 through M.45.

High-Level Waste (HWR, SRS)

A total of 2000 m³/yr (P) of liquid HLW would be generated from the processing of spent driver fuel and targets from NPR operations. Processing in the waste tanks would yield about 150 m³/yr (P) of insoluble sludge solids that would be combined with 240 m³/yr (P) of washed precipitate slurry before being converted to 30 m³/yr (P) of borosilicate glass in the Defense Waste Processing Facility. Decontaminated waste salt solutions would be immobilized as 6,400 m³/yr (P) of saltstone for on-site disposal. A total of 3,400 m³/yr (P) of evaporator overheads from waste concentration operations would be treated and discharged to on-site streams.

Transuranic Waste (HWR, SRS)

The TRU waste generated at the fuel and target processing facilities would be processed and packaged at the Transuranic Waste Facility. Prior to off-site disposal, the packaged TRU waste would be sent to the Waste Certification Facility to be certified as complying with TRU waste acceptance criteria for WIPP. The HWR technology would generate 220 m³/yr (P) of processed, packaged TRU waste. This waste would include contaminated spent clothing, high-efficiency particulate air filters, extraction resins, scrap equipment, and laboratory glassware. The processing of TRU waste would remove 100 m³/yr of the initial waste from the TRU waste category. This material would then be routed and handled as LLW. The 110 m³/yr (P) of TRU waste is proposed to be disposed of off-site at the WIPP facility.

Low-Level Waste (HWR, SRS)

The HWR facilities technology would generate 51,000 m³/yr (P) of liquid LLW. from NPR processing facilities. As a result of the various processes of the F- and H-Area Effluent Treatment Facility (e.g., particulate and carbon filtration, ion exchange, and reverse osmosis), the liquid waste would be separated into 680 m³/yr (P) of solid LLW and 63,000 m³/yr (P) of treated effluent. The LLW from this facility would be converted into a saltstone slurry and ultimately solidified into 880 m³/yr (P) of LLW that would be disposed of on-site at the Saltstone Facility Disposal Vaults (Z Area). The treated effluent would contain radionuclides at levels that would not exceed discharge criteria under the National Pollutant Discharge Elimination System. Overall, HWR facilities would produce 7,200 m³/yr (P) of processed LLW (to be disposed of at the Saltstone Facility Disposal Vaults [Z Area]) and 3800 m³/yr (P) of solid LLW (to be packaged and sent to the Burial Ground Expansion for disposal).

Mixed Waste and Hazardous Waste (HWR, SRS)

Mixed waste would be processed through the planned Hazardous/Mixed Waste Disposal Facility. Mixed waste generated at the various NPR facilities would be sorted into several waste streams. Nonincinerable solid mixed waste that could be packaged for burial would be routed directly to above-grade concrete storage vaults. Combustible mixed and/or hazardous waste would be incinerated and prepared for disposal at the Consolidated Incinerator Facility. The processing would reduce the waste volume by approximately 95%. The ash from incineration would then be immobilized in grout and disposed of in concrete storage vaults. After processing, the total amount of disposable mixed and hazardous waste generated by NPR operations would be 570 m³/yr (P).

Other Waste (HWR, SRS)

The HWR input of other waste to the site landfill would be 2,300 m³/yr (P), which represents a 9% increase in the landfill waste stream. Nonradioactive liquid waste generated by HWR facilities would include sanitary waste, as well as blowdown water and noncontact water from reactor facilities. Such wastes are discharged without treatment in accord with applicable rules and regulations. A total of 31,000 m³/yr (P) of liquid sanitary waste would be treated and discharged to on-site streams.

M.3.3.2 Waste Produced by the LWR Technology at SRS

This section describes the upper-bound waste volumes generated by operating an LWR and associated facilities at SRS. The effects of such operations on SRS waste management facilities are discussed in Section 5.4.10. Waste management flow diagrams for both the tritium production and plutonium production modes are presented in Figures M.46 through M.51.

High-Level Waste (LWR, SRS)

A total of 1,100 m³/yr (T) of liquid HLW would be generated from the processing of spent fuel from NPR operations. Processing in the waste tanks would yield about 82 m³/yr (T) of insoluble sludge solids that would be combined with 120 m³/yr (T) of washed precipitate slurry before being converted to 15 m³/yr (T) of borosilicate glass in the Defense Waste Processing Facility. Decontaminated waste salt solutions would be immobilized as 3,400 m³/yr (T) of saltstone for on-site disposal. A total of 1,800 m³/yr (T) of evaporator overheads from waste concentration operations would be routed to effluent treatment.

Transuranic Waste (LWR, SRS)

The TRU waste generated at the fuel and target processing facilities would be processed and packaged at the Transuranic Waste Facility. Prior to off-site disposal, the packaged TRU waste would be sent to the Waste Certification Facility to be certified as complying with TRU waste acceptance criteria for WIPP. The LWR technology would generate 230 m³/yr (P) of processed, packaged TRU waste. This waste would include contaminated spent clothing, high-efficiency particulate air filters, extraction resins, scrap equipment, and laboratory glassware.

The processing of TRU waste would remove 50% of the initial waste from the TRU waste category. This material would then be routed and handled as LLW. The 110 m³/yr (P) of TRU waste would ultimately be disposed of off-site at the proposed WIPP facility.

Low-Level Waste (LWR, SRS)

The LWR technology would generate 22,000 m³/yr (P) of liquid LLW from NPR processing facilities. As a result of the various processes of the F- and H-Area Effluent Treatment Facility (e.g., particulate and carbon filtration, ion exchange, and reverse osmosis), the liquid waste would be separated into 280 m³/yr (P) of solid LLW and 27,000 m³/yr (P) of treated effluent. The solid LLW from this facility would be converted into a saltstone slurry and ultimately solidified into 365 m³/yr (P) of LLW that would be disposed of on-site at the Saltstone Facility Disposal Vaults (Z Area). The treated effluent would contain radionuclides at levels that would not exceed discharge criteria under the National Pollutant Discharge Elimination System. Overall, LWR facilities would produce 3,400 m³/yr (T) of processed solid LLW (to be disposed of at the Saltstone Facility Disposal Vaults [Z Area]) and 4,100 m³/yr (T) of solid LLW (to be packaged and sent to the Burial Ground Expansion for disposal).

Mixed Waste and Hazardous Waste (LWR, SRS)

Mixed waste would be processed through the planned Hazardous/Mixed Waste Disposal Facility. Mixed waste generated at the various NPR facilities would be sorted into several waste streams. Nonincinerable solid mixed waste that could be packaged for burial would be routed directly to above-grade concrete storage vaults. Combustible mixed and/or hazardous waste would be incinerated and prepared for disposal at the Consolidated Incinerator Facility. The processing would reduce the waste volume by approximately 95%. The ash from incineration would then be immobilized in grout and disposed of in concrete storage vaults. After processing, the total amount of disposable mixed and hazardous waste generated by NPR operations would be 480 m³/yr (TP).

Other Waste (LWR, SRS)

The LWR input of other waste to the site landfill would be 3,800 m³/yr (T), which represents a 14% increase in the landfill waste stream. Nonradioactive liquid waste generated by LWR facilities would include sanitary waste, as well as blowdown water and noncontact water from reactor facilities. A total of 51,000 m³/yr (T) of liquid sanitary waste would be treated and discharged to on-site streams.

M.3.3.3 Waste Produced by the MHTGR Technology at SRS

This section describes the upper-bound waste volumes generated by operating an MHTGR and associated facilities at SRS. The effects of such operations on SRS waste management facilities are discussed in Section 5.4.10. Waste management flow diagrams for both the tritium production and plutonium production modes are presented in Figures M.52 through M.57.

High-Level Waste (MHTGR, SRS)

Approximately 2,000 m³/yr (P) of liquid HLW would be generated from the processing of spent fuel from NPR operations. Processing in the waste tanks would yield 150 m³/yr (P) of insoluble sludge solids that would be combined with 240 m³/yr (P) of washed precipitate slurry before being converted to 29 m³/yr (P) of borosilicate glass in the Defense Waste Processing Facility. Decontaminated waste salt solutions would be immobilized as 7,200 m³/yr (P) of saltstone for on-site disposal. A total of 3,400 m³/yr (P) of evaporator overheads from waste concentration operations would be routed to effluent treatment.

Transuranic Waste (MHTGR, SRS)

The TRU waste generated at the fuel and target processing facilities would be processed and packaged at the Transuranic Waste Facility. Prior to off-site disposal, the packaged waste would be sent to the Waste Certification Facility to be certified as complying with TRU waste acceptance criteria for WIPP. The MHTGR technology would generate 260 m³/yr (P) of processed, packaged TRU waste. This waste would include contaminated spent clothing, high-efficiency particulate air filters, extraction resins, scrap equipment, and laboratory glassware. The processing of TRU waste would remove 126 m³/yr of the initial waste from the TRU waste category. This material would then be routed and handled as LLW. The 150 m³/yr (P) of TRU waste would ultimately be disposed of off-site at the proposed WIPP facility.

Low-Level Waste (MHTGR, SRS)

The MHTGR technology would generate 51,000 m³/yr (P) of liquid LLW from NPR processing facilities. As a result of the various processes of the F- and H-Area Effluent Treatment Facility (e.g., particulate and carbon filtration, ion exchange, and reverse osmosis), the liquid waste would be separated into 680 m³/yr (P) of solid LLW and 63,000 m³/yr (P) of treated effluent. The solid LLW from this facility would be converted into a saltstone slurry and ultimately solidified into 875 m³/yr (P) of LLW that would be disposed of on-site at the Saltstone Facility Disposal Vaults (Z Area). The treated effluent would contain radionuclides at levels that would not exceed discharge criteria under the National Pollutant Discharge Elimination System. Overall, MHTGR facilities would generate 12,000 m³/yr (P) of solid LLW. The combustible portion of this waste, 7,200 m³/yr (P), would be processed at the Consolidated Incinerator Facility. Ultimately, 5,400 m³/yr (P) of processed solid LLW would be disposed of in on-site vaults.

Mixed Waste and Hazardous Waste (MHTGR, SRS)

Mixed waste would be processed through the planned Hazardous/Mixed Waste Disposal Facility. Mixed waste generated at the various NPR facilities would be sorted into several waste streams. Nonincinerable solid mixed waste that could be packaged for burial would be routed directly to above-grade concrete storage vaults. Combustible mixed and/or hazardous waste would be incinerated and prepared for disposal at the Consolidated Incinerator Facility. The processing would reduce the waste volume by approximately 95%. The ash from incineration would then be immobilized in grout and disposed of in concrete storage vaults.

After processing, the total amount of disposable mixed and hazardous waste generated by NPR operations would be 135 m³/yr (P).

Other Waste (MHTGR, SRS)

The MHTGR input of other waste to the site landfill would be 3,000 m³/yr (P), which represents a 12% increase in the landfill waste stream. Nonradioactive liquid waste generated by MHTGR facilities would include sanitary waste, as well as blowdown water and noncontact water from reactor facilities. A total of 40,000 m³/yr (P) of liquid sanitary waste would be treated and discharged to on-site streams. A total of 1,385 m³ (P) of this waste would be settled solids from treatment of nonradioactive liquid waste. These solids would be collected and sent to the site landfill for disposal.

TABLE M.1 Abbreviations of the Waste Management Facilities Depicted
in Figures M.1 through M.57

Abbreviation	Facility	Location
CIF	Consolidated Incineration Facility	SRS
CSSF	Calcined Solids Storage Facility	INEL
DWPF	Defense Waste Processing Facility	SRS
ETF	Effluent Treatment Facility	SRS
FMEF	Fuels and Materials Examination Facility (400 Area)	Hanford
FPB	Fuel Processing Building in new Spent Fuel Processing Complex (200-E Area)	Hanford
GTF	Grout Treatment Facility	Hanford
HWPF	Heavy-Water Processing Facility	All
HWSF	Hazardous Waste Shipping Facility	INEL
HWVP	Hanford Waste Vitrification Plant	Hanford
ICPP	Idaho Chemical Processing Plant	INEL
LET&D	Liquid Effluent Treatment and Disposal Facility	INEL
MWSF	Mixed Waste Storage Facility	Hanford
NWCF	New Waste Calcining Facility	INEL
NDWSF	Nonradioactive Dangerous Waste Storage Facility	Hanford
PEW	Process Equipment Waste Evaporator	INEL
PREPP	Process Reduction Experimental Pilot Plant ^a	INEL
REPF	Replacement Extraction and Purification Facility	SRS
RMWSF	Radioactive Mixed Waste Storage Facility	INEL
RWMC	Radioactive Waste Management Complex	INEL
SF-Y	Saltstone Facility (Y Area)	SRS
SF-Z	Saltstone Facility (Z Area)	SRS
SWEPP	Stored Waste Examination Pilot Plant	INEL
TSA	Transuranic (waste) Storage Area	INEL
TWF	Transuranic Waste Facility	SRS
UPB	Uranium Processing Building in new Spent Fuel Processing Complex (200-E Area)	Hanford
WCF	Waste Certification Facility	SRS
WERF	Waste Experimental Reduction Facility	INEL
WIF	Waste Immobilization Facility	INEL
WIPP	Waste Isolation Pilot Plant	-
WNP-1	Washington Public Power Supply System Reactor No. 1	Hanford
WRAP	Waste Receiving and Processing Facility	Hanford

^aAlthough the Process Reduction Experimental Pilot Plant will not be operational at NPR start-up, the analysis in the EIS assumes the existence of a facility with similar capabilities.

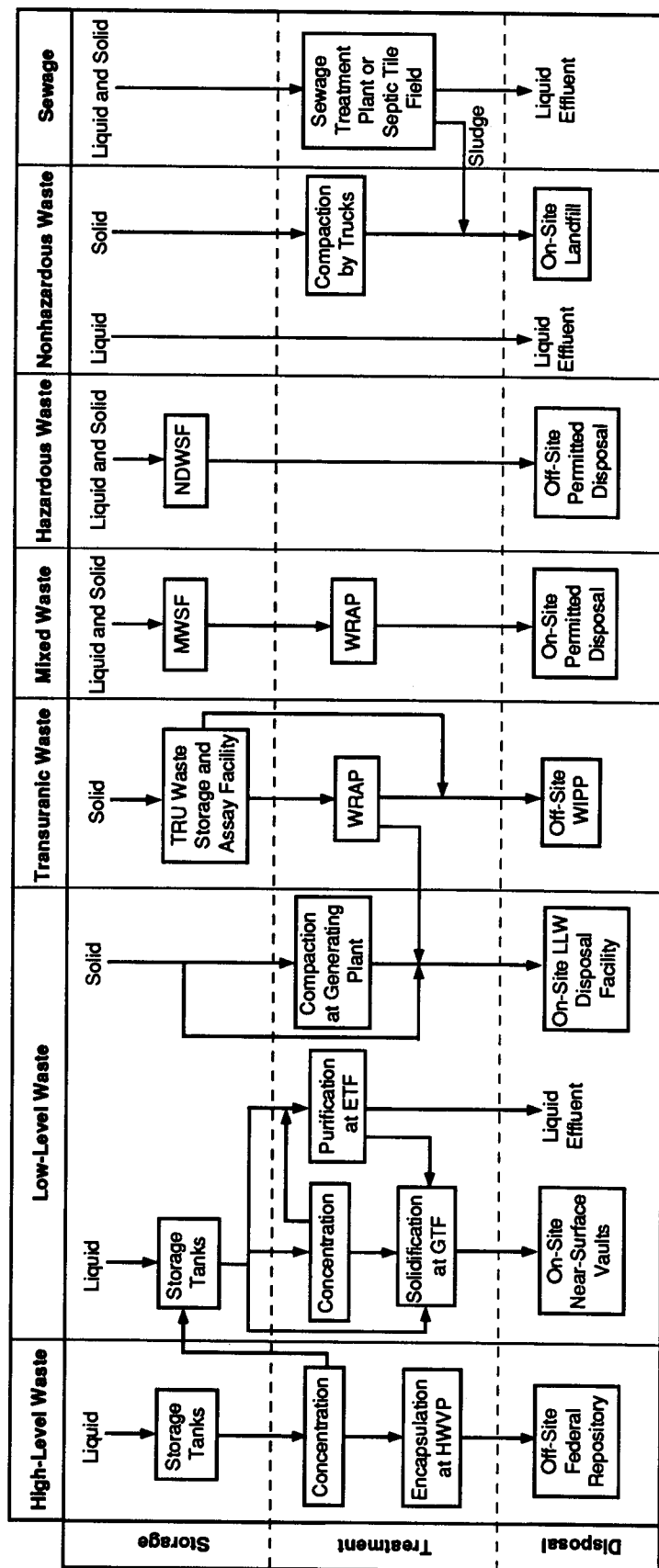


FIGURE M.1 Waste Management for NPR Operations at Hanford

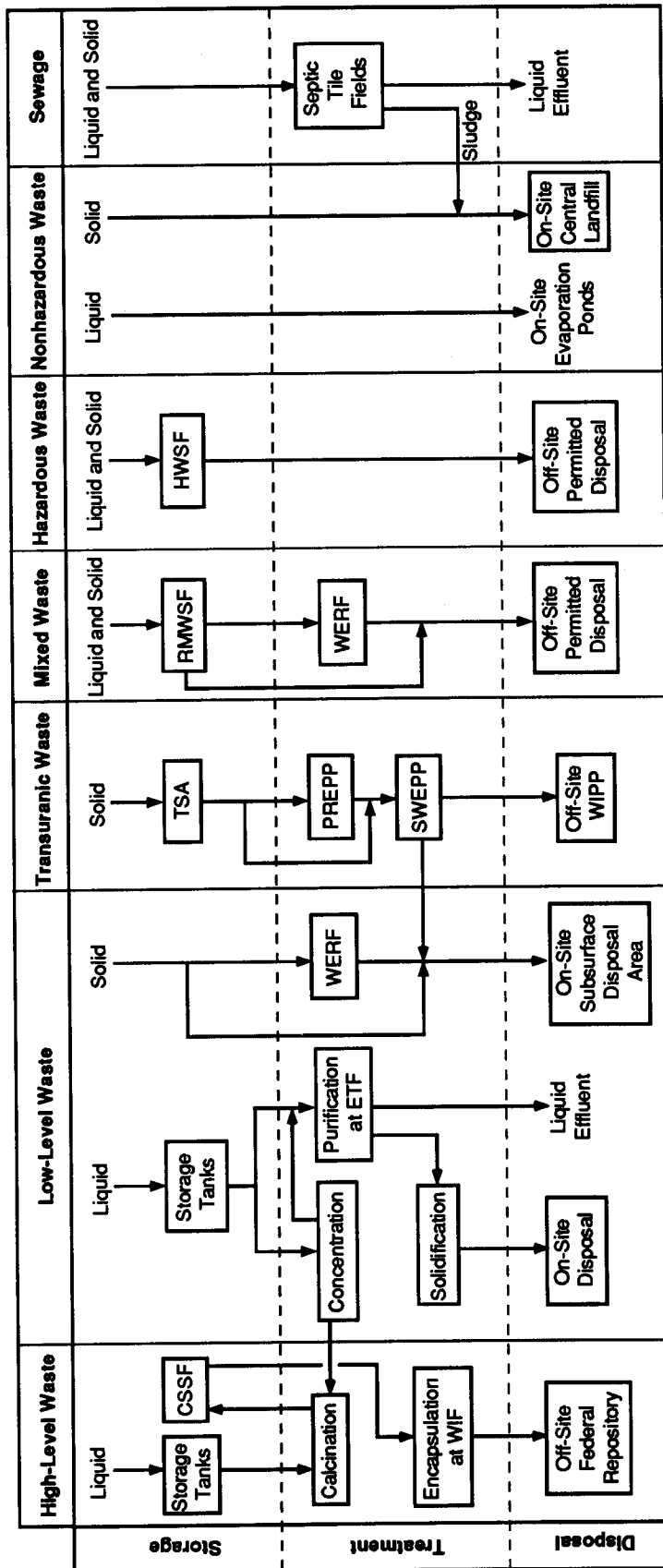


FIGURE M.2 Waste Management for NPR Operations at INEL

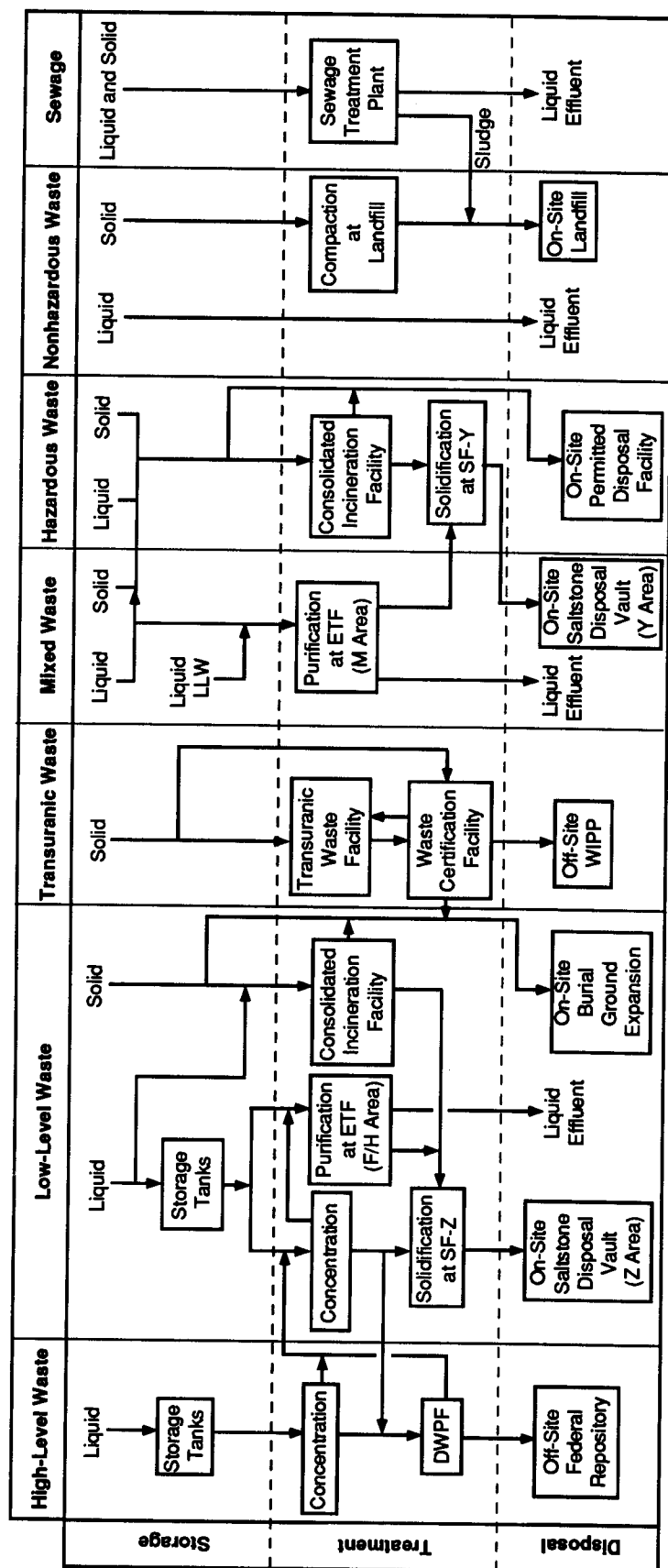


FIGURE M.3 Waste Management for NPR Operations at SRS

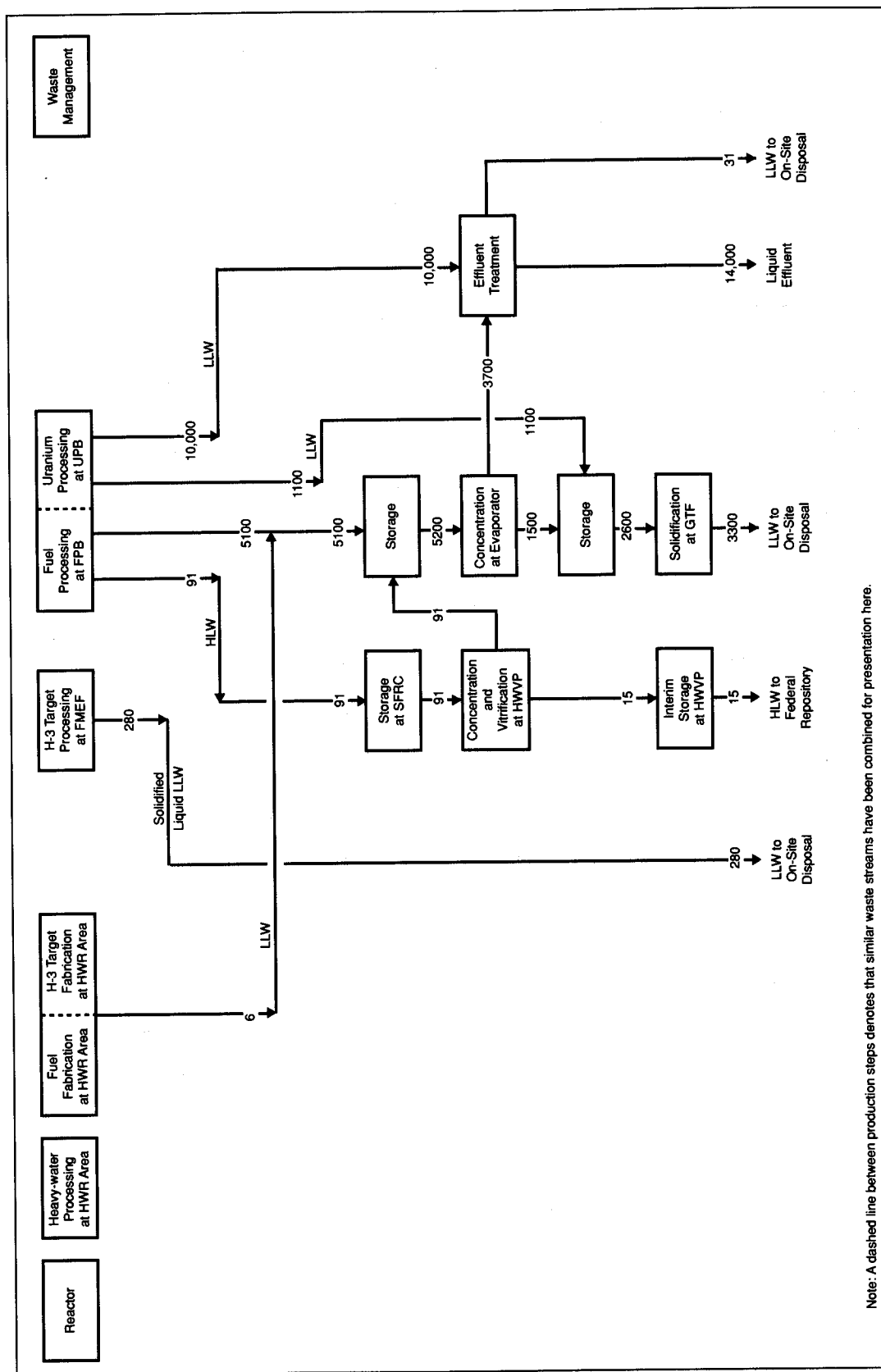


FIGURE M.4 Radioactive Liquid Waste Management for HWR Operations in the Tritium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989a-b; Evans 1990)

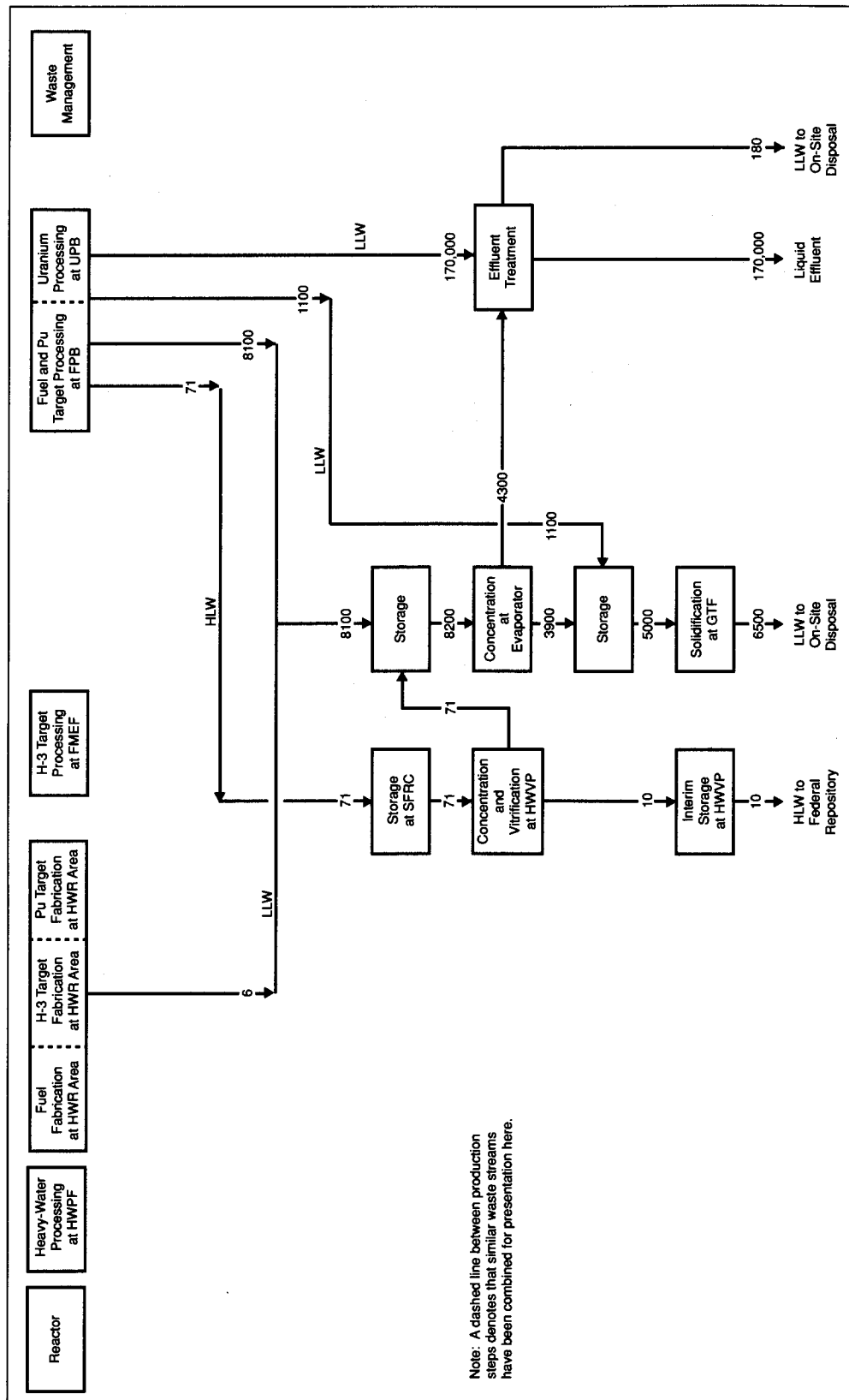


FIGURE M.5 Radioactive Liquid Waste Management for HWR Operations in the Plutonium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989a-b; Evans 1990)

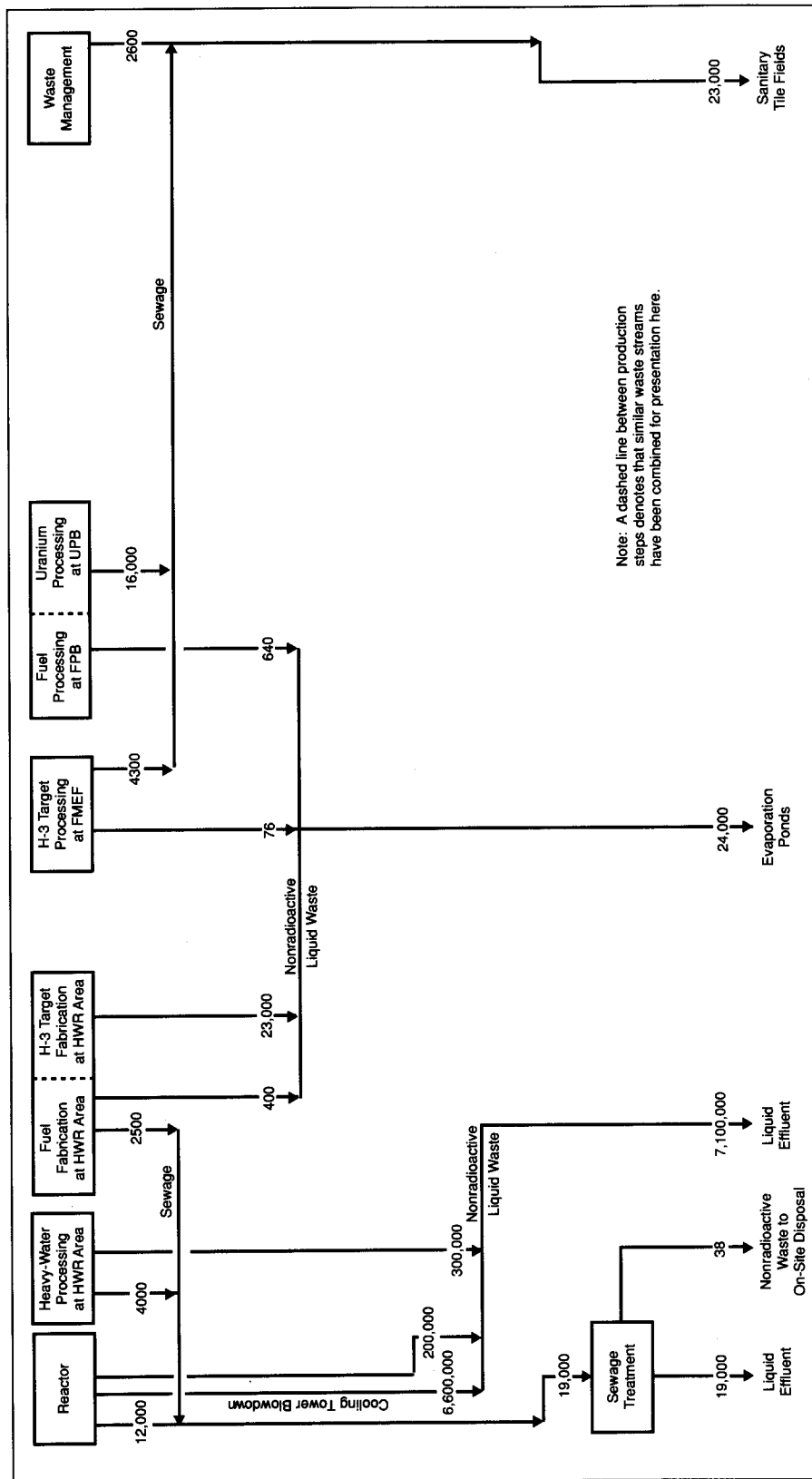


FIGURE M.6 Nonradioactive Liquid Waste Management for HWR Operations in the Tritium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989a-b; Evans 1990)

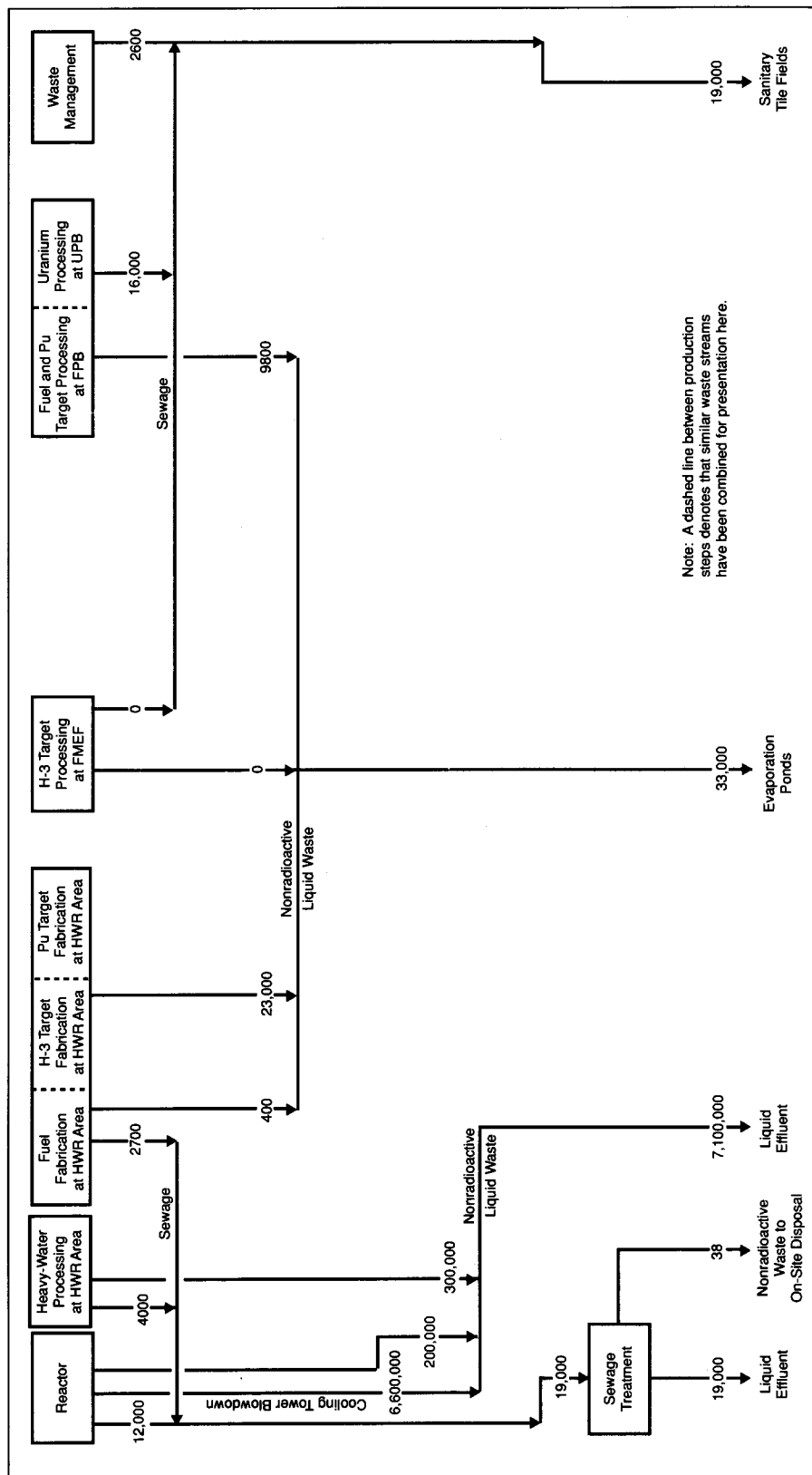


FIGURE M.7 Nonradioactive Liquid Waste Management for HWR Operations in the Plutonium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989a-b; Evans 1990)

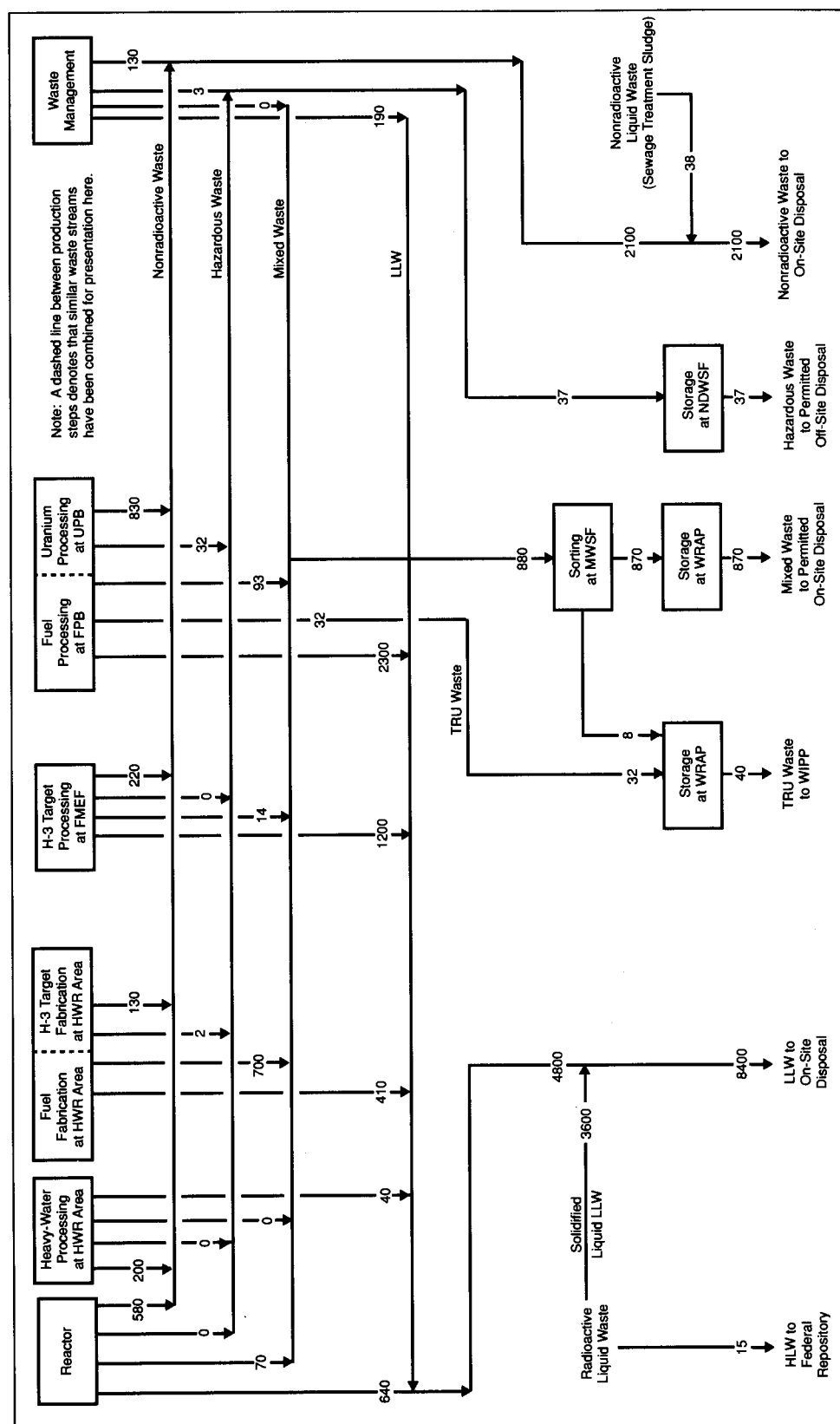


FIGURE M.8 Solid Waste Management for HWR Operations in the Tritium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989a-b; Evans 1990)

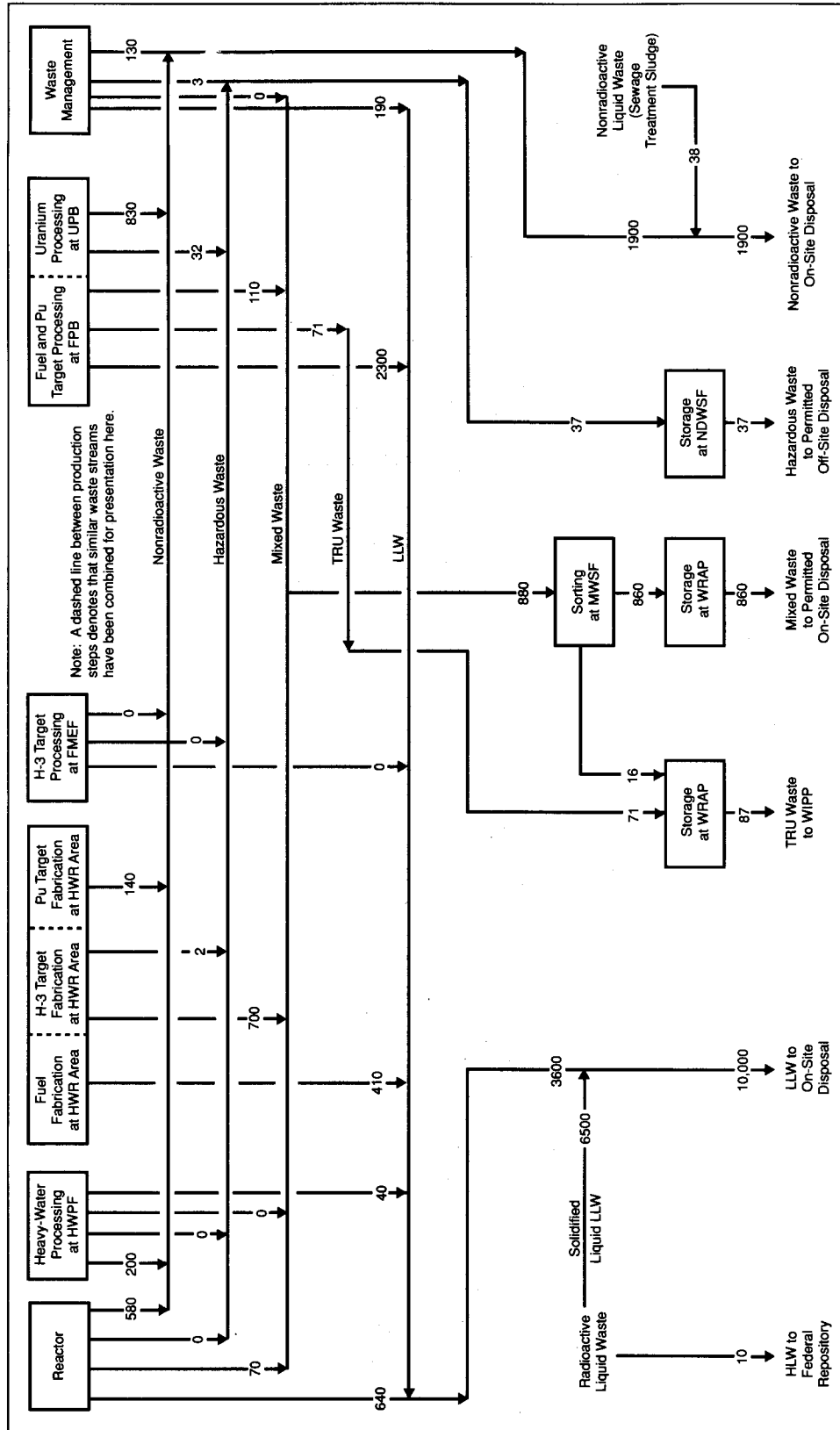


FIGURE M.9 Solid Waste Management for HWR Operations in the Plutonium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989a-b; Evans 1990)

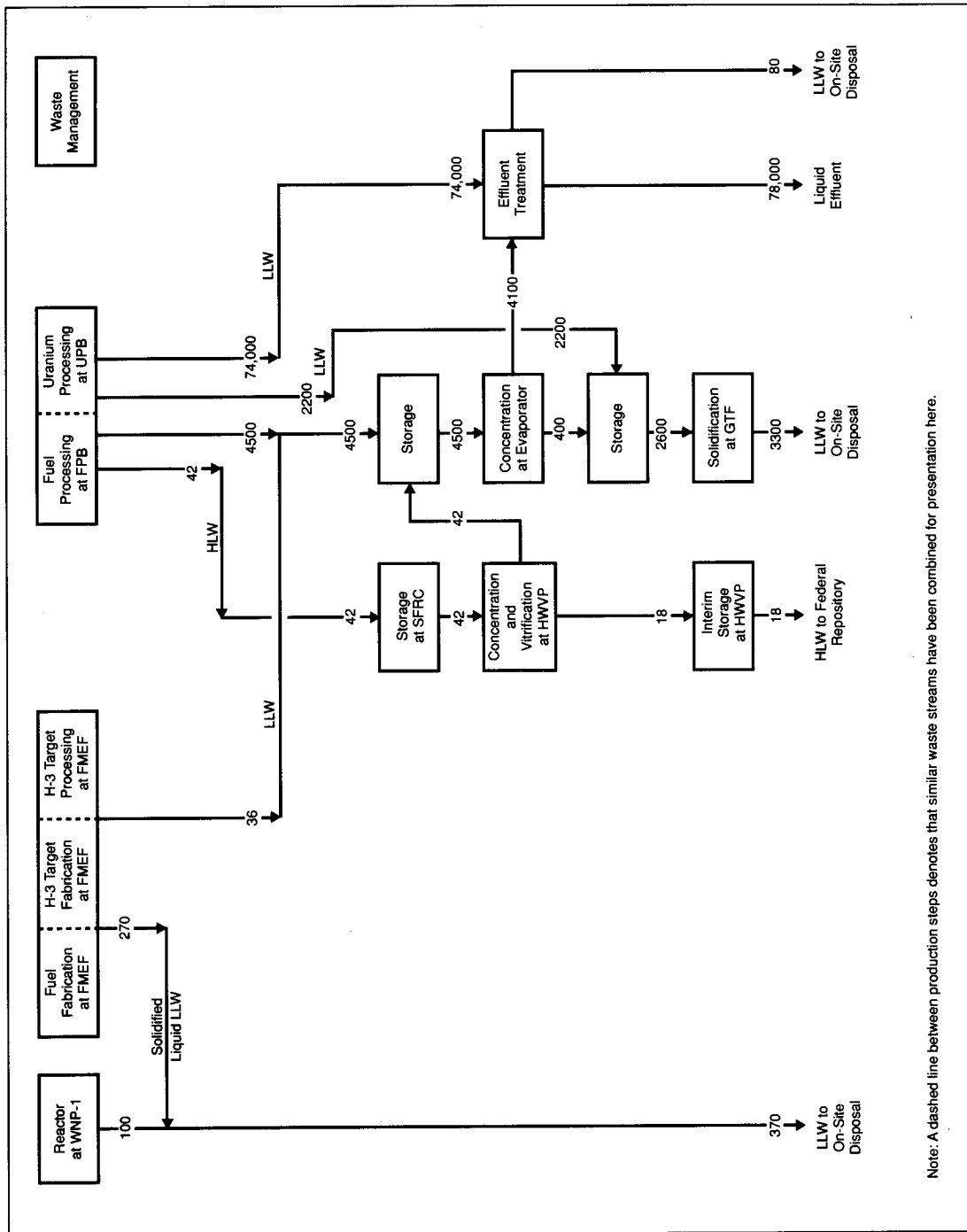


FIGURE M.10 Radioactive Liquid Waste Management for LWR Operations in the Tritium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1990a-b; Evans 1990)

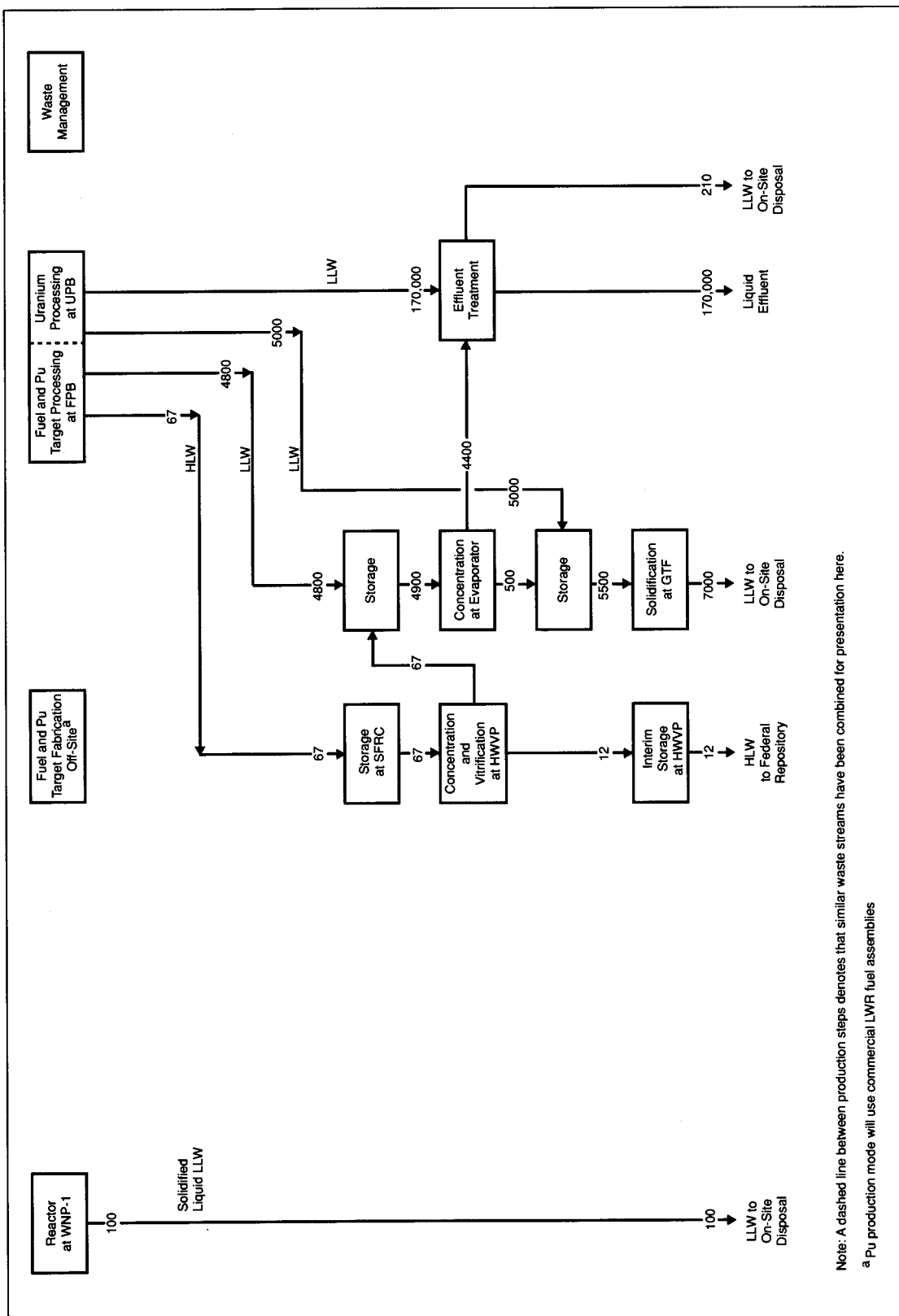


FIGURE M.11 Radioactive Liquid Waste Management for LWR Operations in the Plutonium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1990a-b; Evans 1990)

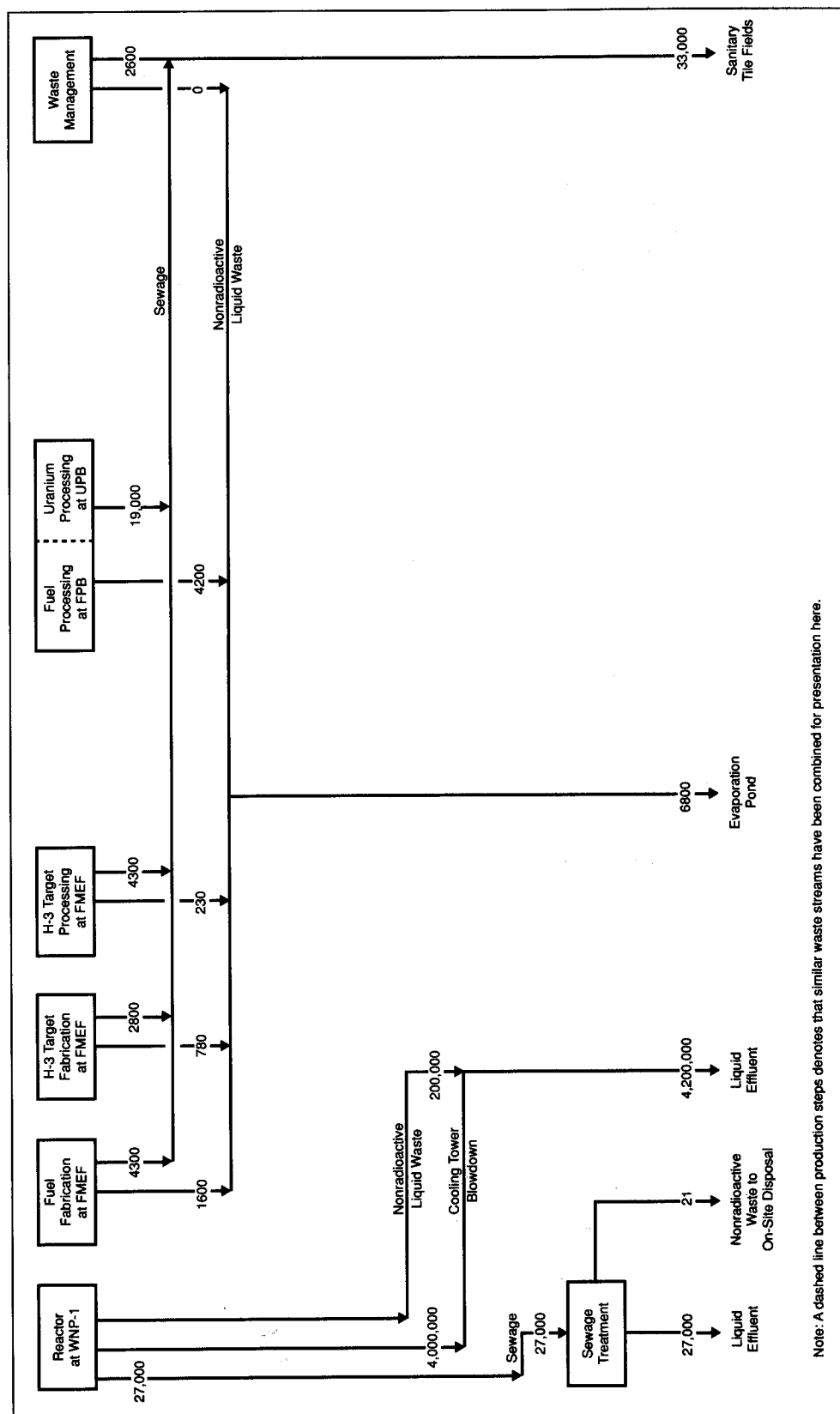


FIGURE M.12 Nonradioactive Liquid Waste Management for LWR Operations in the Tritium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1990a-b; Evans 1990)

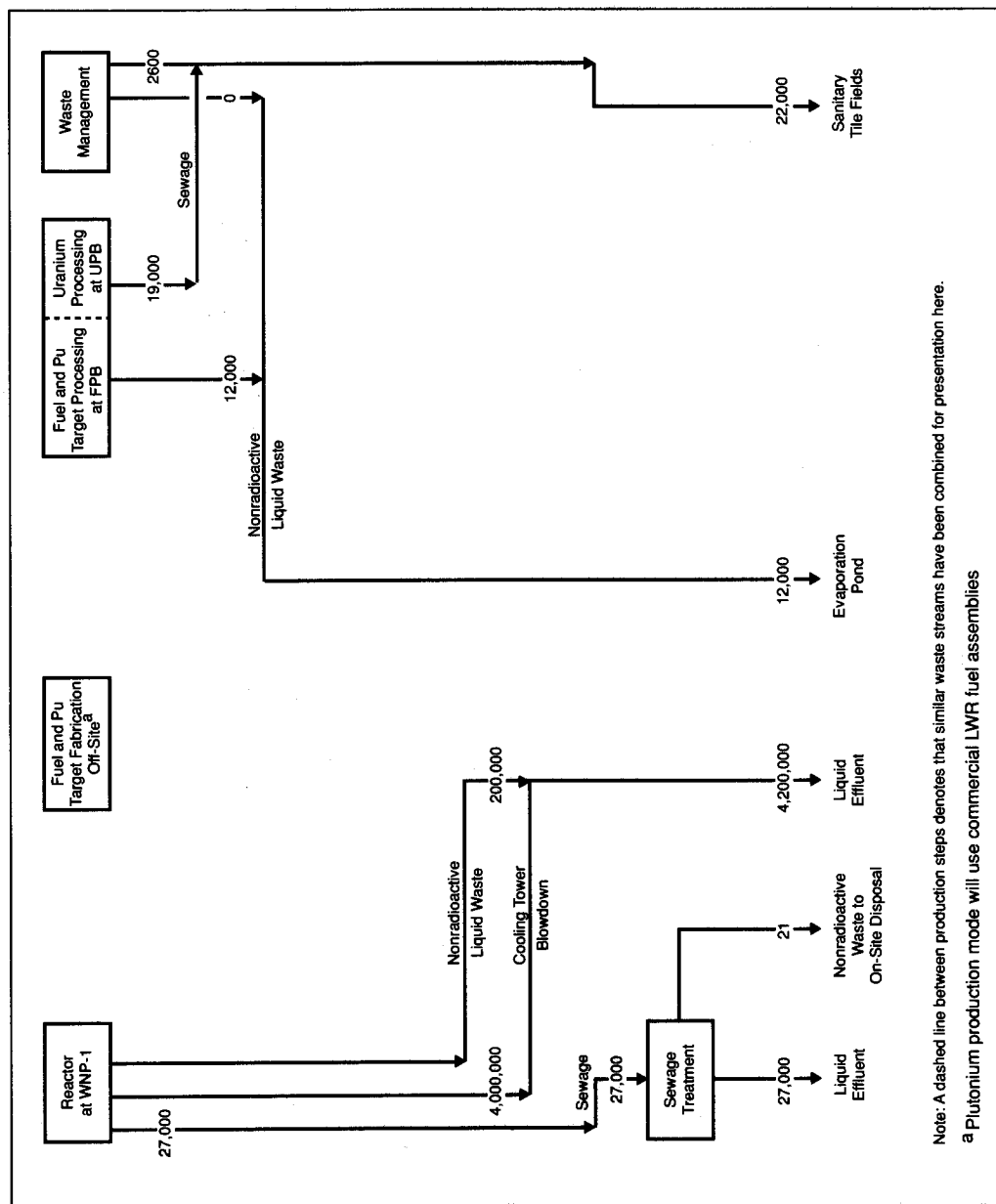


FIGURE M.13 Nonradioactive Liquid Waste Management for LWR Operations in the Plutonium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1990a-b; Evans 1990)

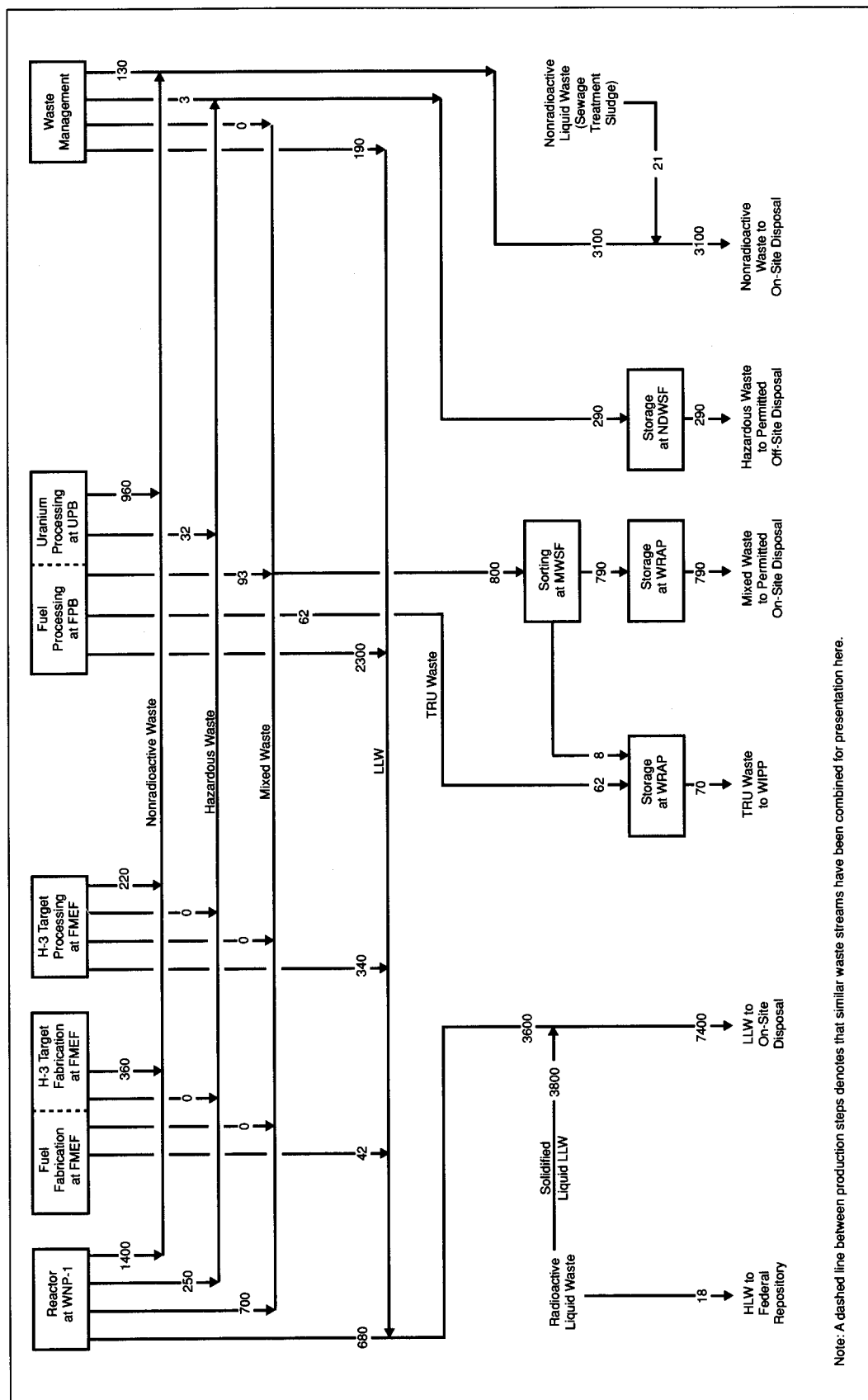


FIGURE M.14 Solid Waste Management for LWR Operations in the Tritium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1990a-b; Evans 1990)

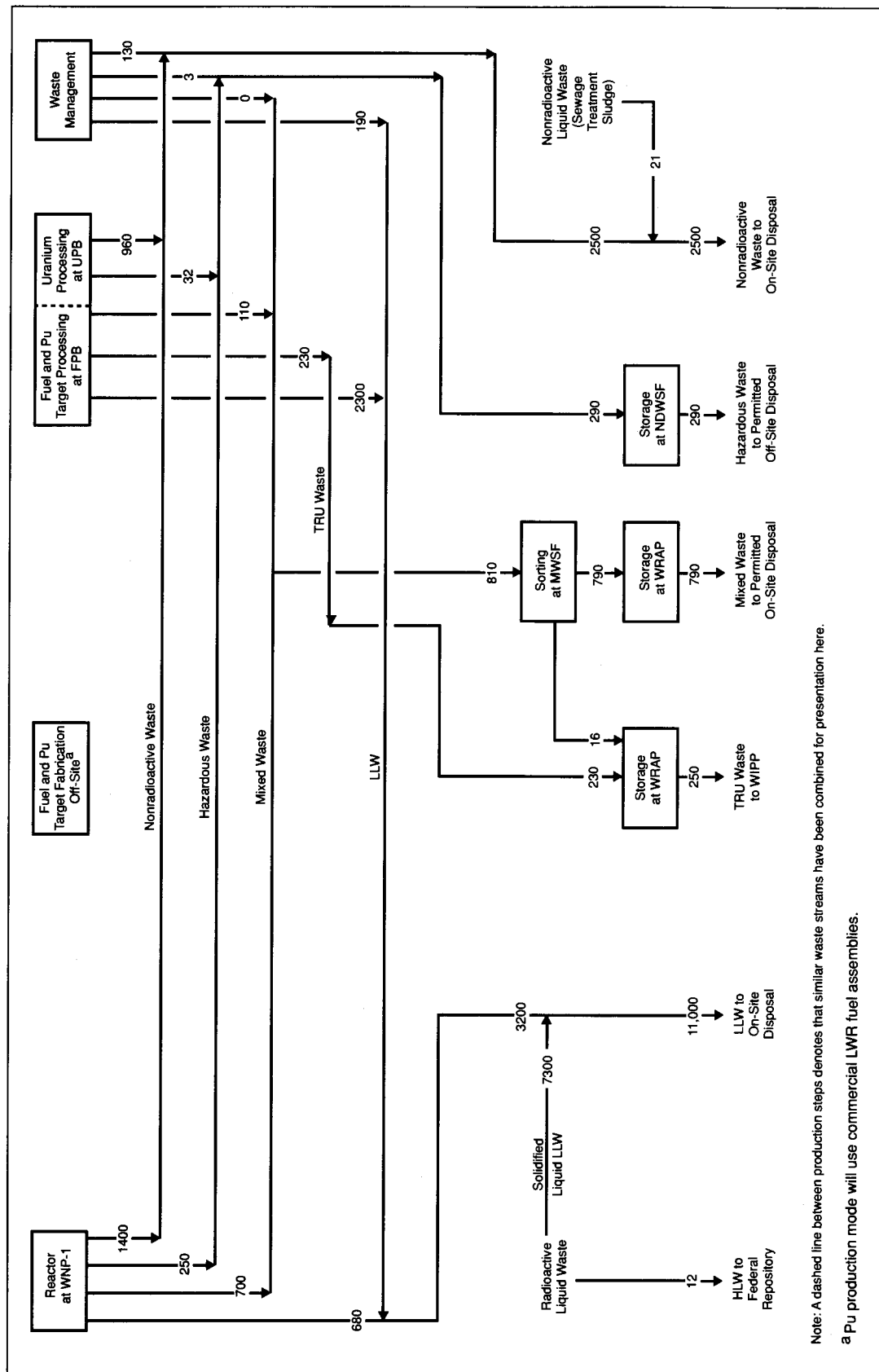


FIGURE M.15 Solid Waste Management for LWR Operations in the Plutonium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1990a-b; Evans 1990)

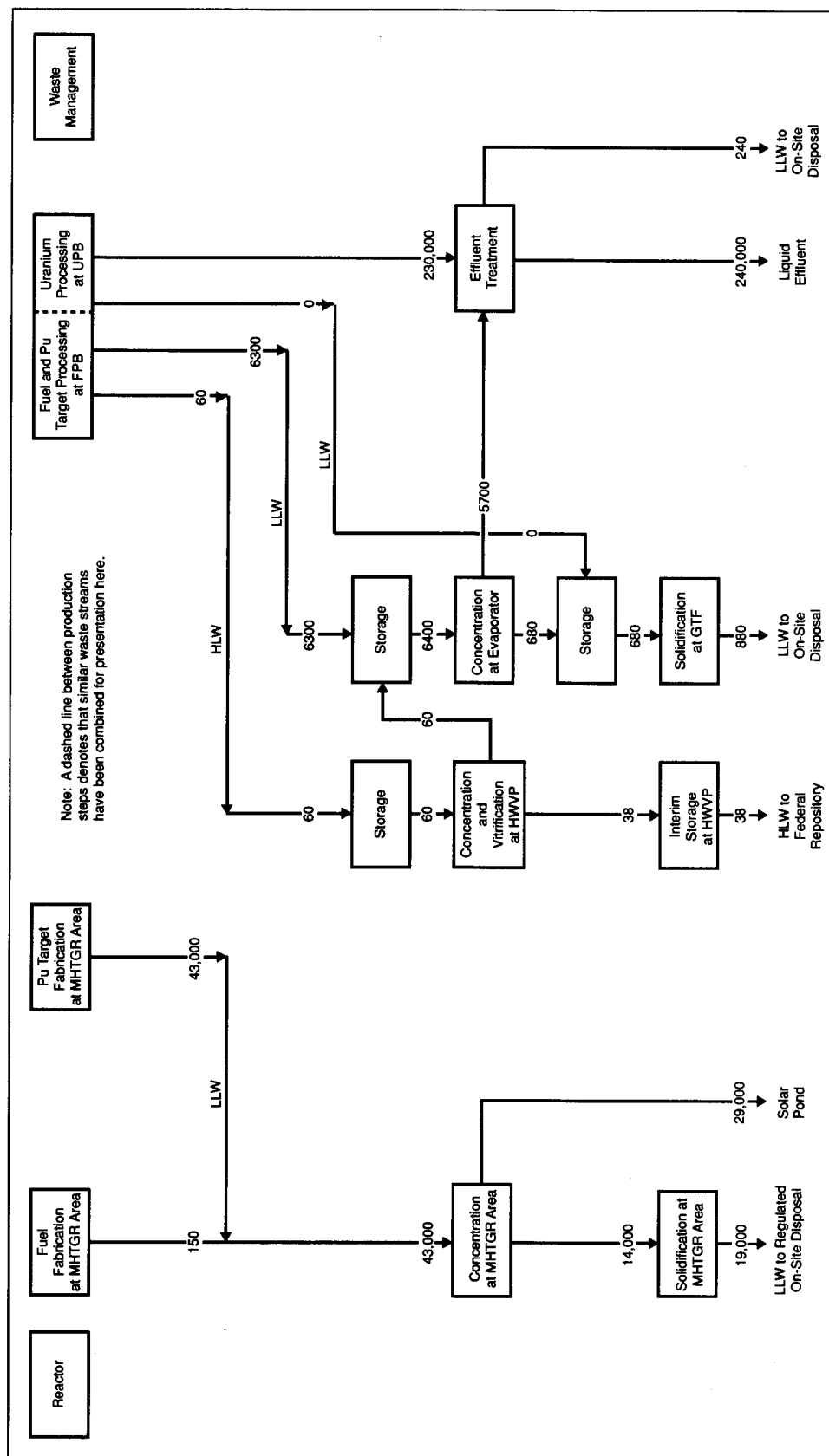


FIGURE M.17 Radioactive Liquid Waste Management for MHTGR Operations in the Plutonium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989c-d; Evans 1990)

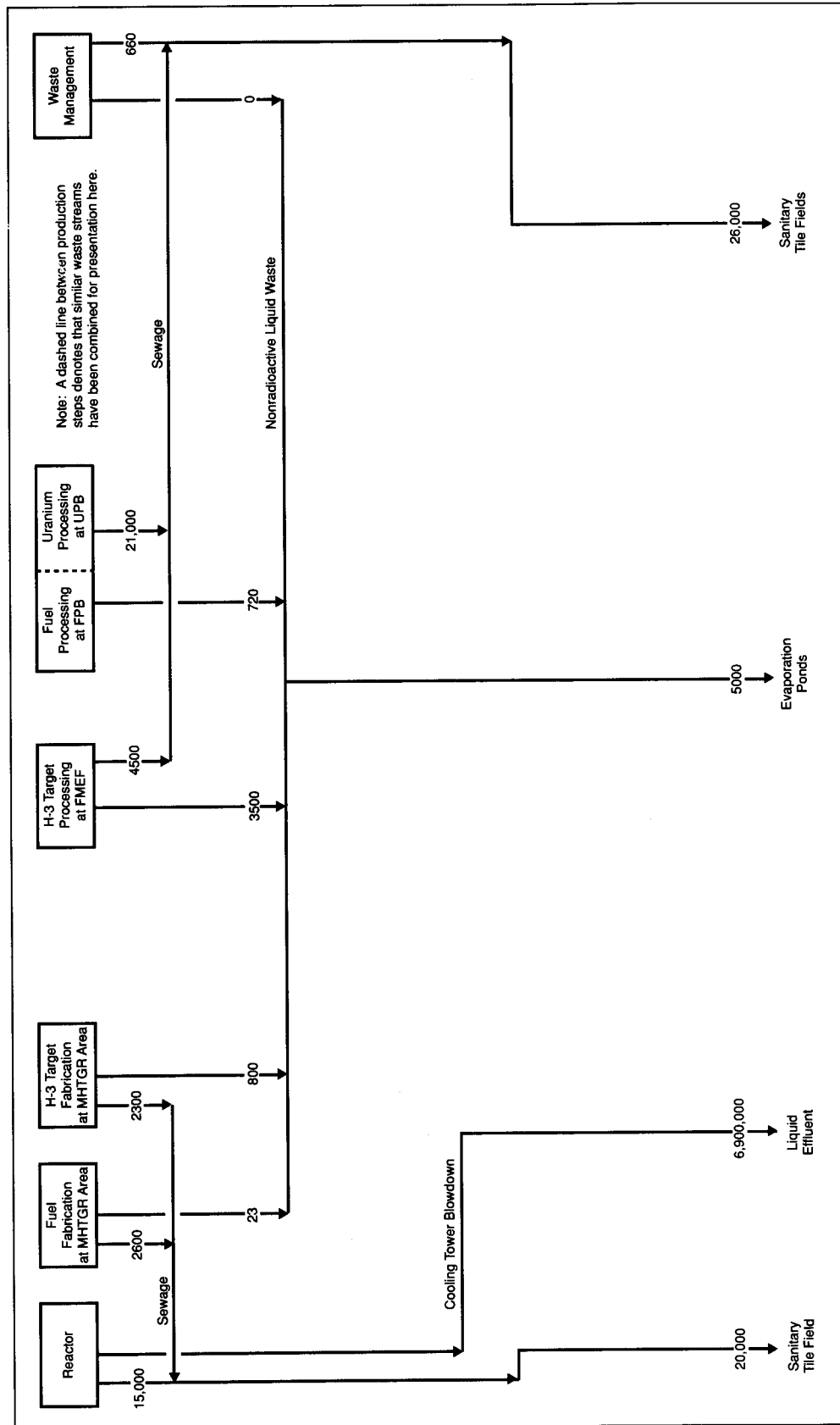


FIGURE M.18 Nonradioactive Liquid Waste Management for MHTGR Operations in the Tritium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989c-d; Evans 1990)

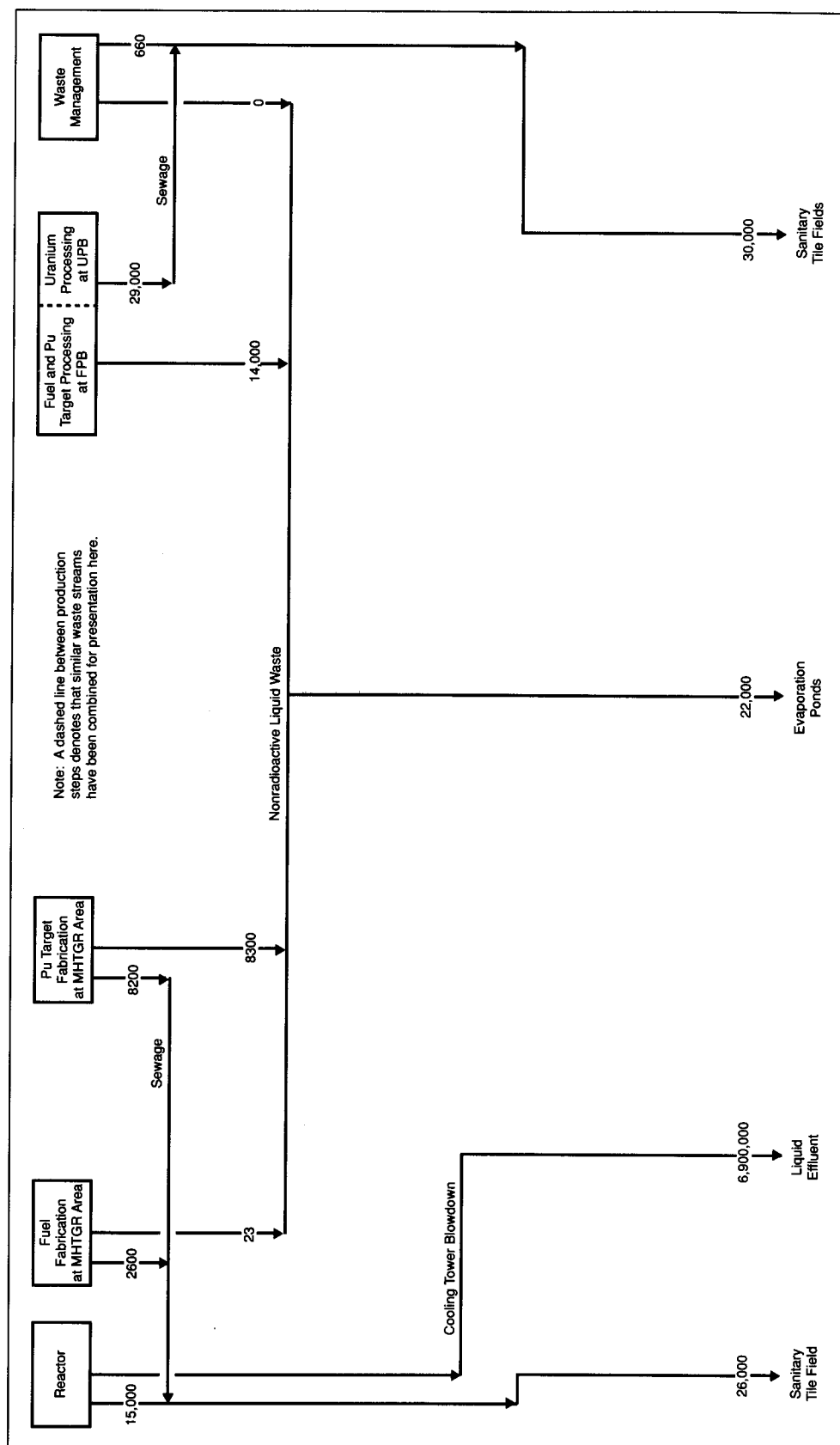


FIGURE M.19 Nonradioactive Liquid Waste Management for MHTGR Operations in the Plutonium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989c-d; Evans 1990)

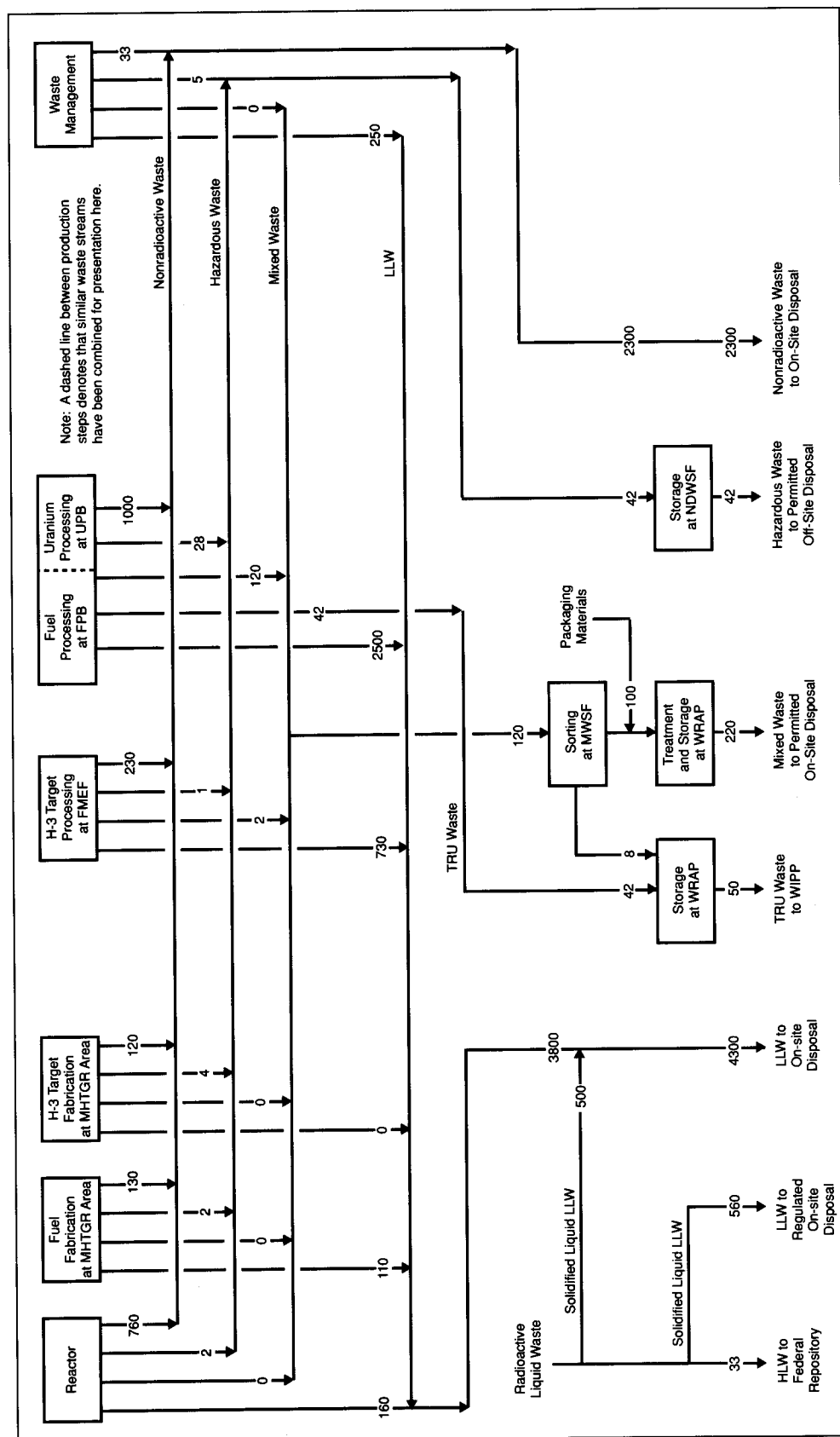


FIGURE M.20 Solid Waste Management for MHTGR Operations in the Tritium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989c-d; Evans 1990)

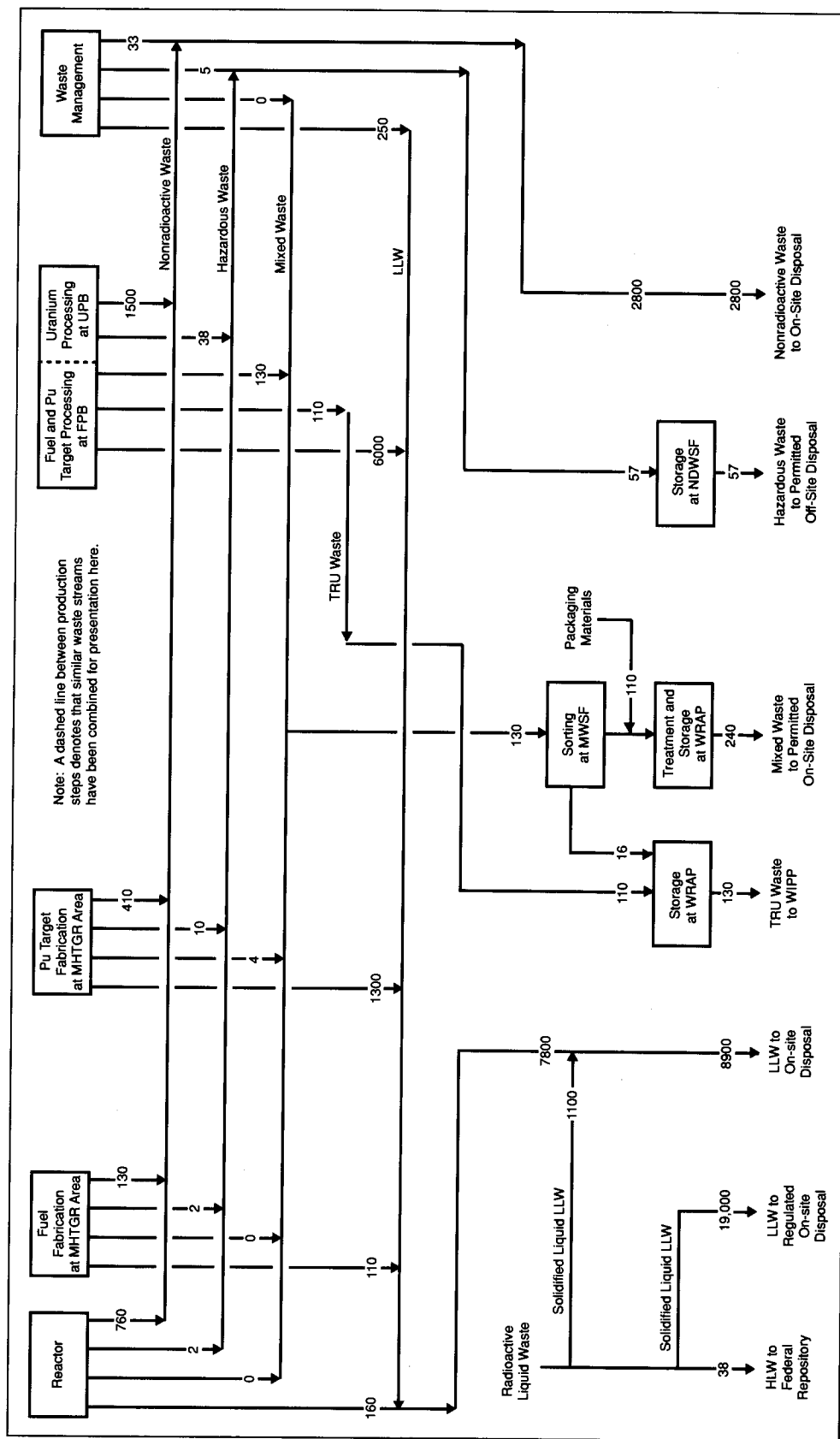


FIGURE M.21 Solid Waste Management for MHTGR Operations in the Plutonium Production Mode at Hanford (waste volumes expressed in cubic meters per year) (Sources: Based on data from WHC 1989c-d; Evans 1990)

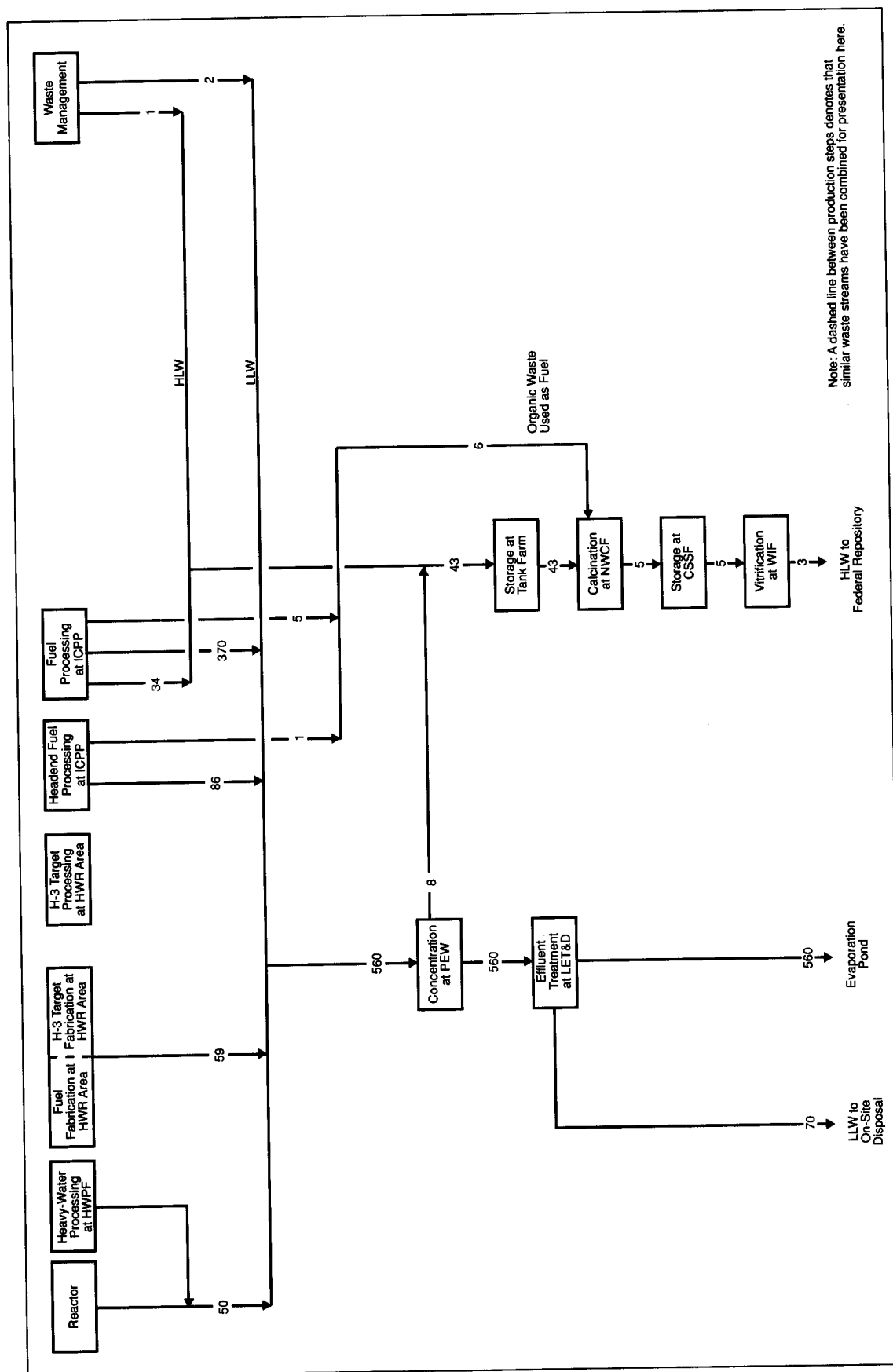


FIGURE M.22 Radioactive Liquid Waste Management for HWR Operations in the Tritium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989a; Bowman 1990; Bulmahn 1990)



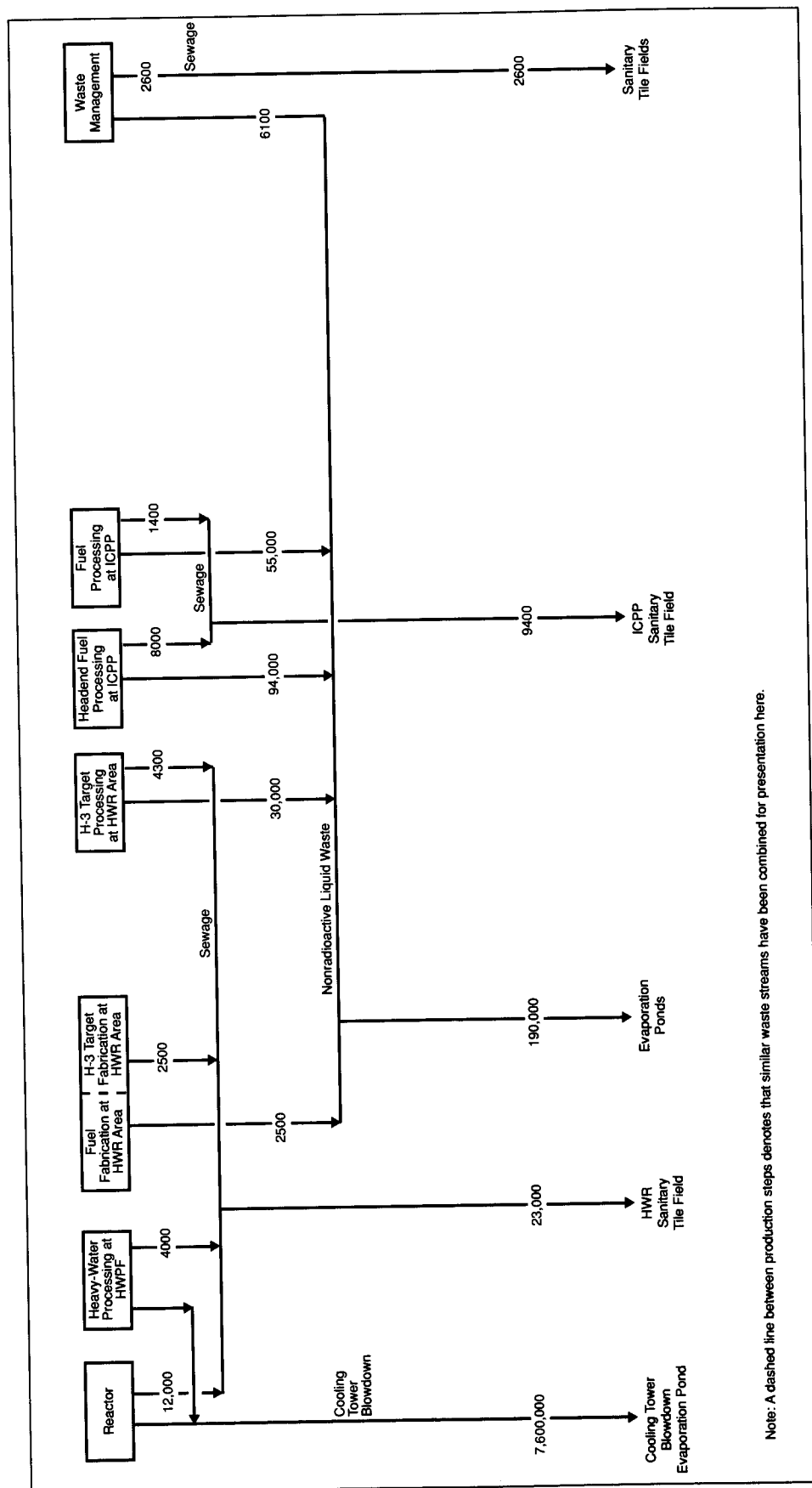


FIGURE M.24 Nonradioactive Liquid Waste Management for HWR Operations in the Tritium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989a; Bowman 1990; Bulmahn 1990)

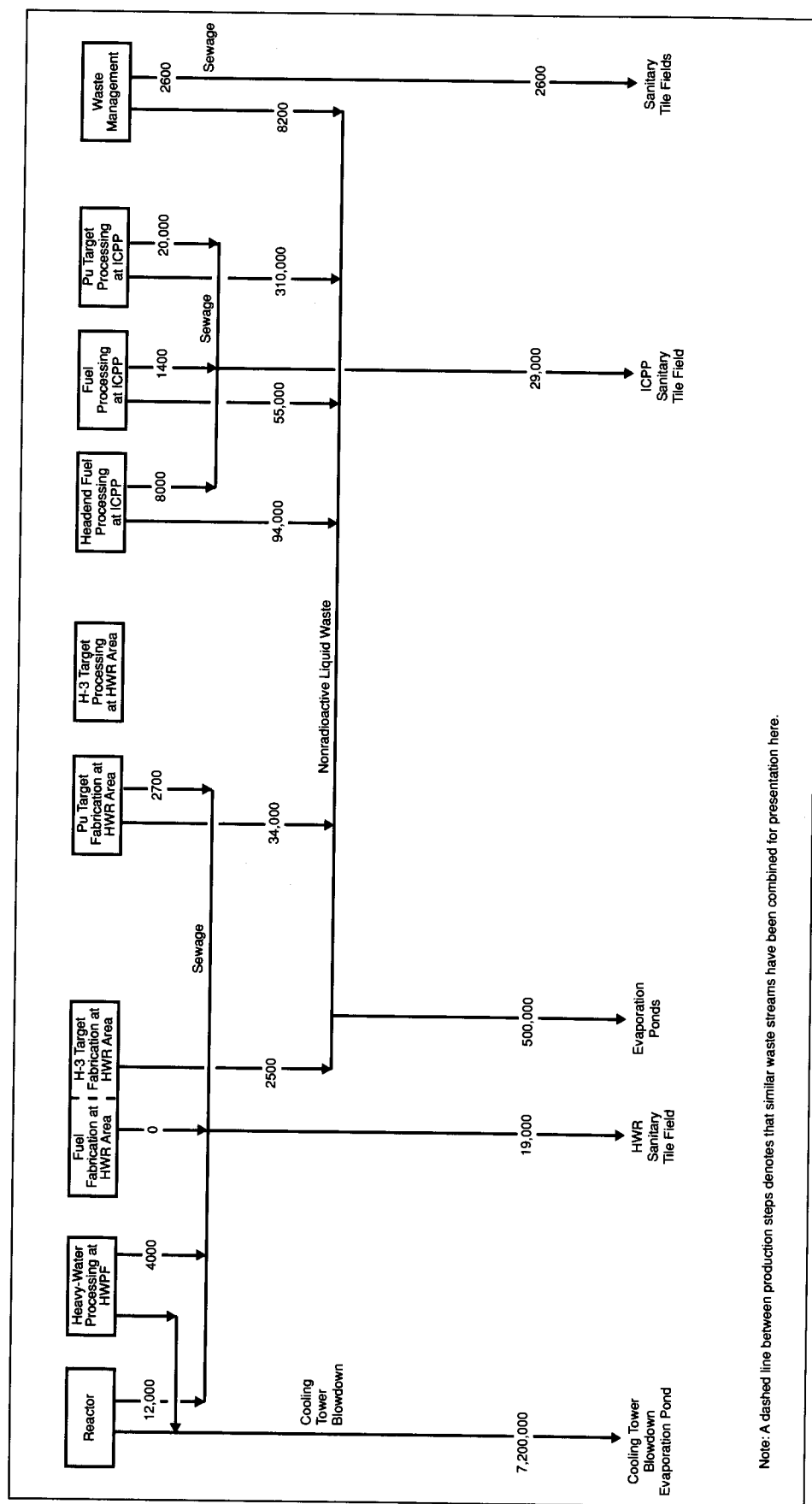


FIGURE M.25 Nonradioactive Liquid Waste Management for HWR Operations in the Plutonium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989a; Bowman 1990; Bulmahn 1990)

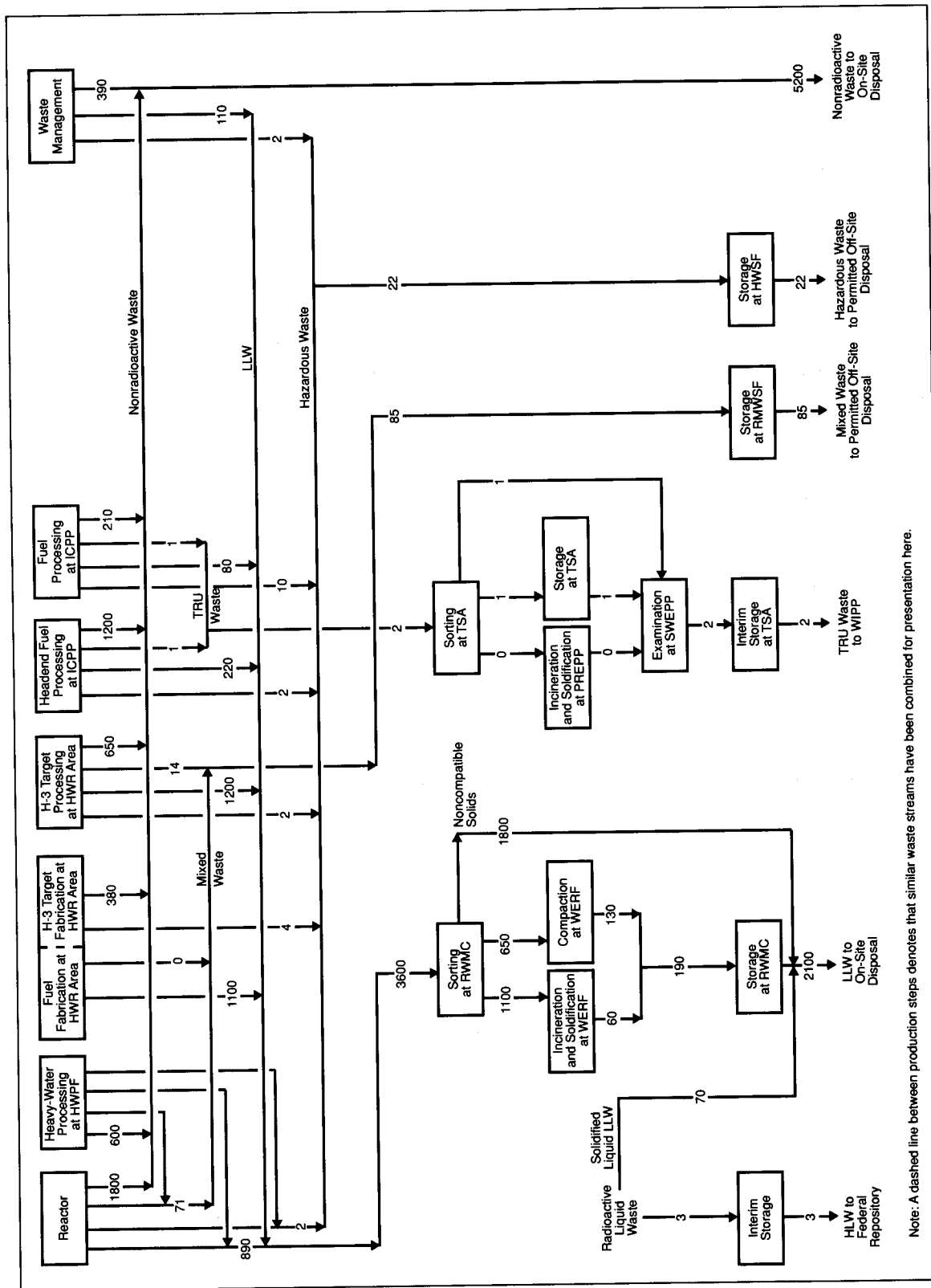


FIGURE M.26 Solid Waste Management for HWR Operations in the Tritium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989a; Bowman 1990; Bulmahn 1990)

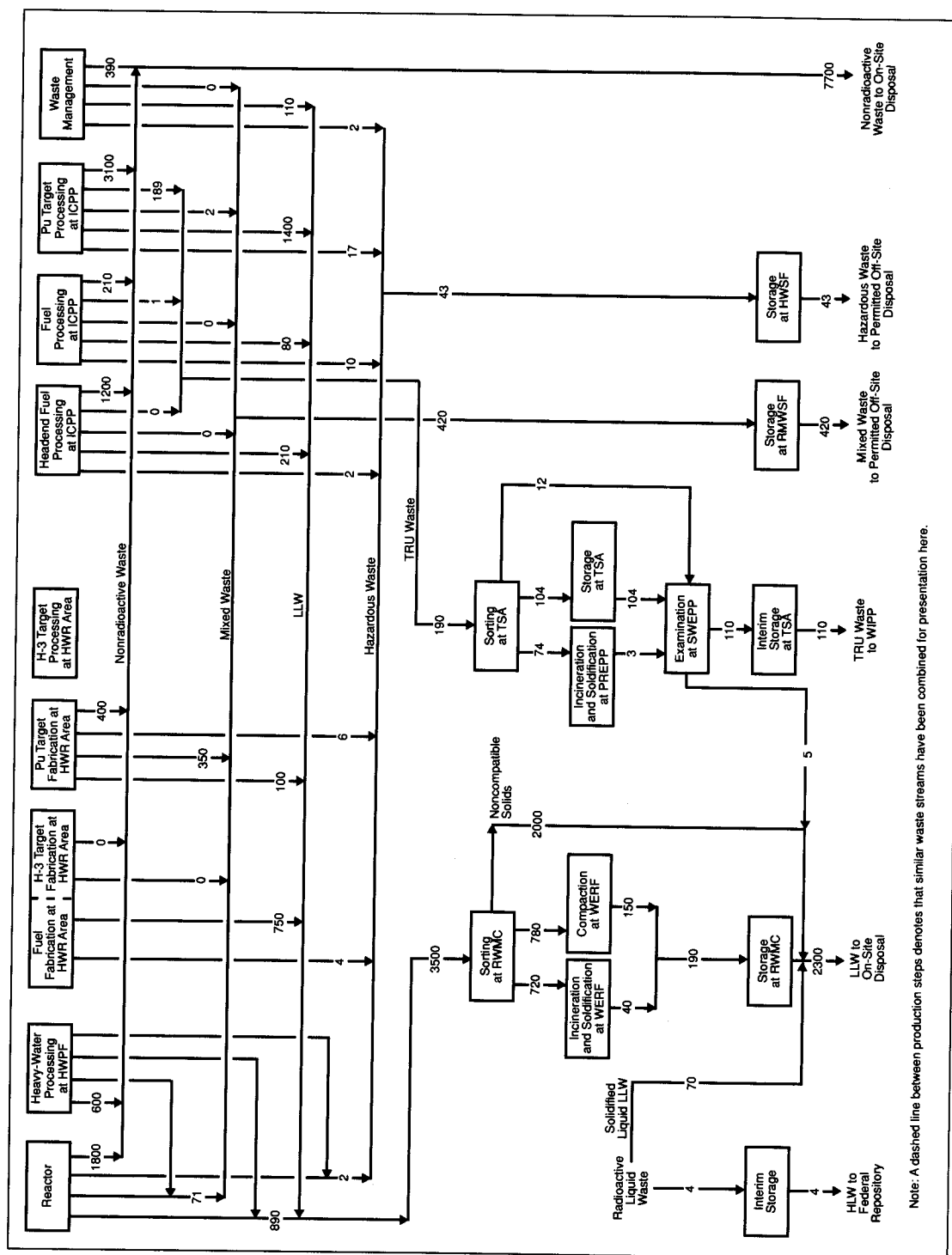


FIGURE M.27 Solid Waste Management for HWR Operations in the Plutonium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989a; Bowman 1990; Bulmahn 1990)

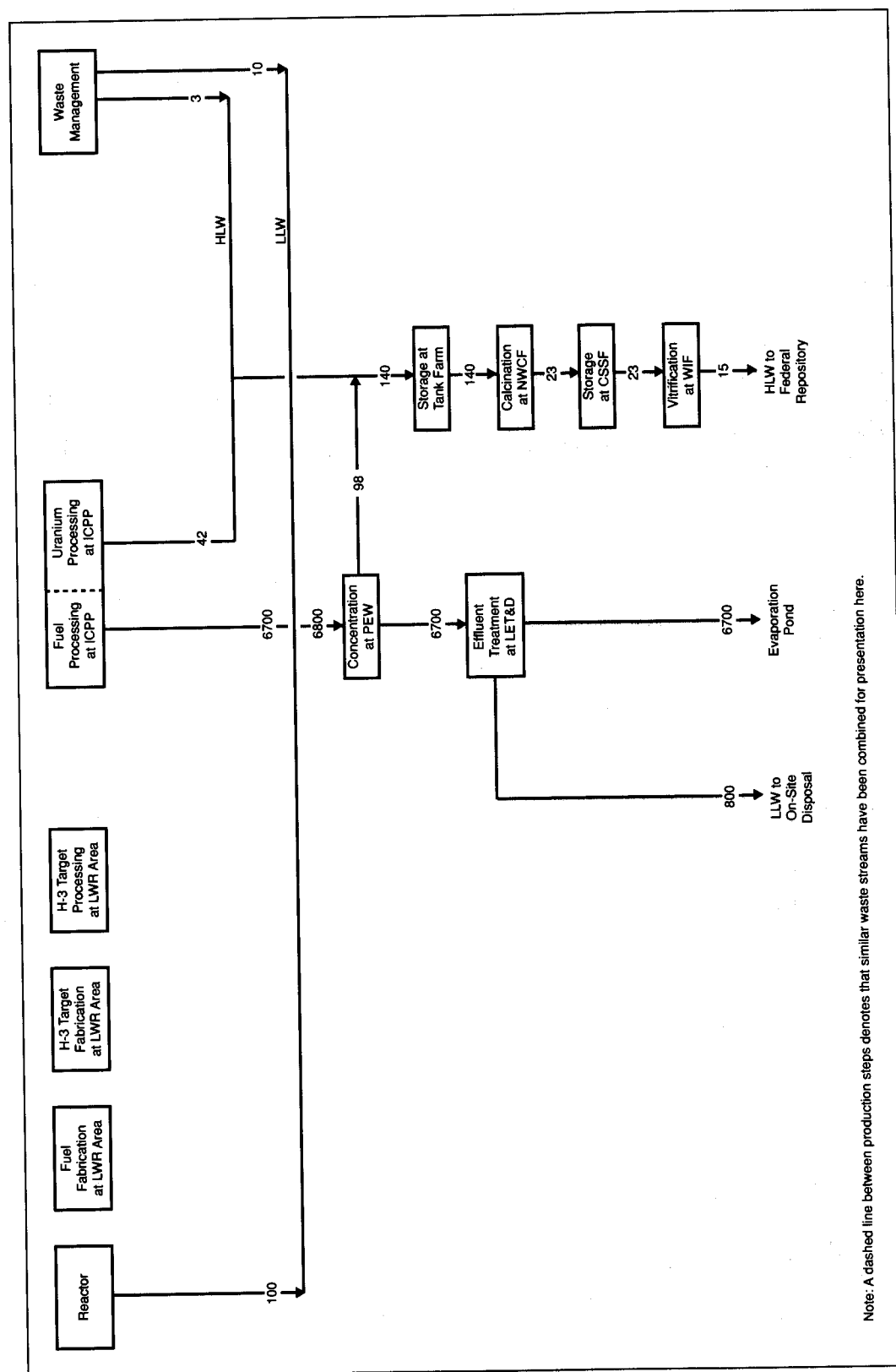


FIGURE M.28 Radioactive Liquid Waste Management for LWR Operations in the Tritium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989b; Bowman 1990; Bulmahn 1990)

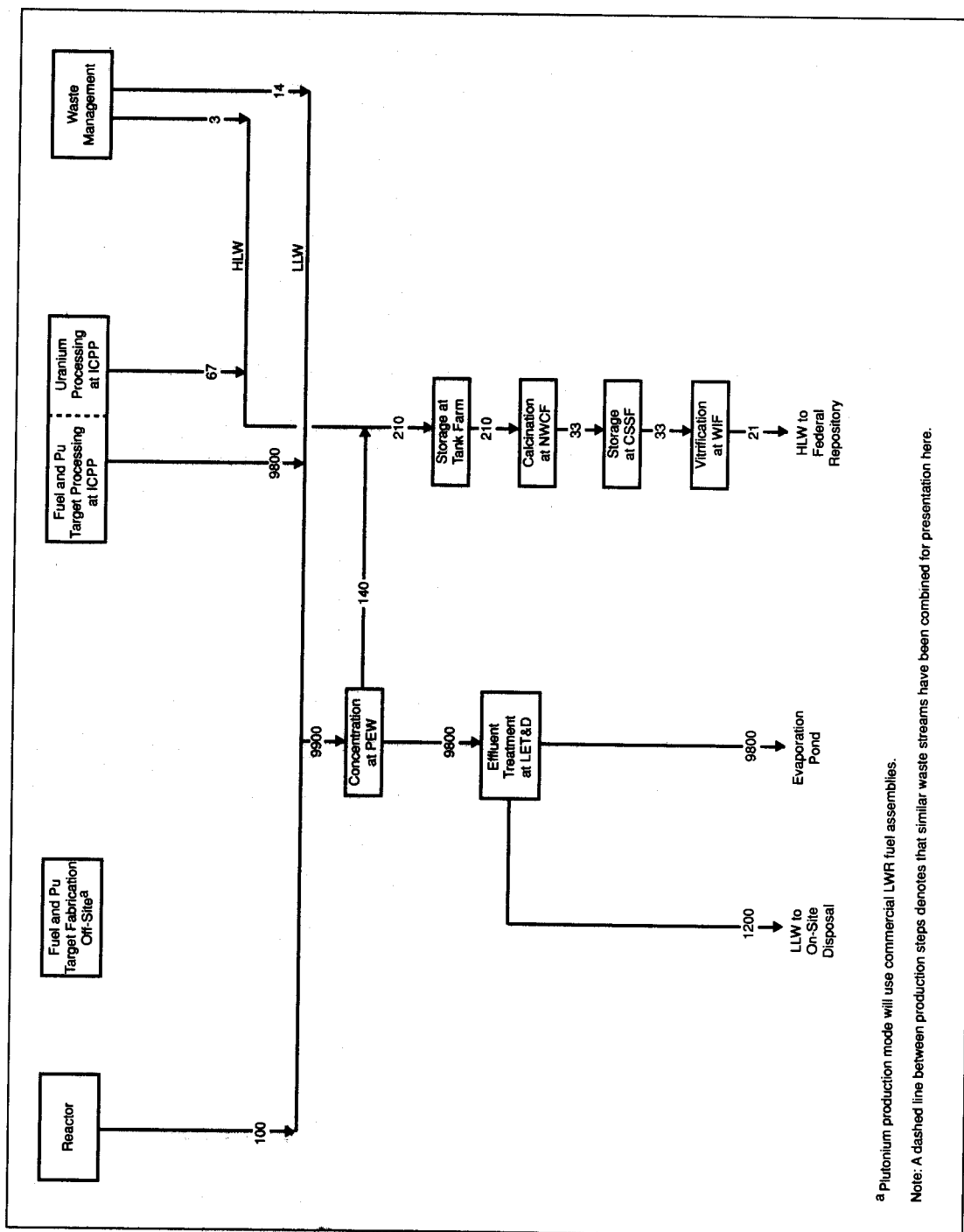


FIGURE M.29 Radioactive Liquid Waste Management for LWR Operations in the Plutonium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989b; Bowman 1990; Bulmahn 1990)

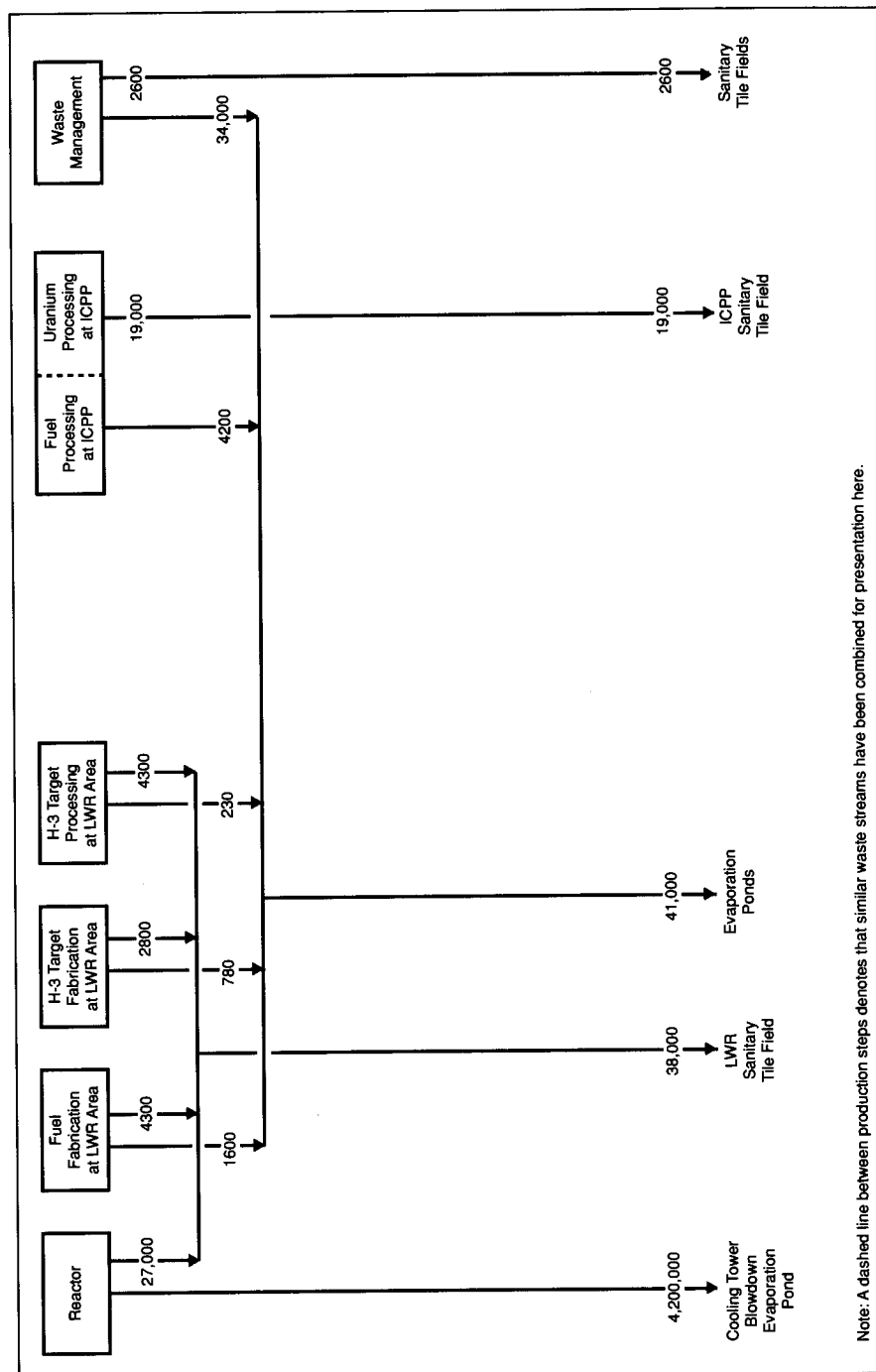


FIGURE M.30 Nonradioactive Liquid Waste Management for LWR Operations in the Tritium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989b; Bowman 1990; Bulmahn 1990)

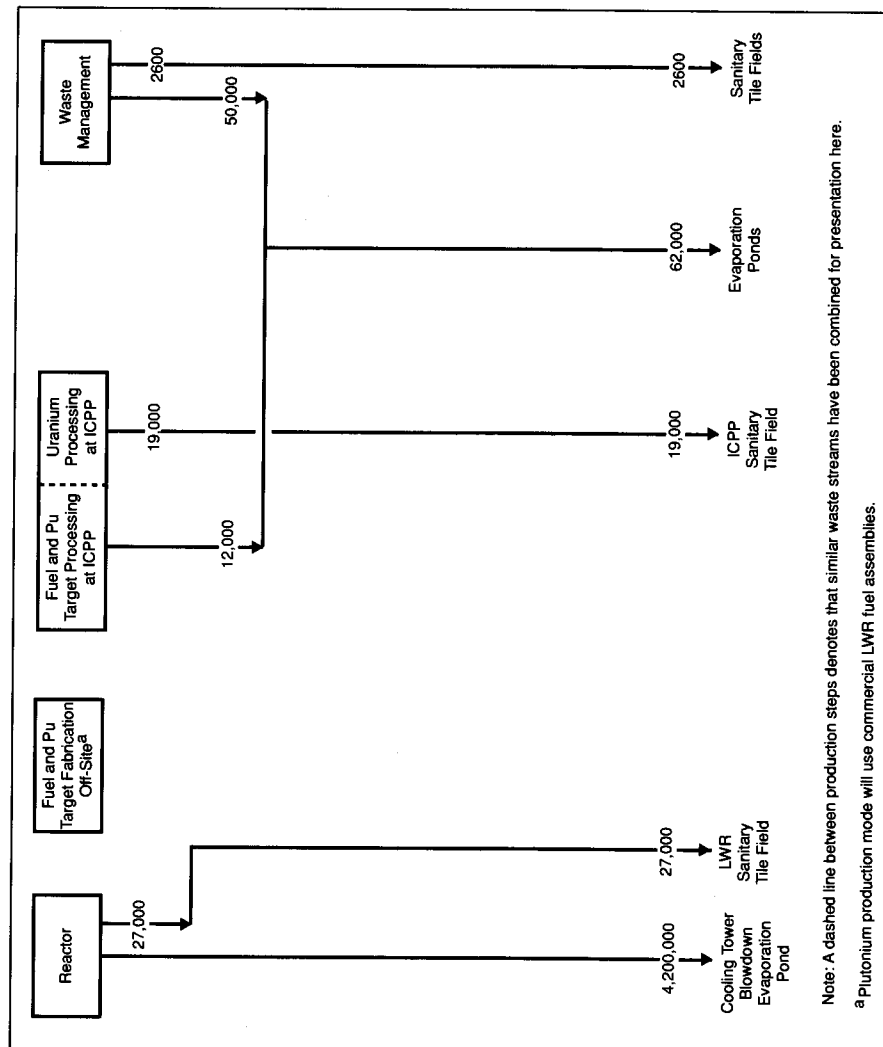


FIGURE M.31 Nonradioactive Liquid Waste Management for LWR Operations in the Plutonium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989b; Bowman 1990; Bulmahn 1990)

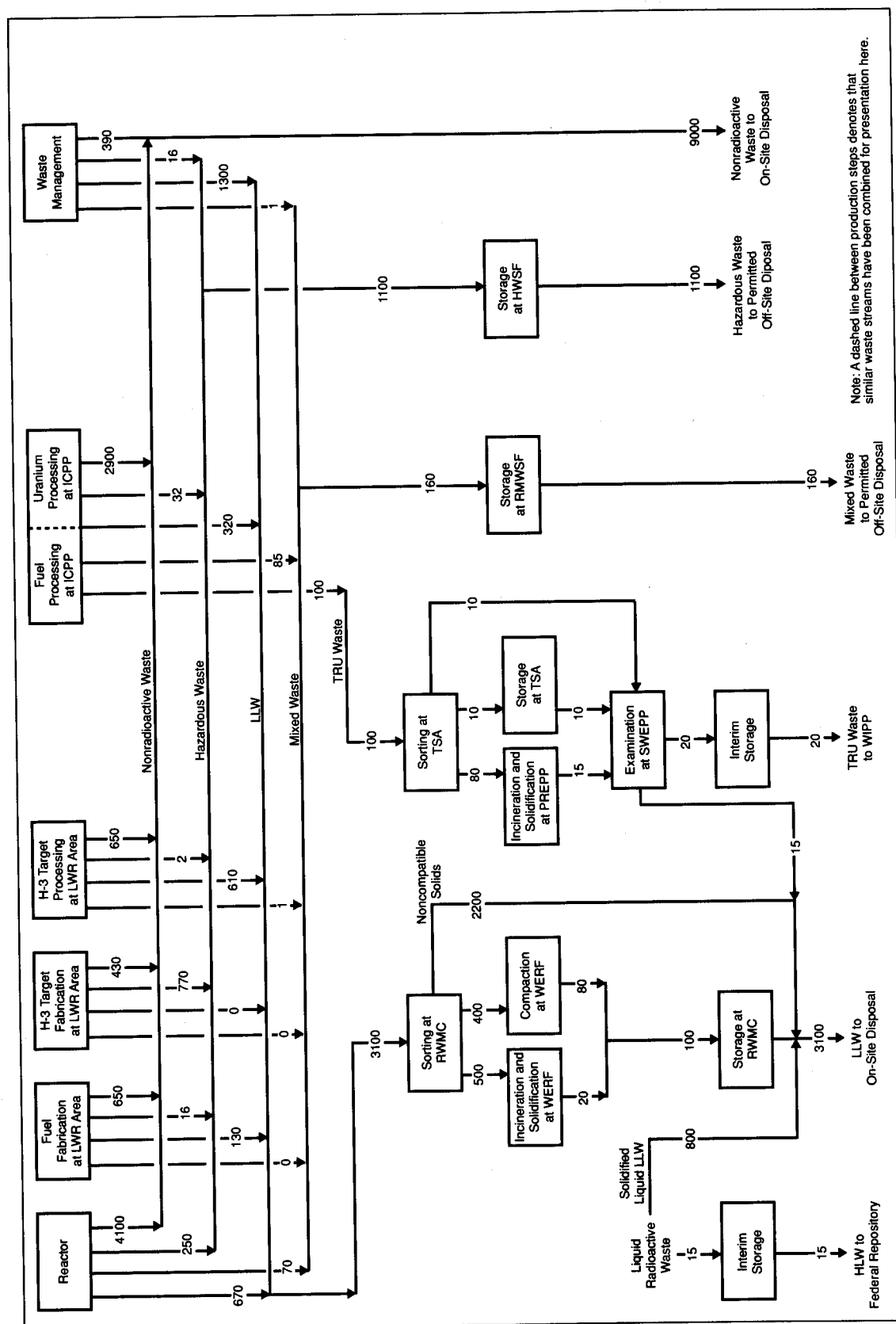


FIGURE M.32 Solid Waste Management for LWR Operations in the Tritium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989b; Bowman 1990; Bulmahn 1990)

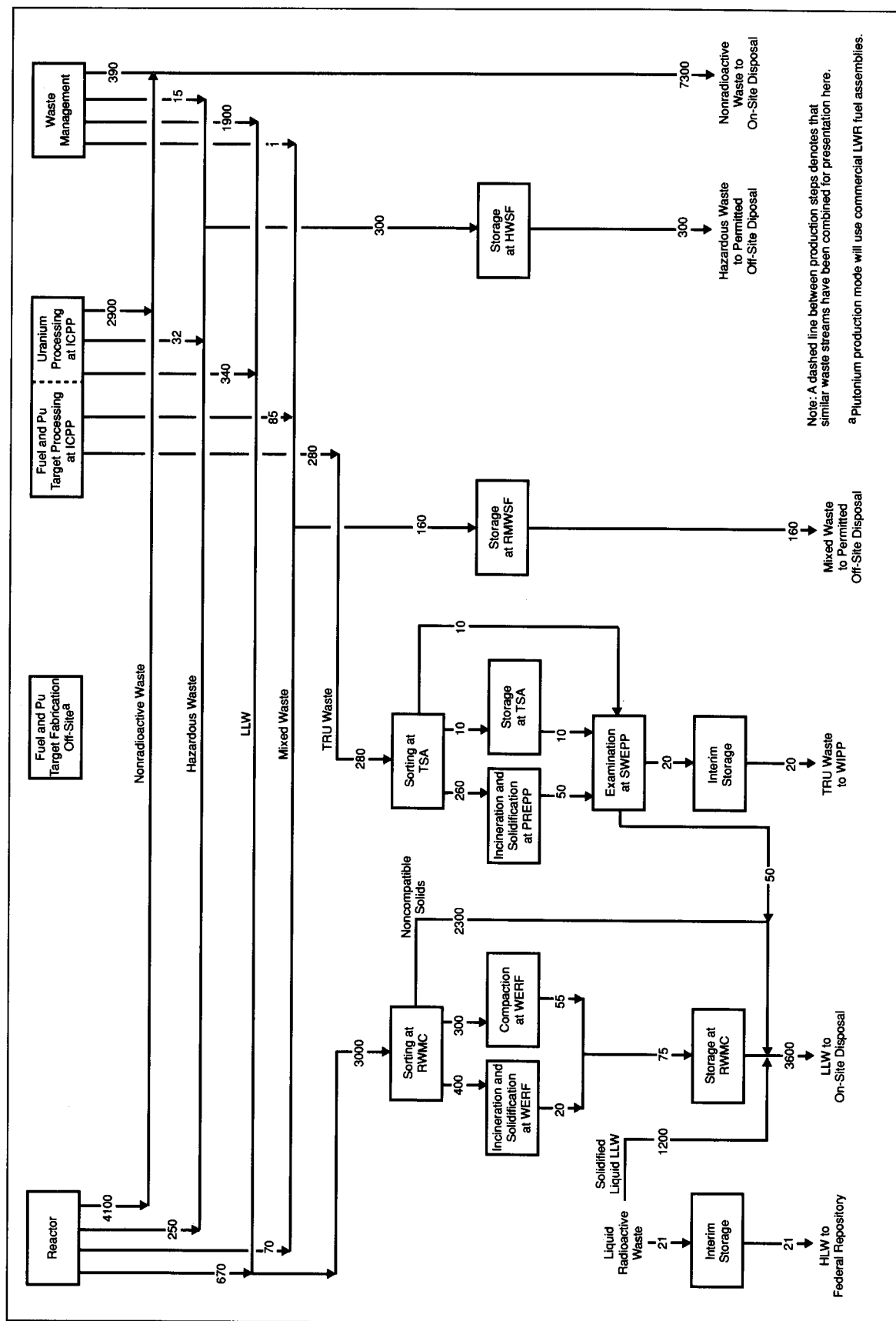


FIGURE M.33 Solid Waste Management for LWR Operations in the Plutonium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989b; Bowman 1990; Bulmahn 1990)

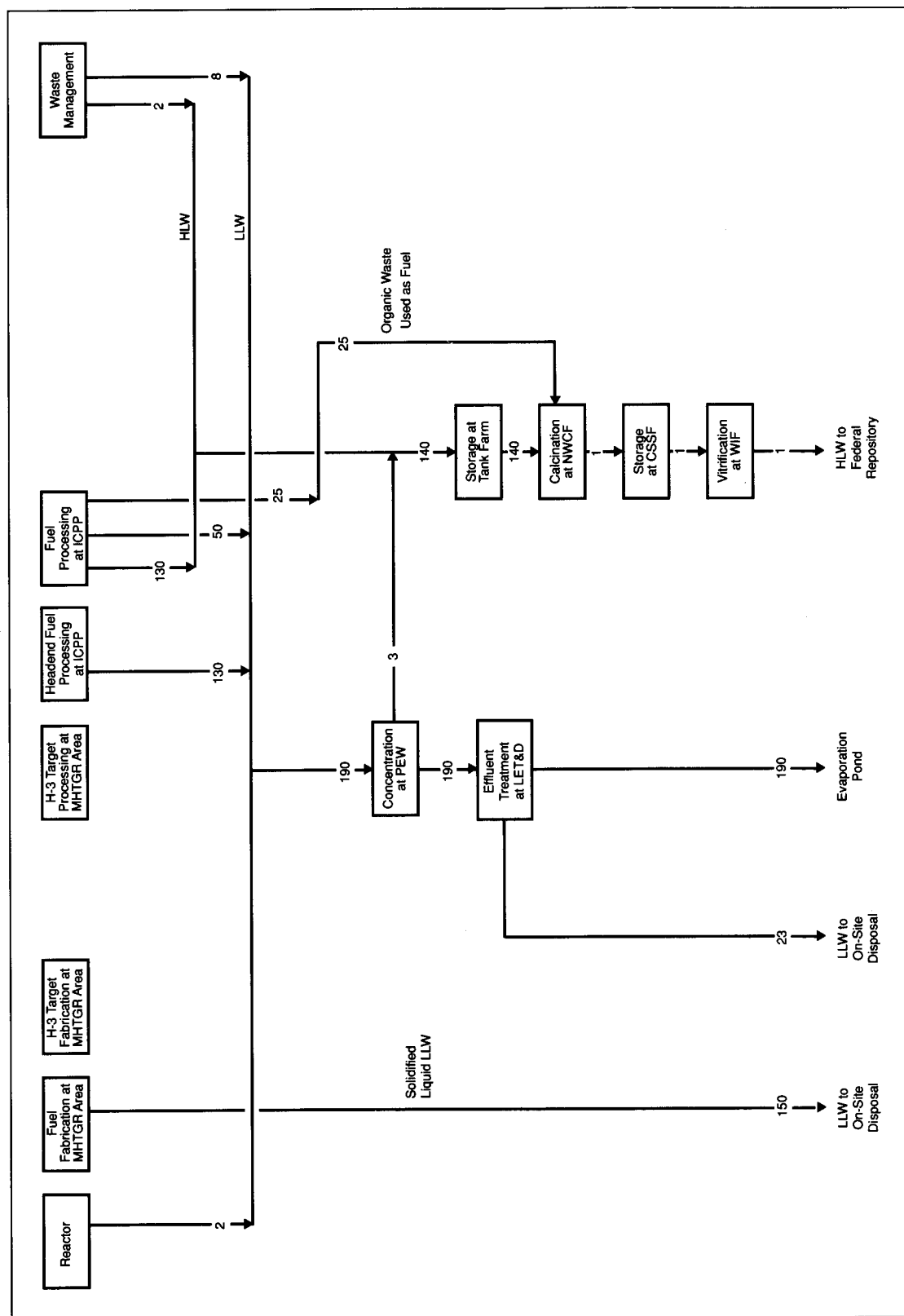


FIGURE M.34 Radioactive Liquid Waste Management for MHTGR Operations in the Tritium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989c; Bowman 1990; Bulmahn 1990)

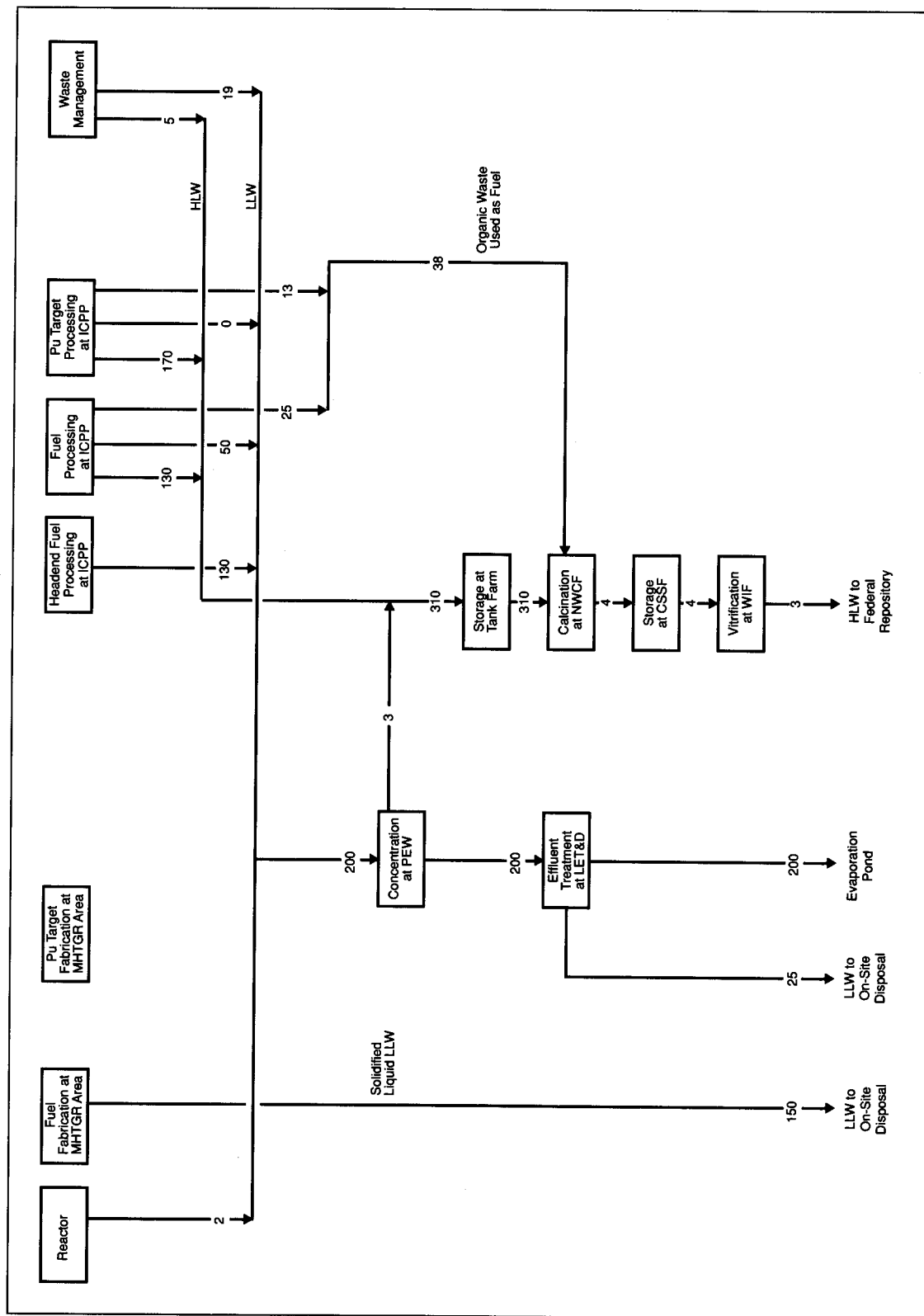


FIGURE M.35 Radioactive Liquid Waste Management for MHTGR Operations in the Plutonium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989c; Bowman 1990; Bulmahn 1990)

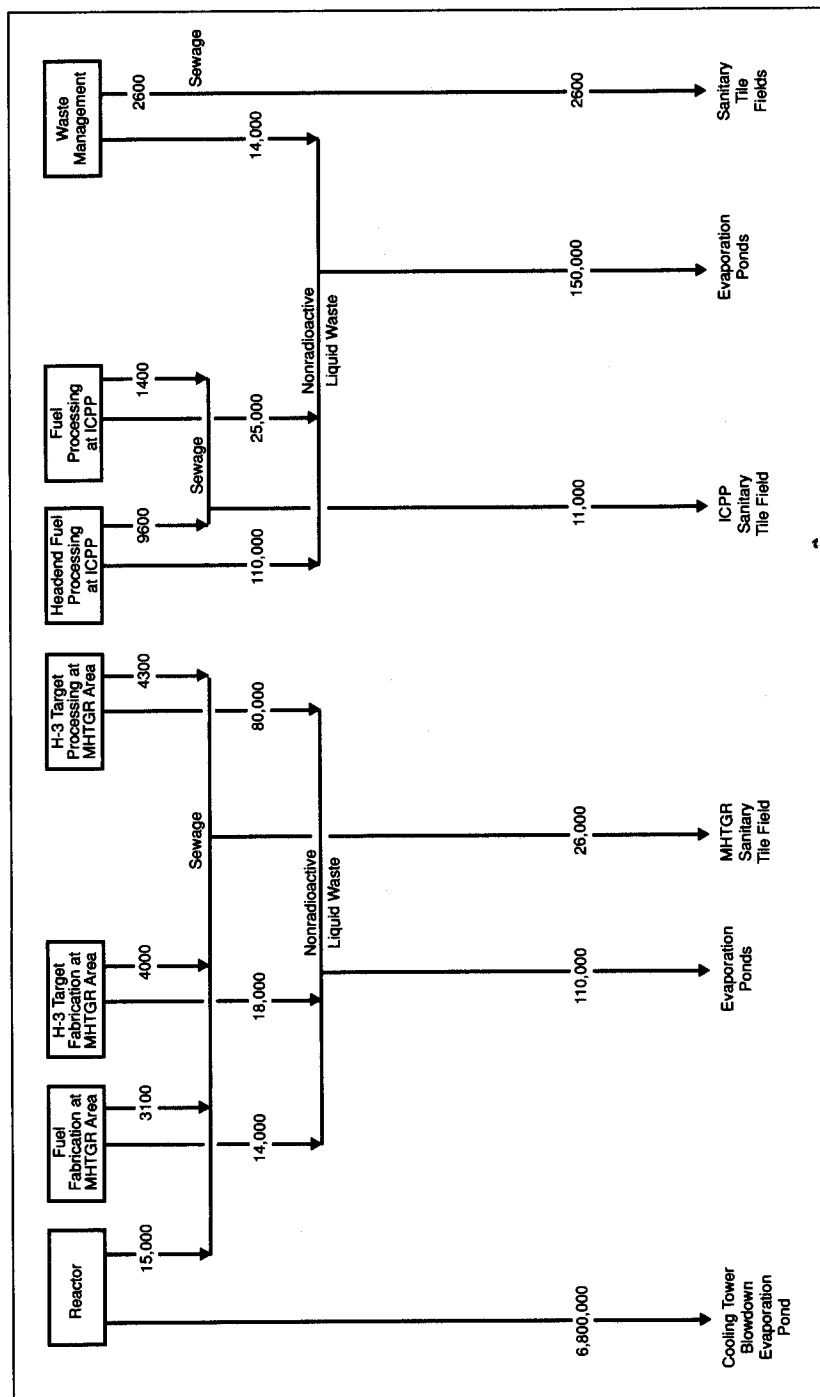


FIGURE M.36 Nonradioactive Liquid Waste Management for MHTGR Operations in the Tritium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989c; Bowman 1990; Bulmahn 1990)

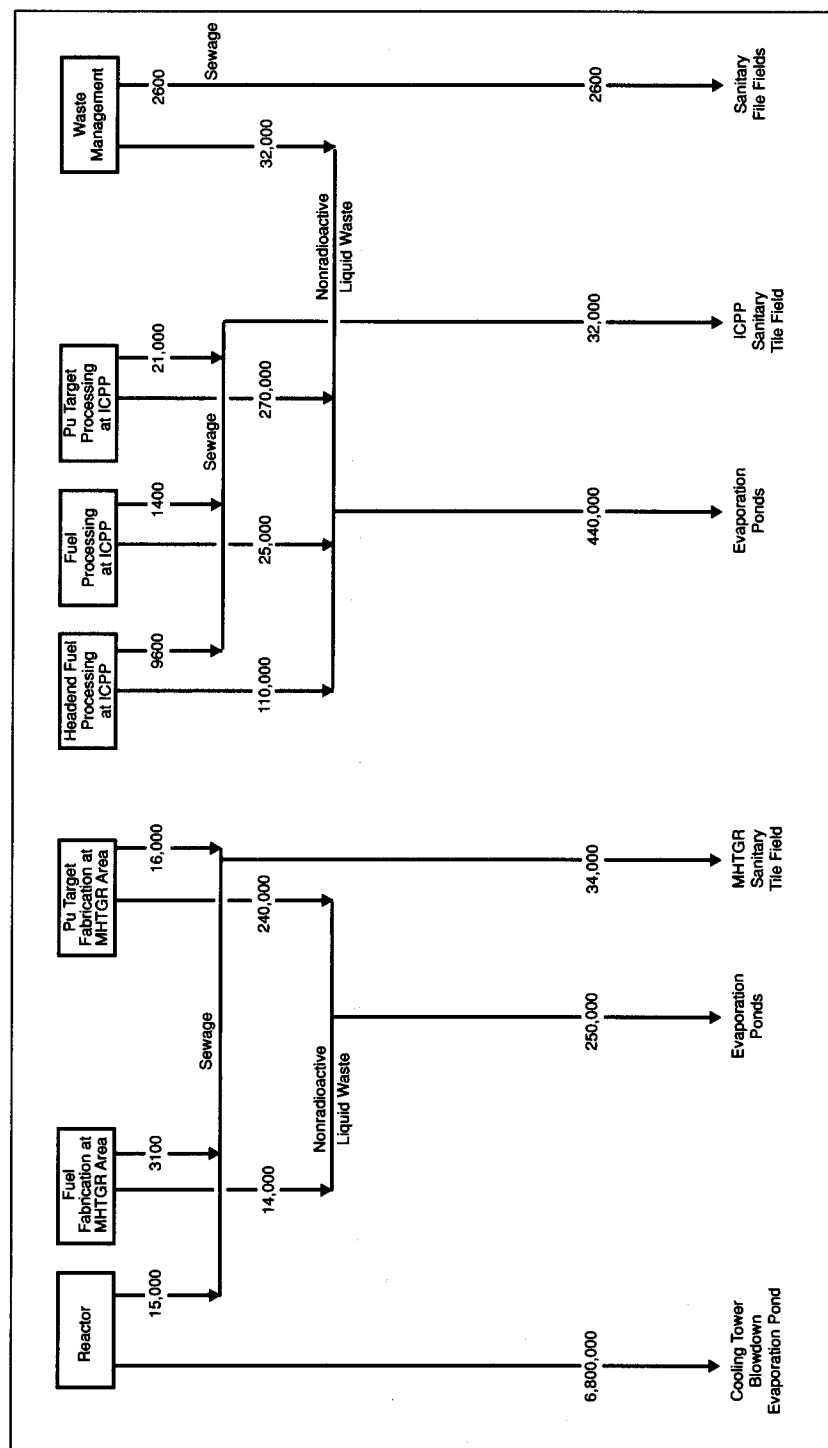


FIGURE M.37 Nonradioactive Liquid Waste Management for MHTGR Operations in the Plutonium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989c; Bowman 1990; Bulmahn 1990)

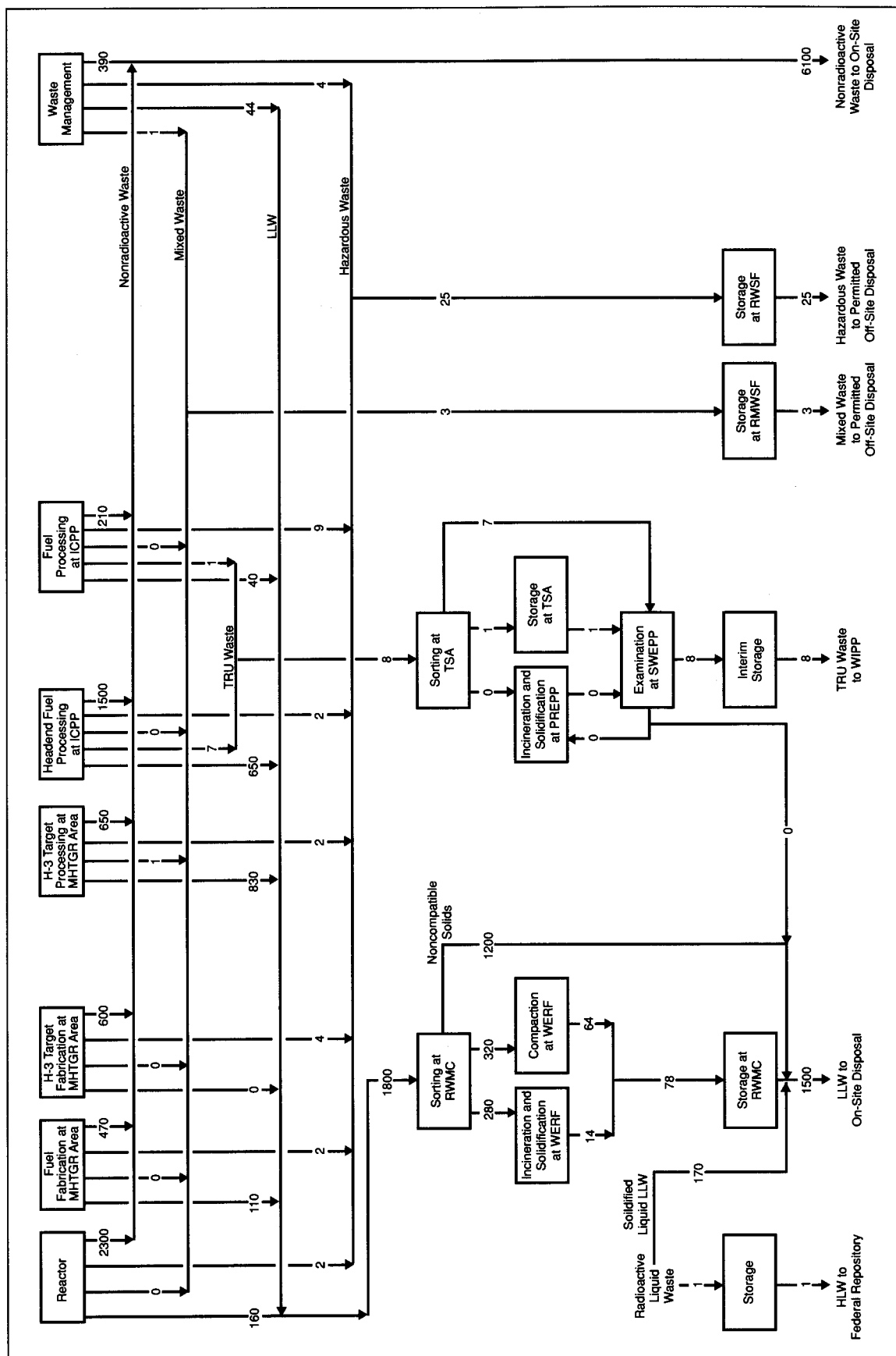


FIGURE M.38 Solid Waste Management for MHTGR Operations in the Tritium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989c; Bowman 1990; Bulmahn 1990)

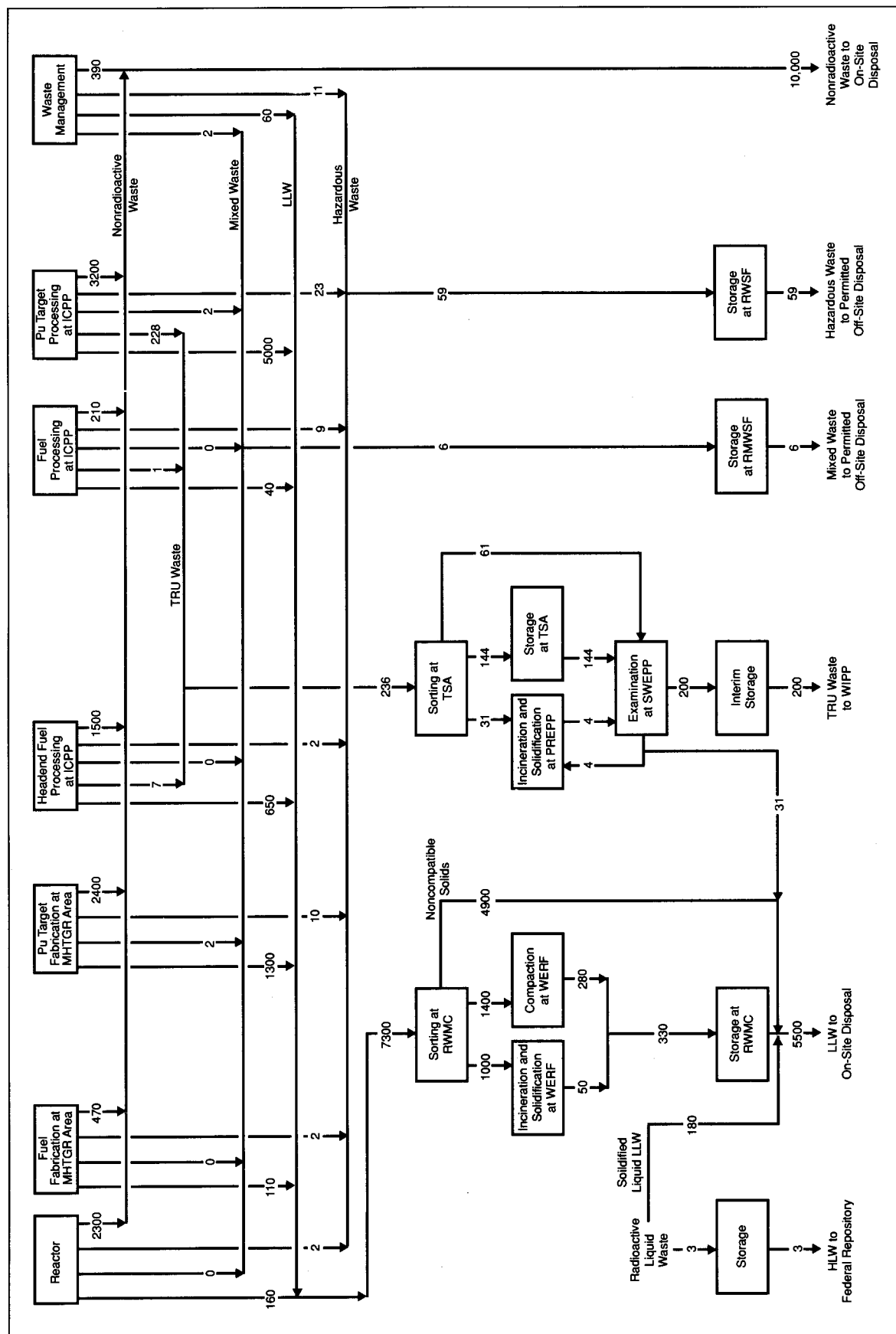


FIGURE M.39 Solid Waste Management for MHTGR Operations in the Plutonium Production Mode at INEL (waste volumes expressed in cubic meters per year) (Sources: Based on data from EG&G 1989c; Bowman 1990; Bulmahn 1990)

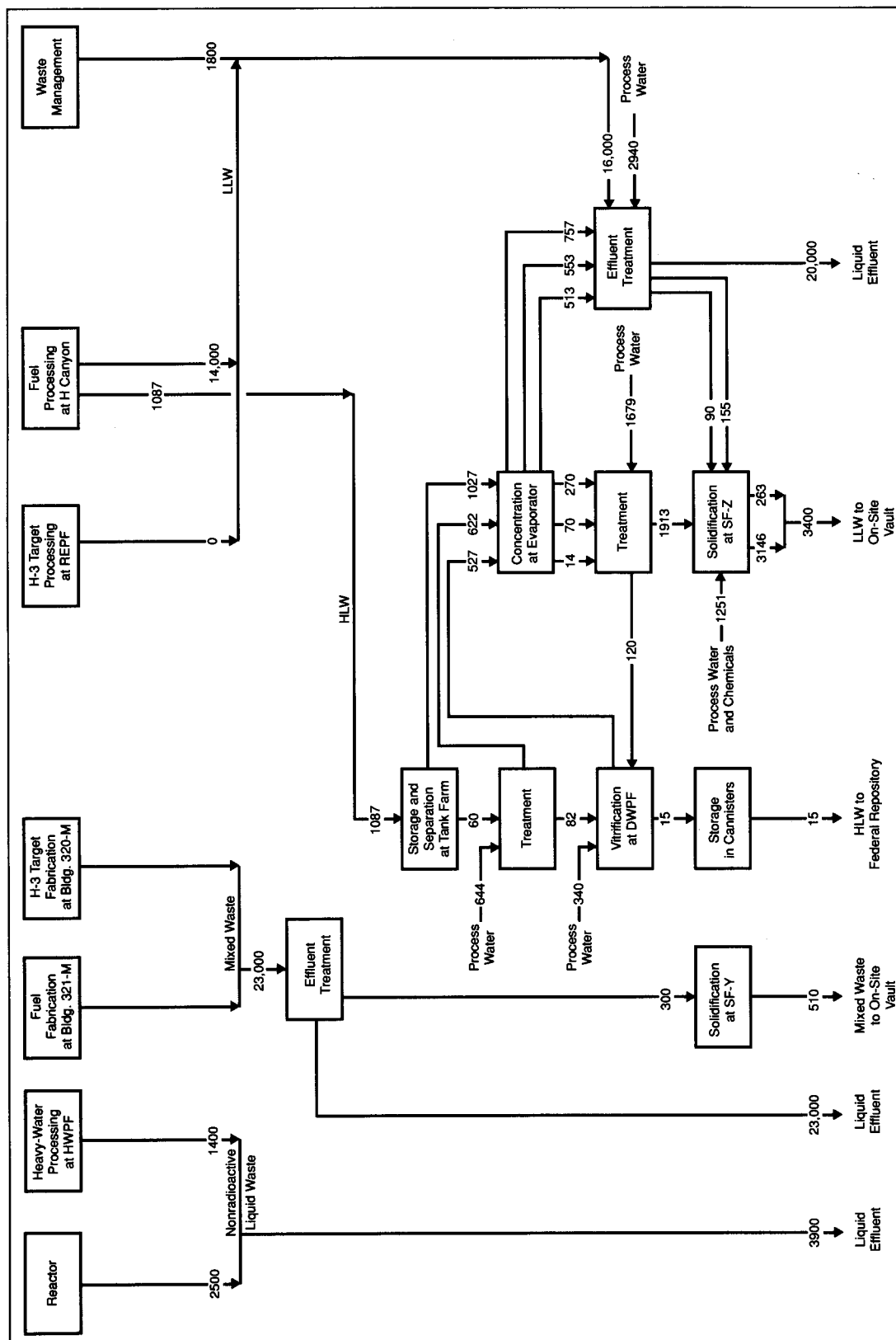


FIGURE M.40 Radioactive Liquid Waste Management for HWR Operations in the Tritium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991a-b; McClain 1990a-b; Bean 1990; McGough 1990)

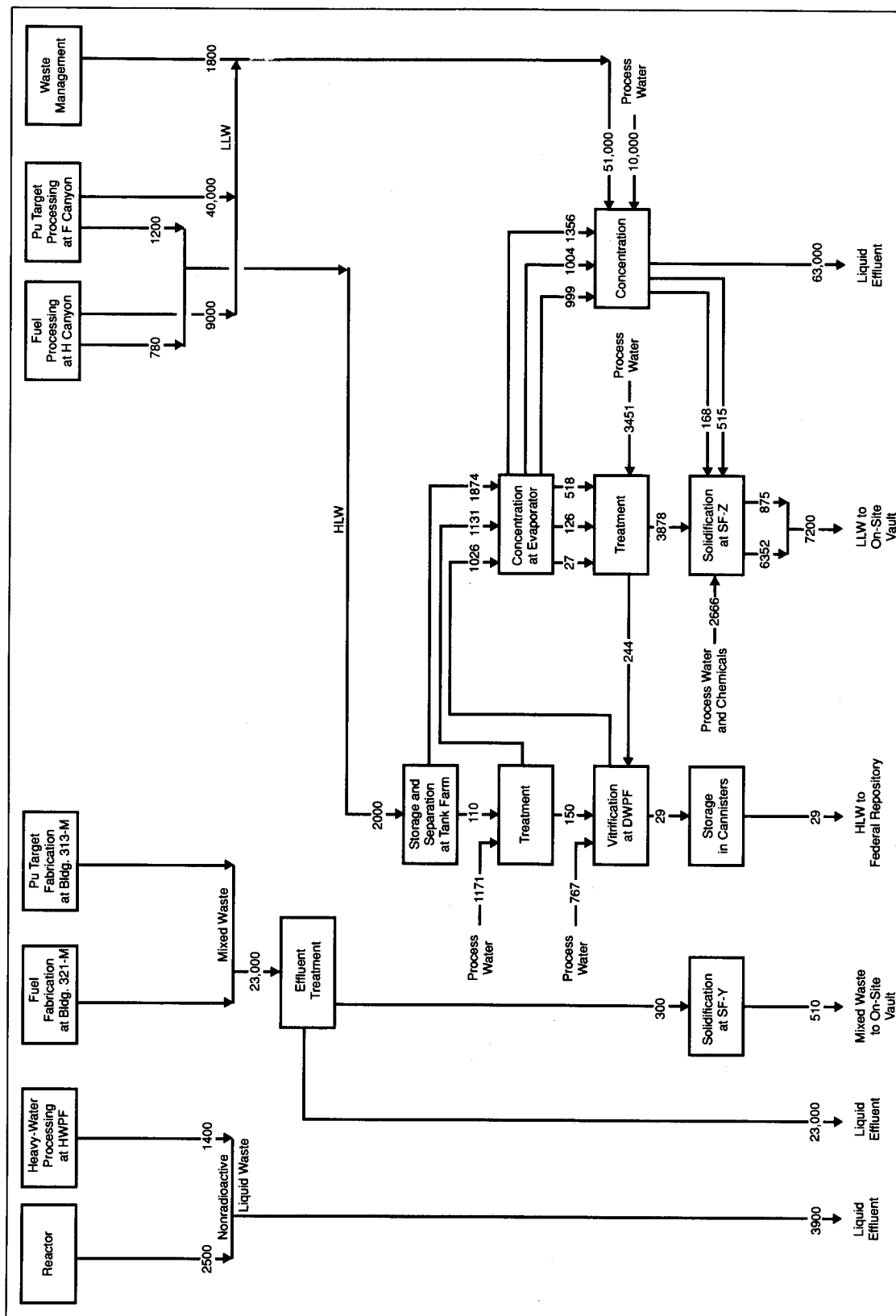


FIGURE M.41 Radioactive Liquid Waste Management for HWR Operations in the Plutonium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991a-b; McClain 1990a-b; Bean 1990; McGough 1990)

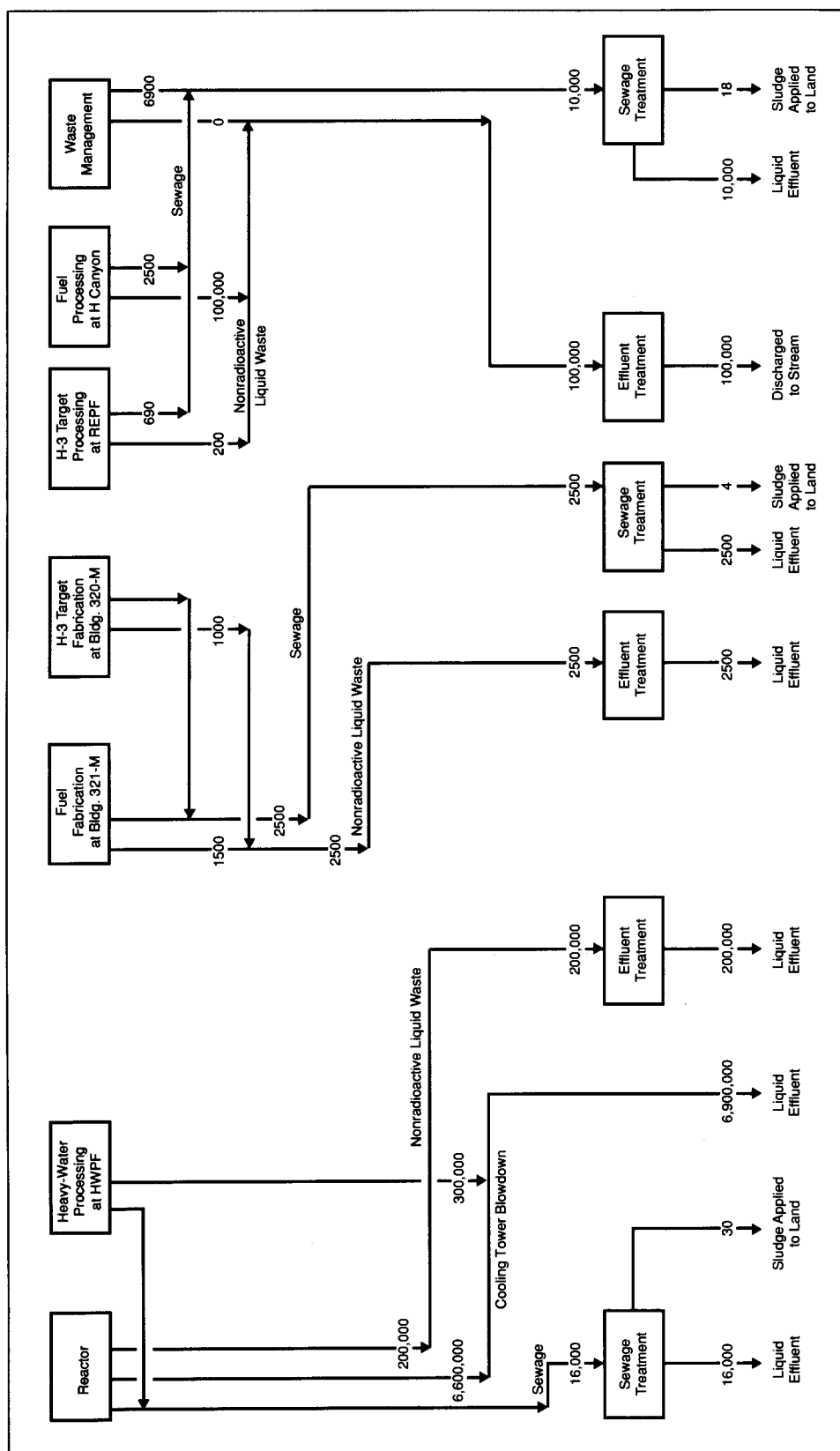


FIGURE M.42 Nonradioactive Liquid Waste Management for HWR Operations in the Tritium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991a-b; McClain 1990a-b; Bean 1990; McGough 1990)

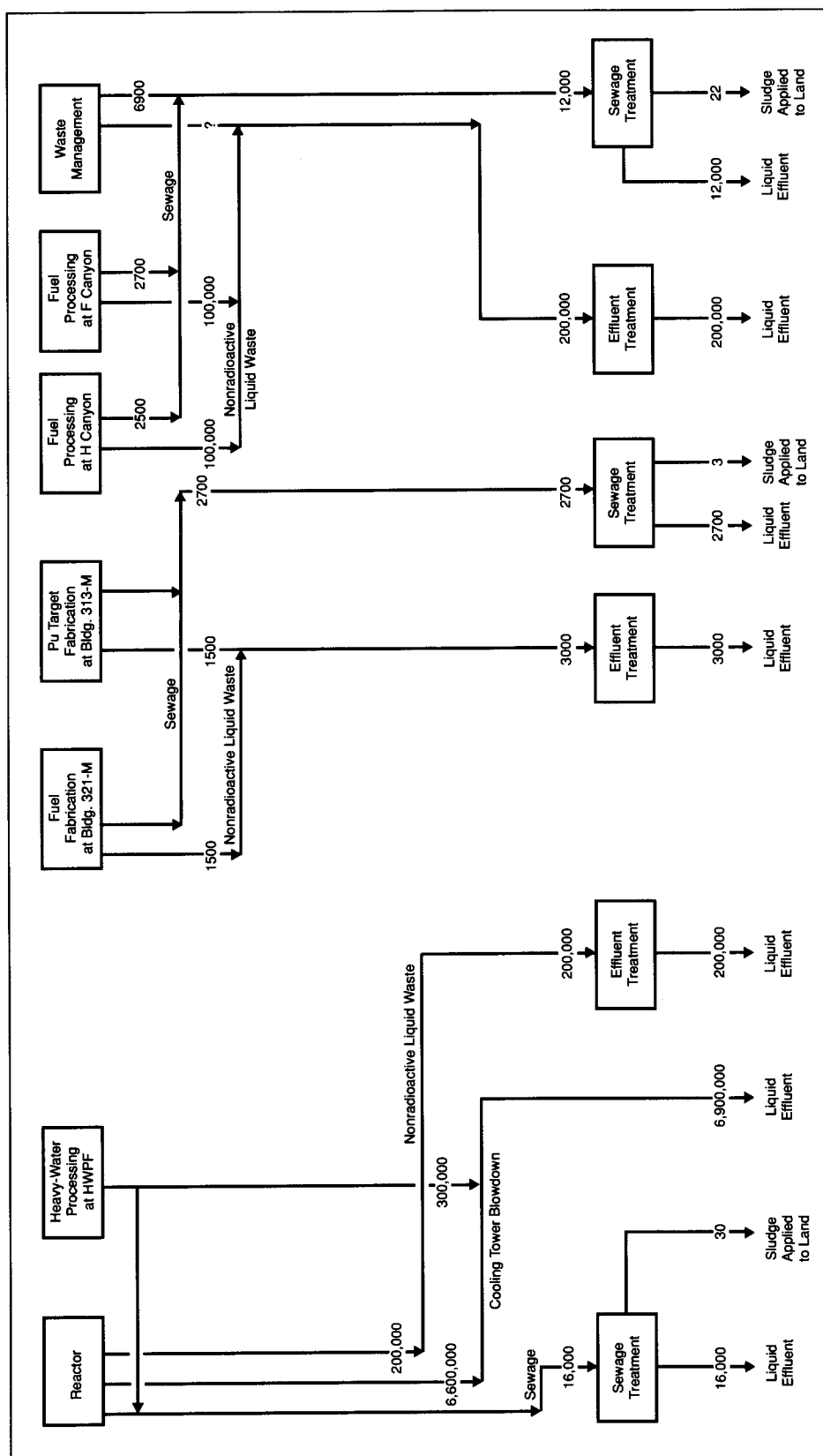


FIGURE M.43 Nonradioactive Liquid Waste Management for HWR Operations in the Plutonium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991a-b; McClain 1990a-b; Bean 1990; McGough 1990)

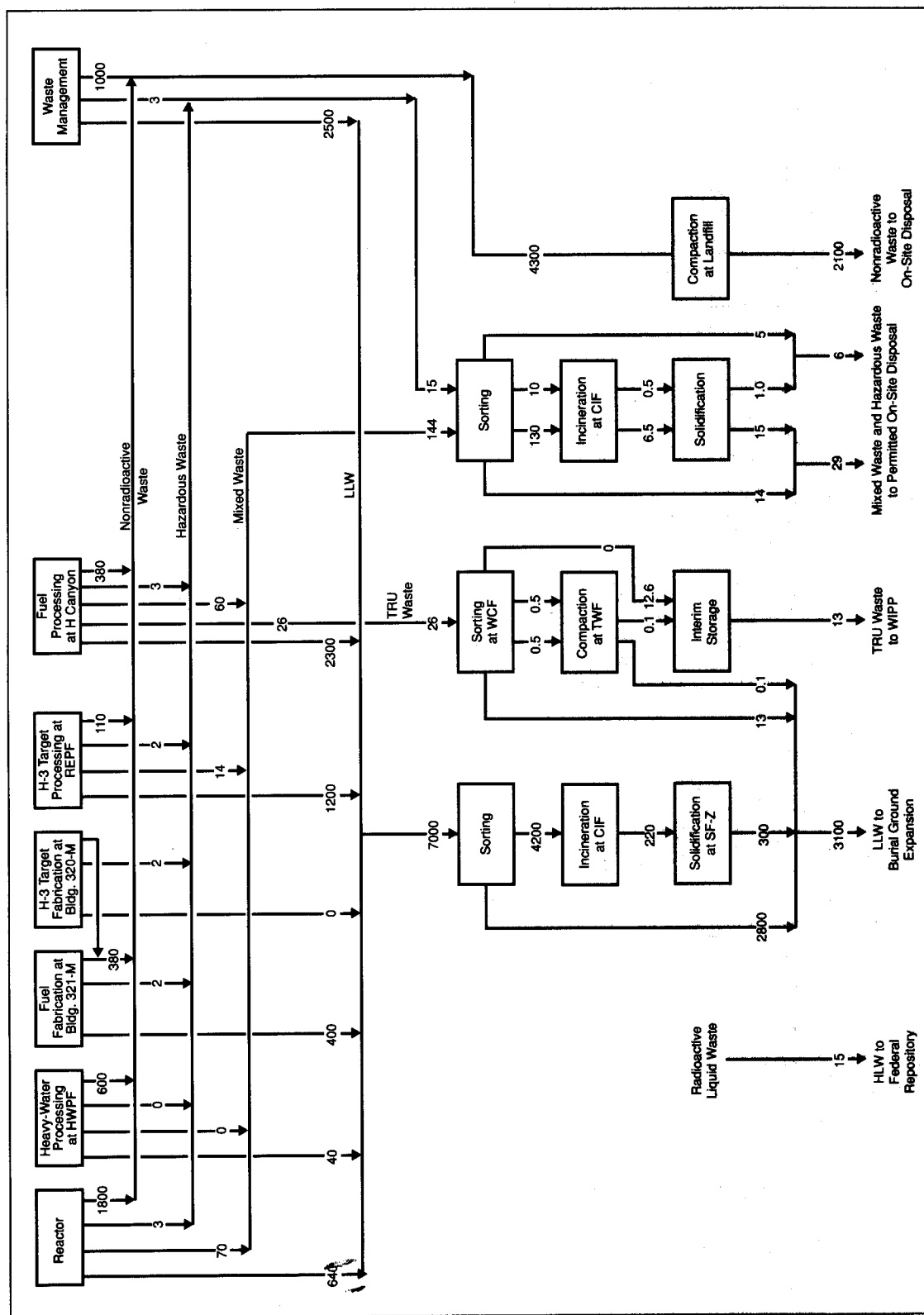


FIGURE M.44 Solid Waste Management for HWR Operations in the Tritium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991a-b; McClain 1990a-b; Bean 1990; McGough 1990)

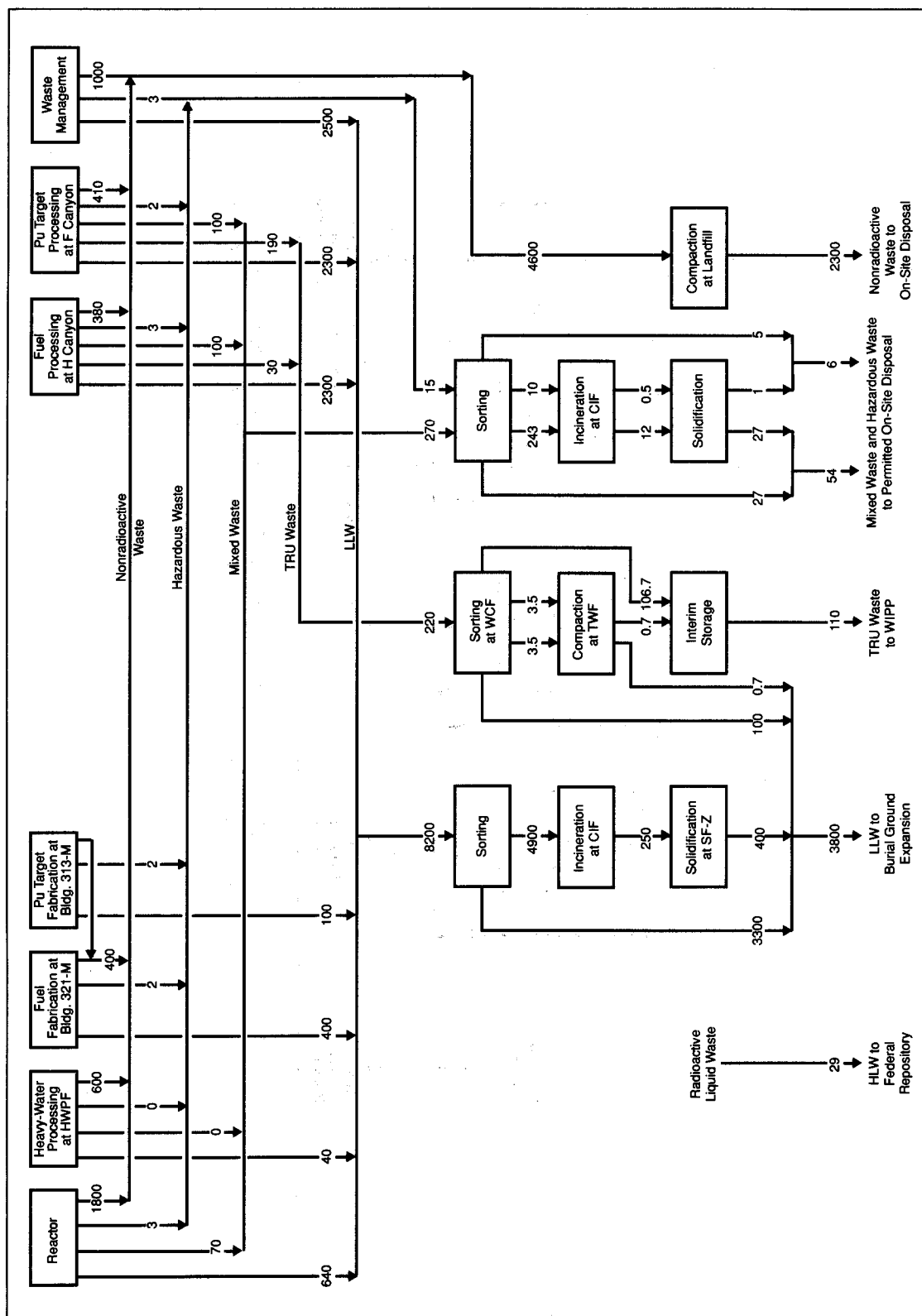


FIGURE M.45 Solid Waste Management for HWR Operations in the Plutonium Production Mode at SRS (waste volumes expressed in cubic meters per year (Sources: Based on data from WSRC 1991a-b; McClain 1990a-b; Bean 1990; McGough 1990)

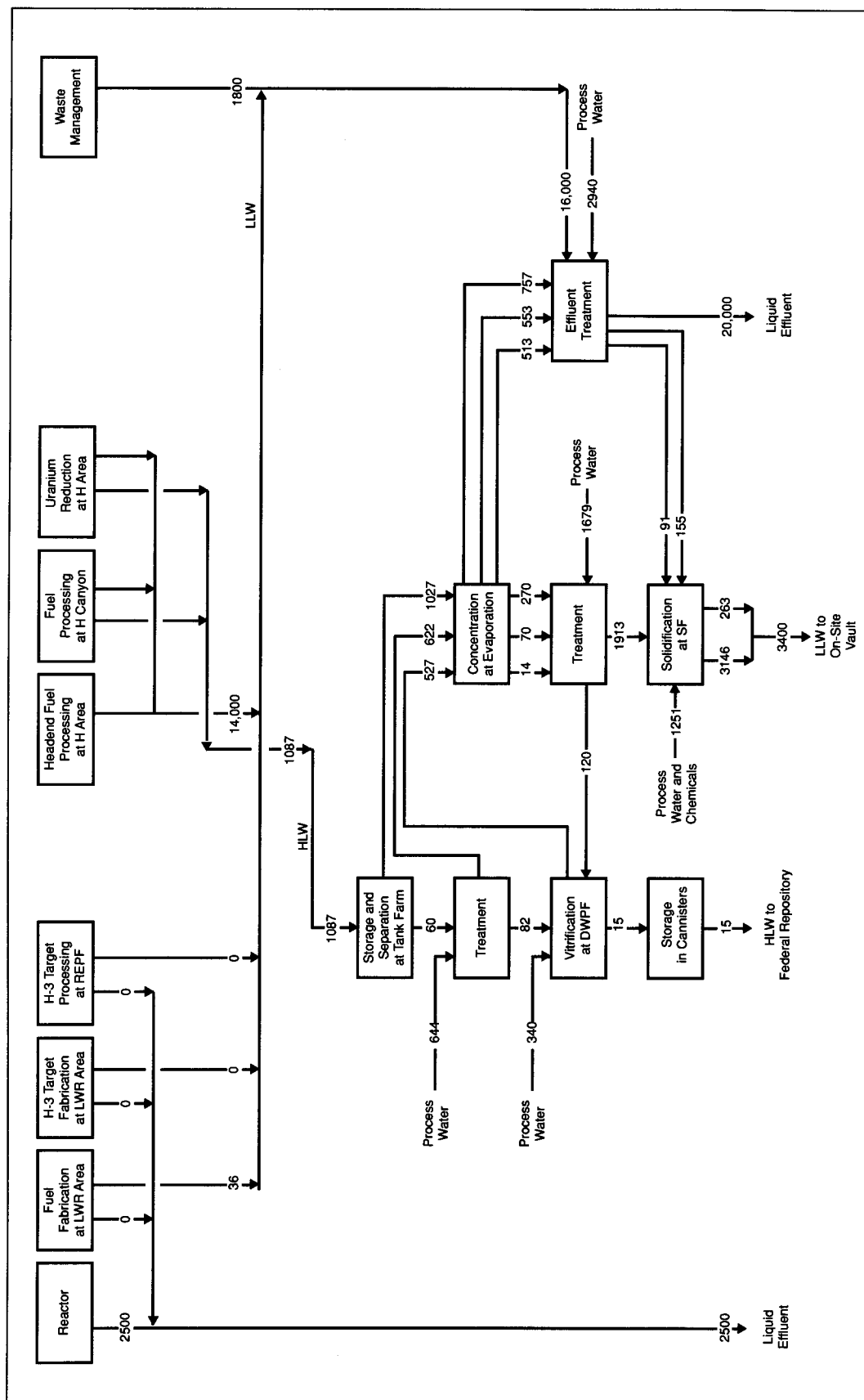


FIGURE M.46 Radioactive Liquid Waste Management for LWR Operations in the Tritium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991c-d; McClain 1990a-b; McGough 1990)



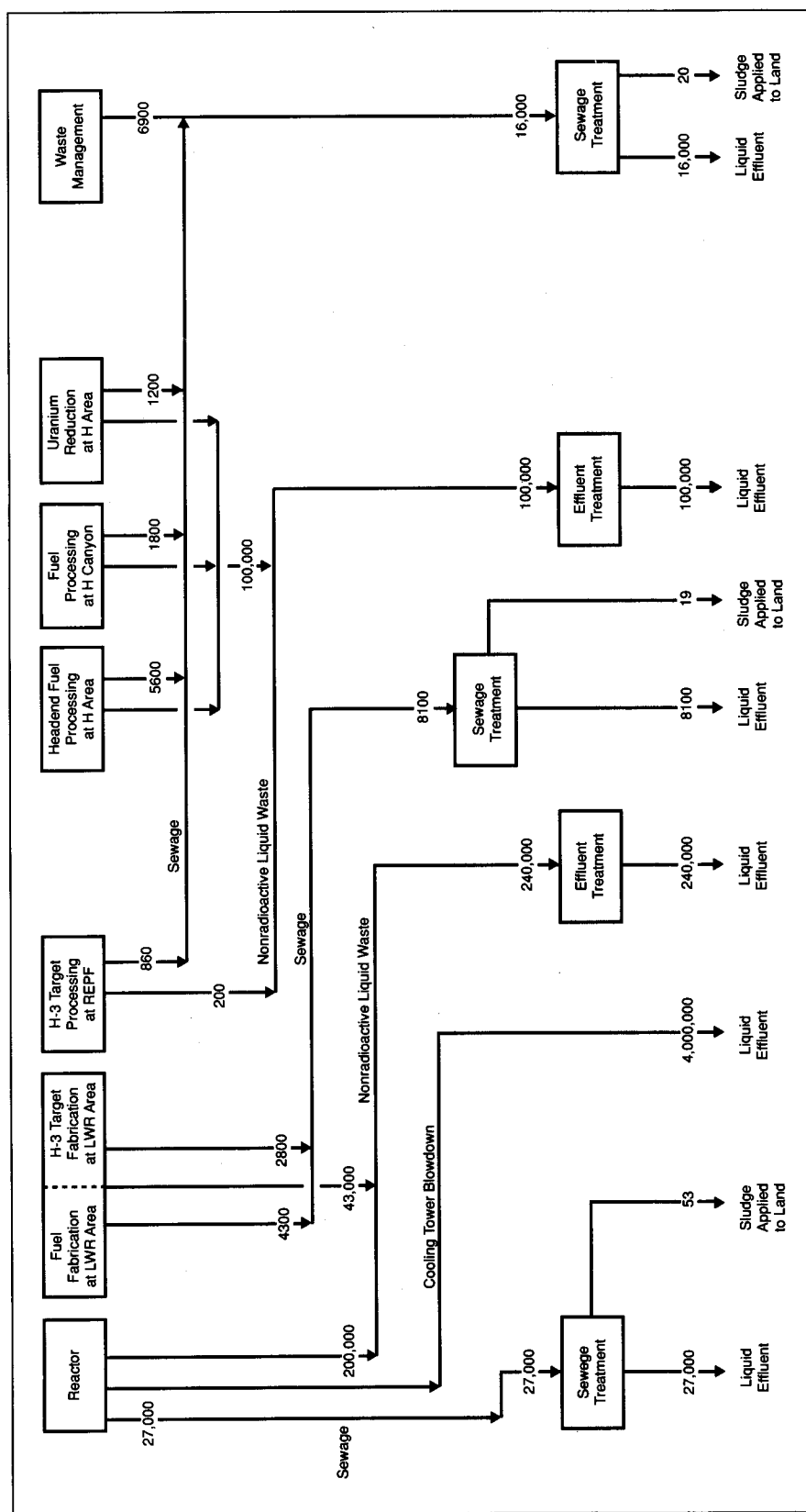


FIGURE M.48 Nonradioactive Liquid Waste Management for LWR Operations in the Tritium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991c-d; McClain 1990a-b; McGough 1990)

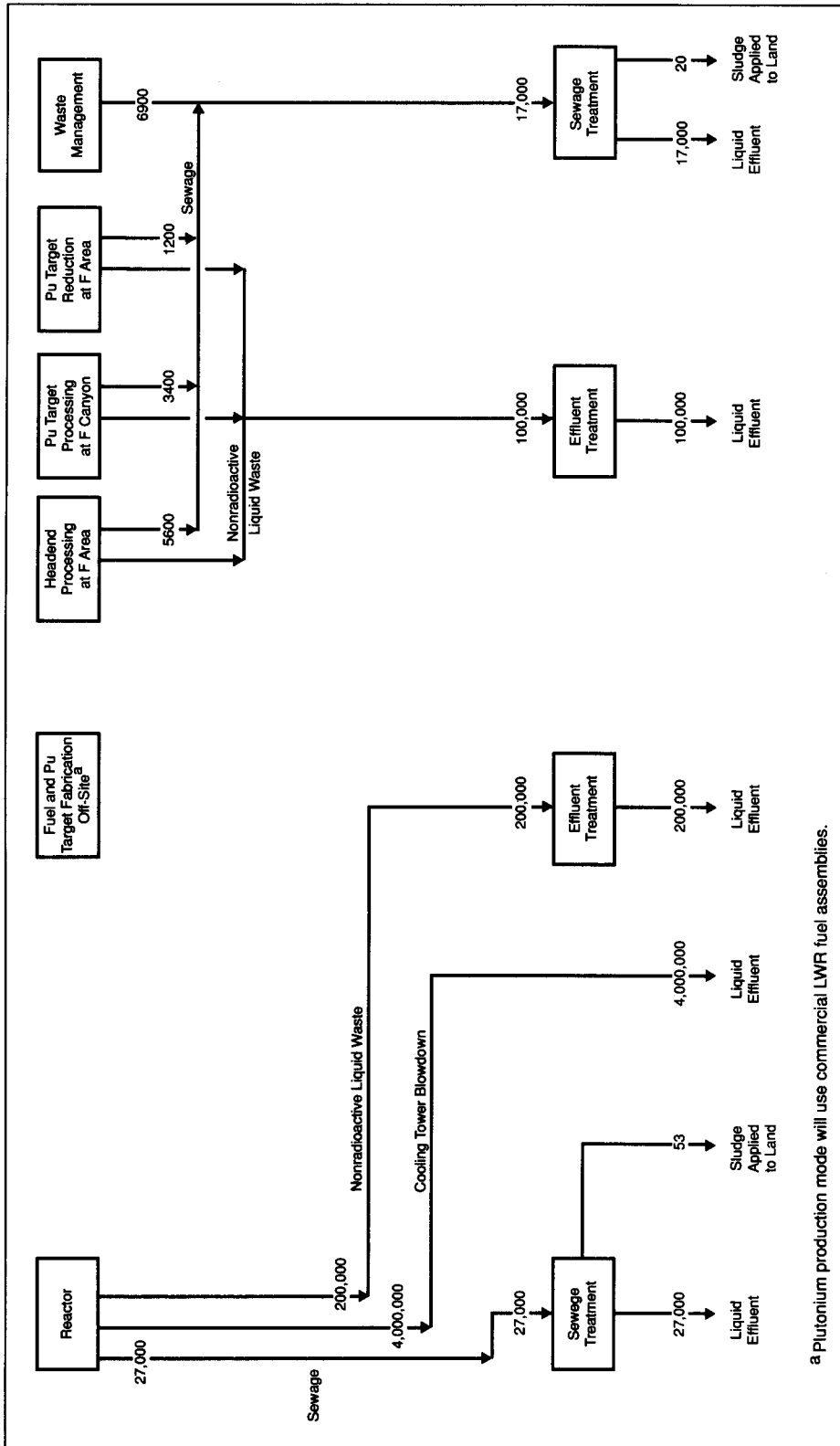


FIGURE M.49 Nonradioactive Liquid Waste Management for LWR Operations in the Plutonium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991c-d; McClain 1990a-b; McGough 1990)

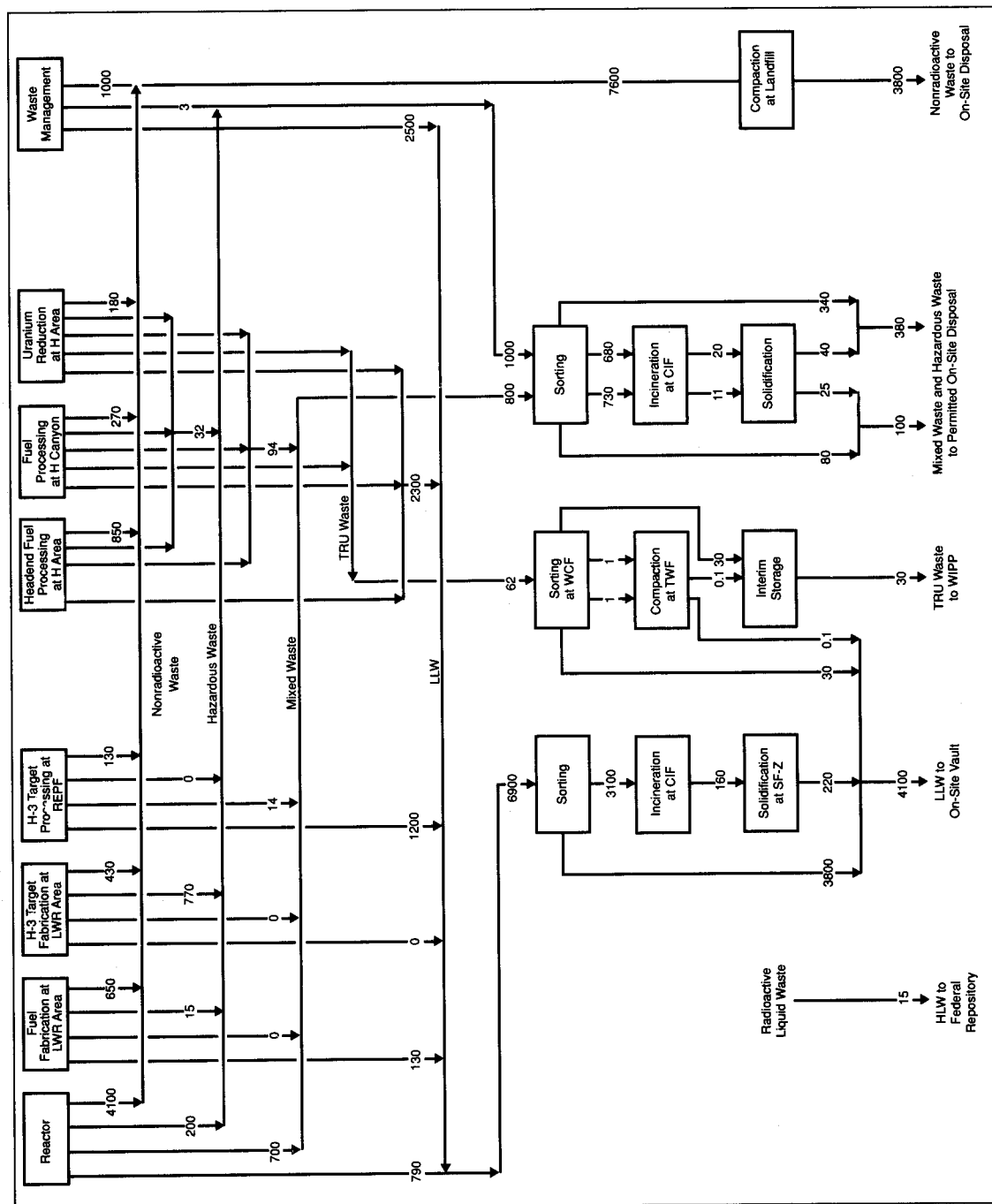


FIGURE M.50 Solid Waste Management for LWR Operations in the Tritium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991c-d; McClain 1990a-b; McGough 1990)

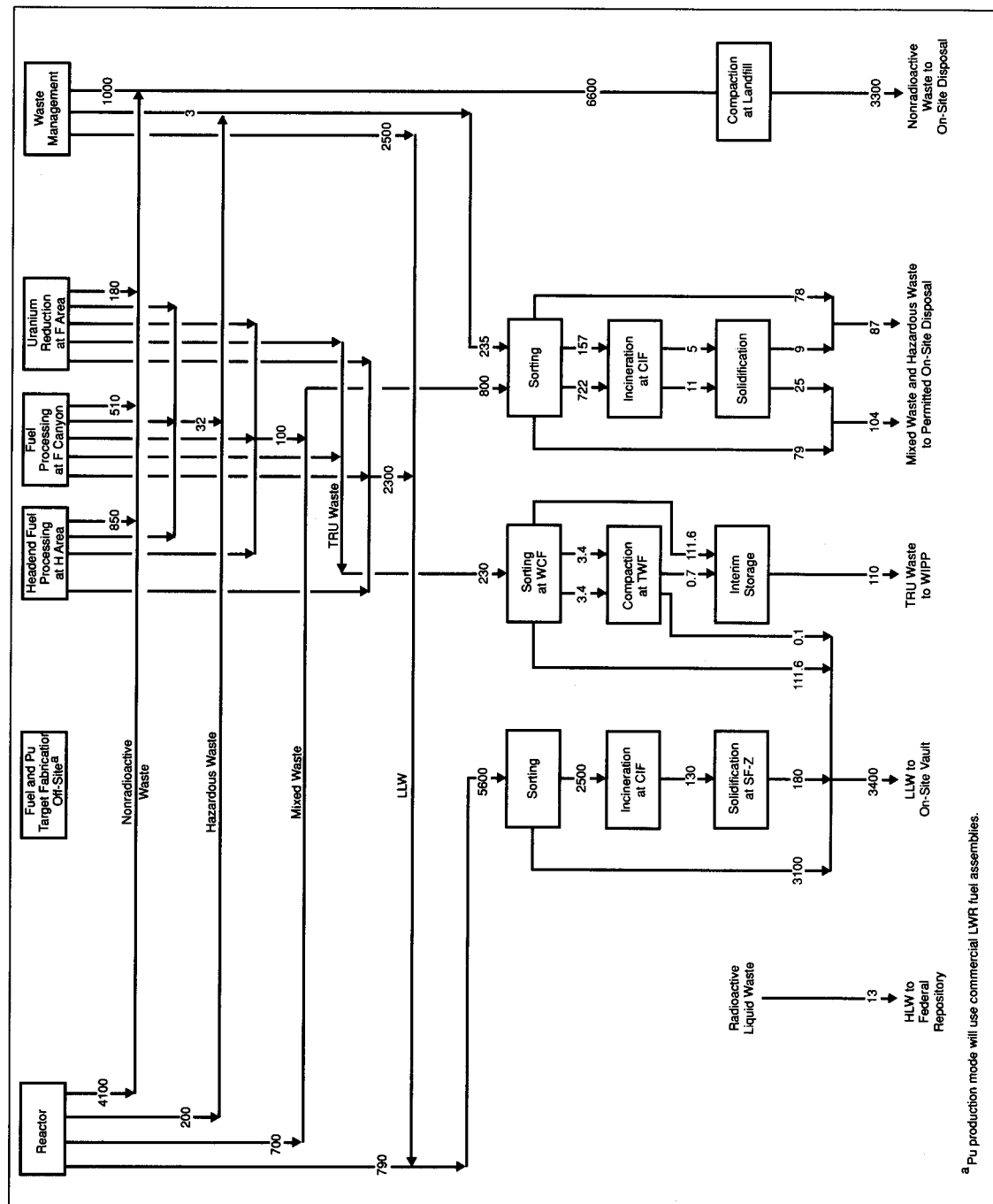


FIGURE M.51 Solid Waste Management for LWR Operations in the Plutonium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991c-d; McClain 1990a-b; McGough 1990)

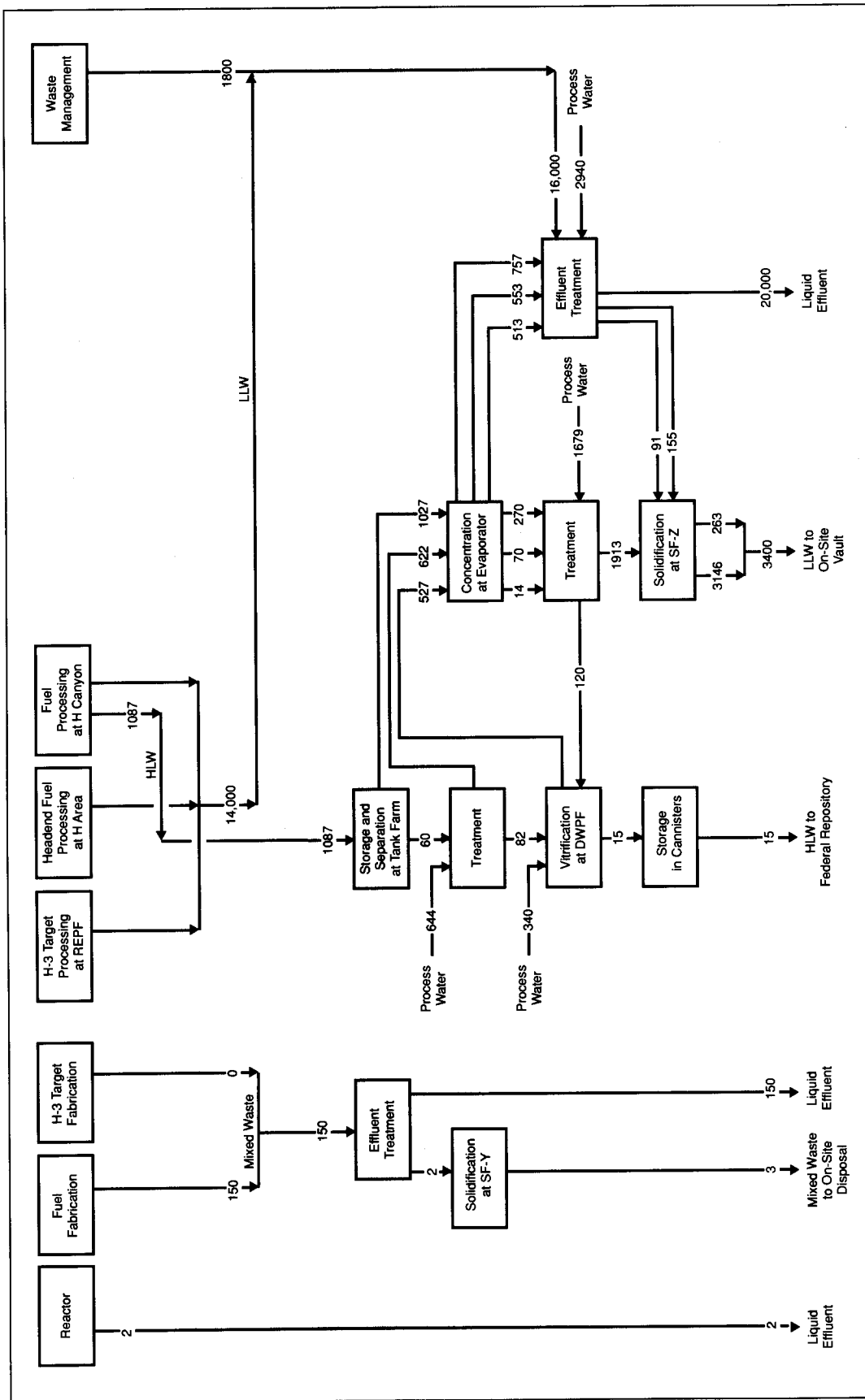


FIGURE M.52 Radioactive Liquid Waste Management for MHTGR Operations in the Tritium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991e-f; McClain 1990a-b; McGough 1990)

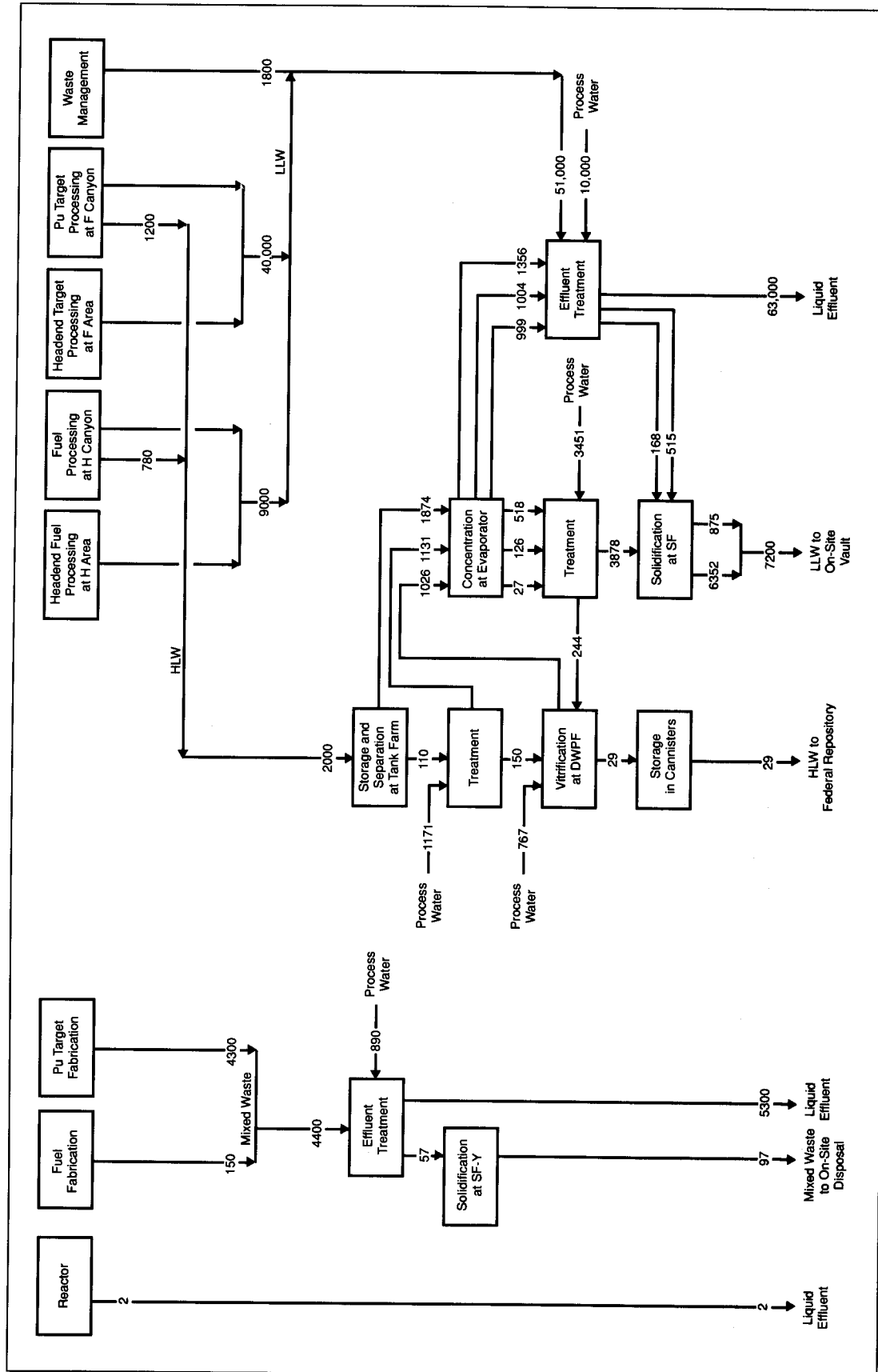


FIGURE M.53 Radioactive Liquid Waste Management for MHTGR Operations in the Plutonium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991e-f; McClain 1990a-b; McGough 1990)

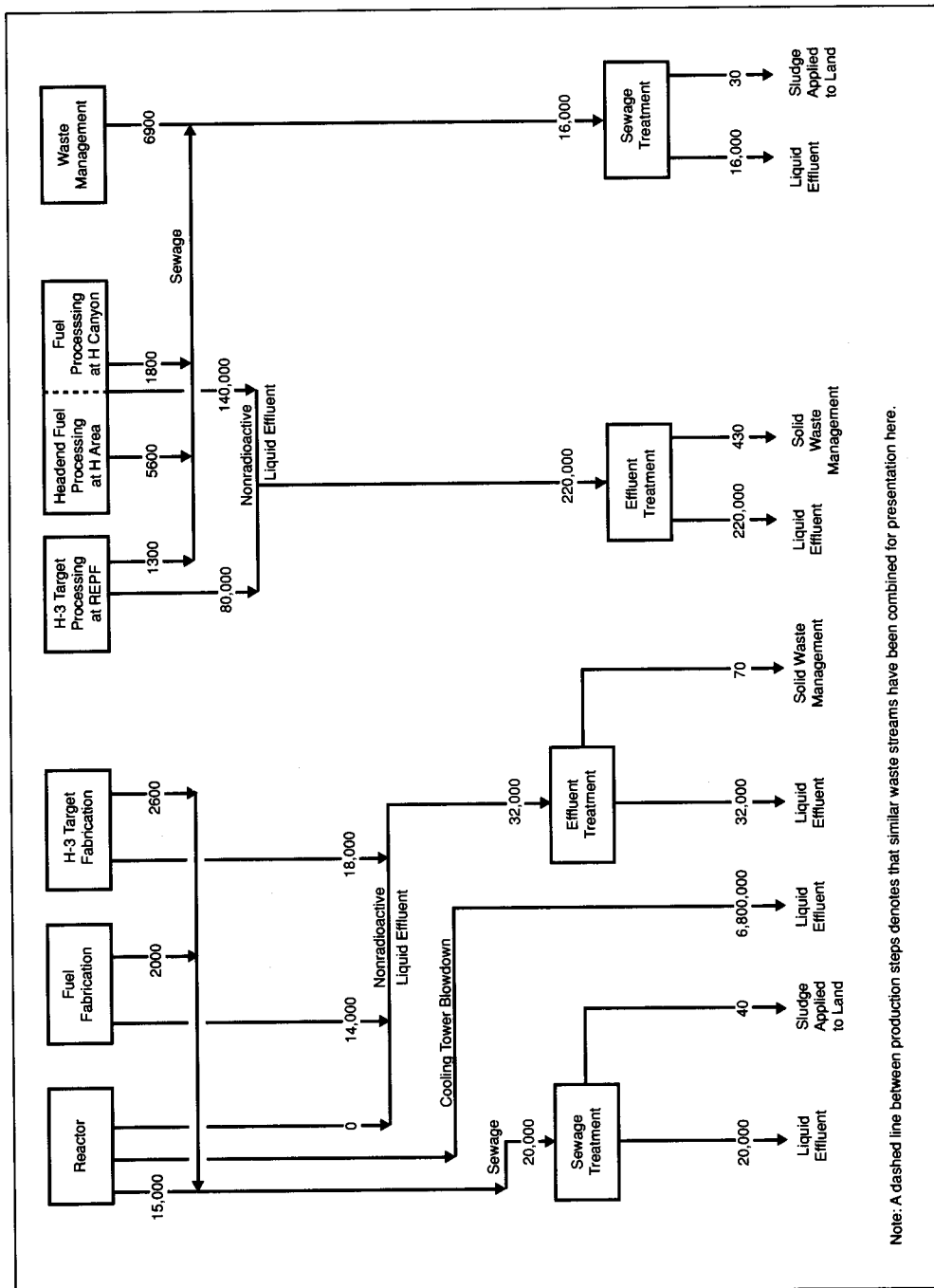


FIGURE M.54 Nonradioactive Liquid Waste Management for MHTGR Operations in the Tritium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991e-f; McClain 1990a-b; McGough 1990)

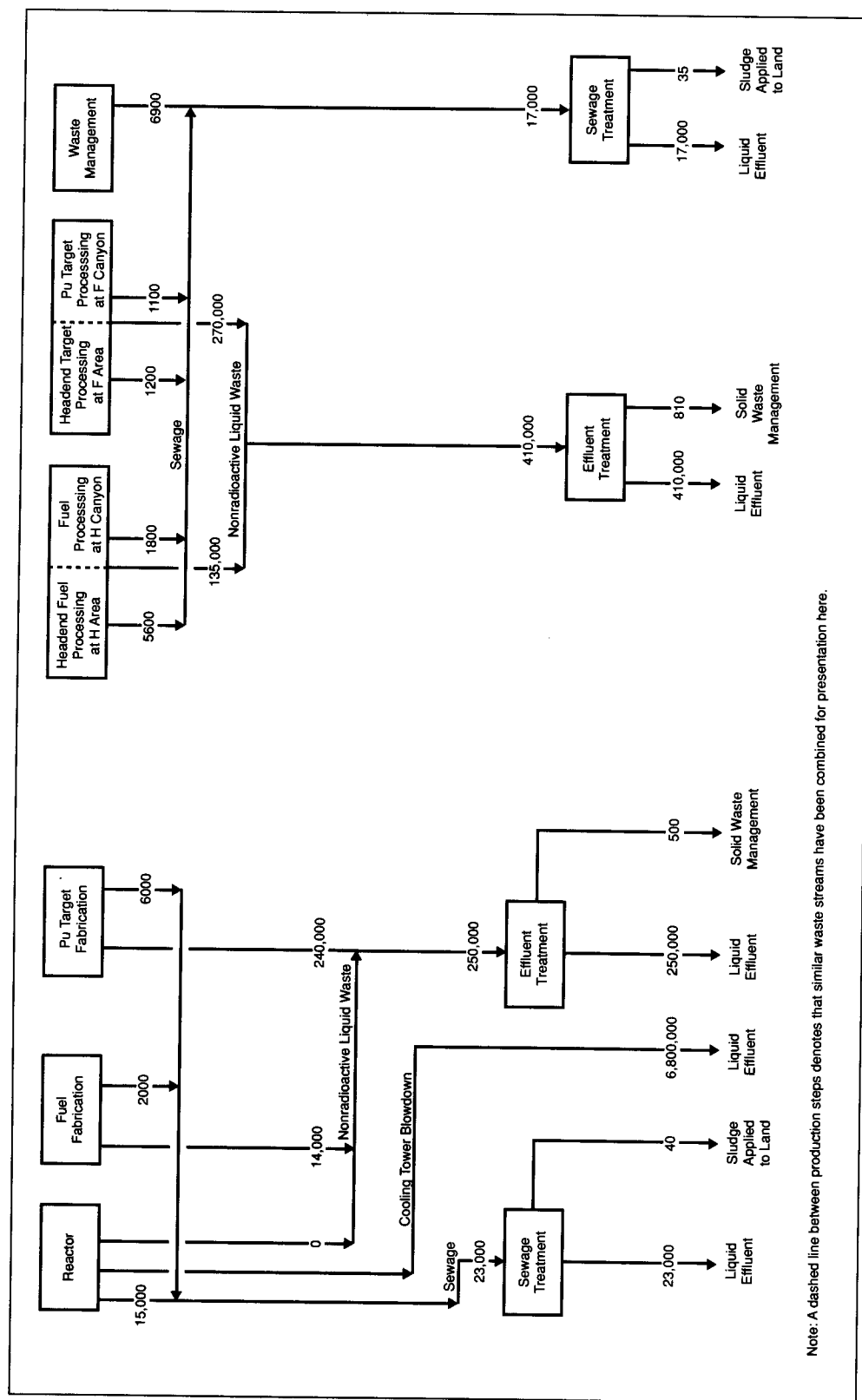


FIGURE M.55 Nonradioactive Liquid Waste Management for MHTGR Operations in the Plutonium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991e-f; McClain 1990a-b; McGough 1990)

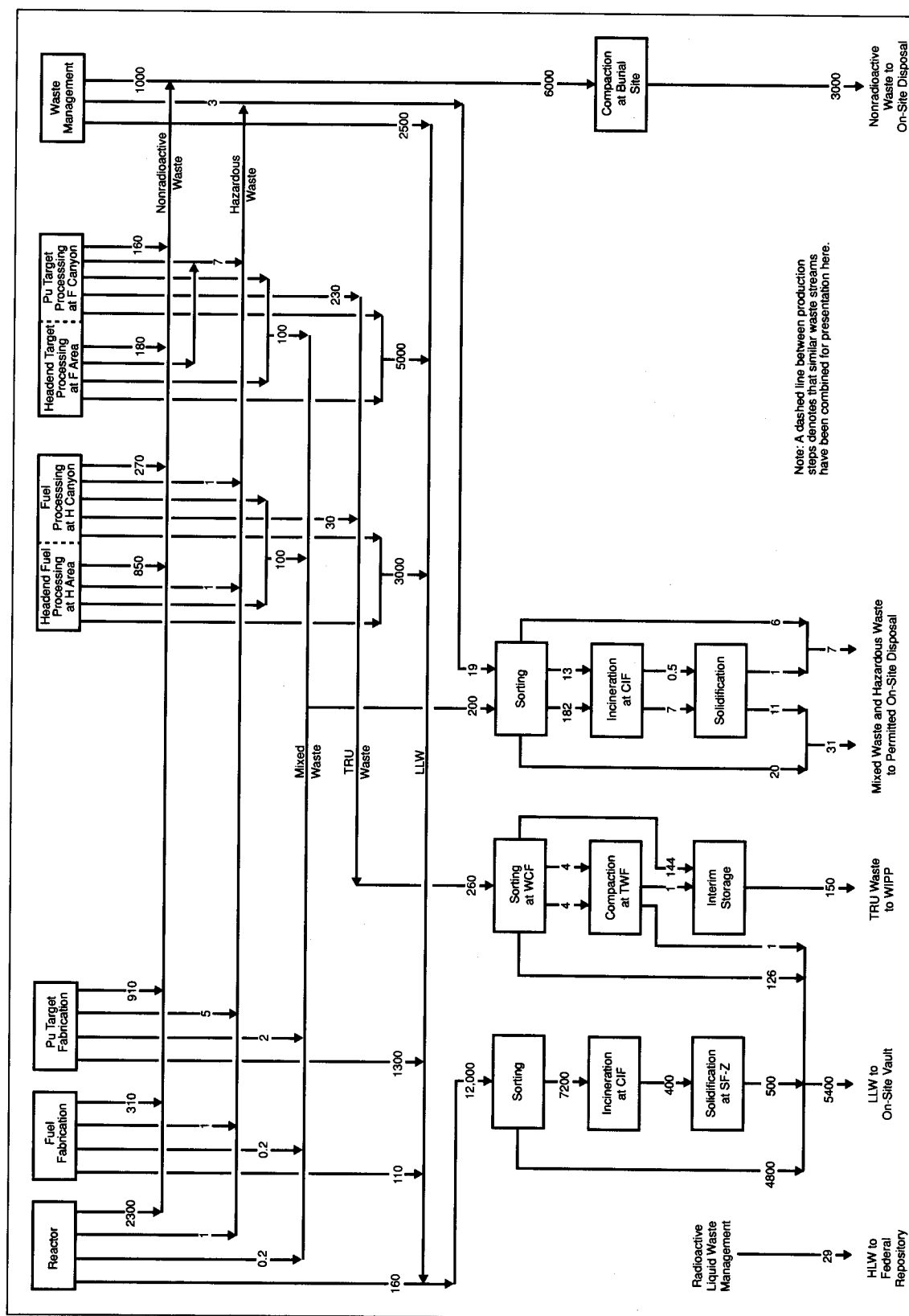


FIGURE M.57 Solid Waste Management for MHTGR Operations in the Plutonium Production Mode at SRS (waste volumes expressed in cubic meters per year) (Sources: Based on data from WSRC 1991e-f; McClain 1990a-b; McGough 1990)

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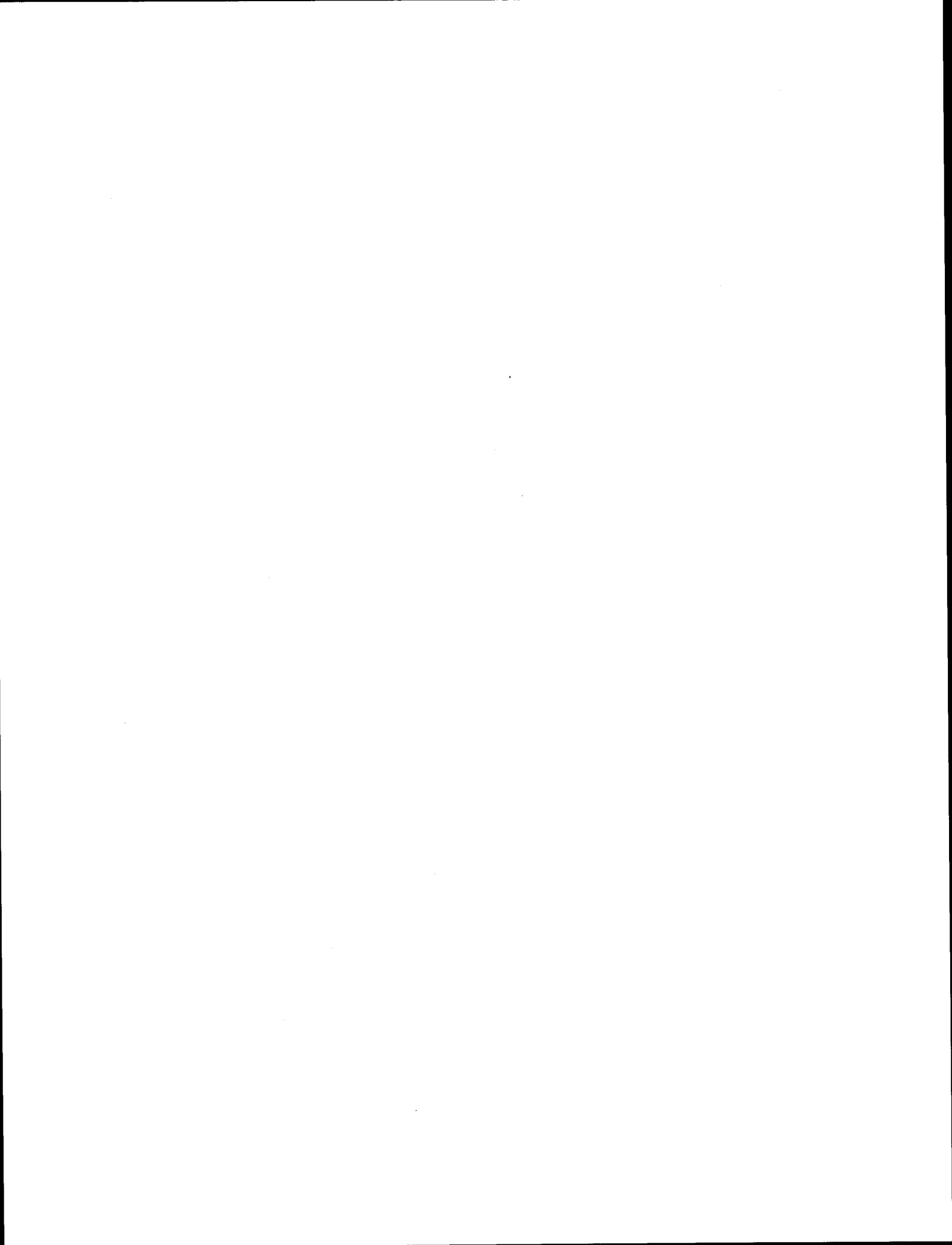
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APPENDIX N:
SOCIOECONOMICS



APPENDIX N: SOCIOECONOMICS

Peak work force projections are discussed in Section N.1. These projections are used in turn to project total regional in-migration and the regional distribution of socioeconomic impacts. Estimates of regional employment, income, and output effects of NPR construction and operations are shown by industry (SIC, standard industrial code) in Section N.2.

N.1 DERIVATION OF PEAK WORK FORCE ESTIMATES AND ALLOCATION OF IMPACTS

Peak work force requirements for each NPR alternative are based on preliminary generic design information and engineering total work force estimates. The work force loading curves for construction were derived from the International Atomic Energy Agency's *Manpower Development for Nuclear Power: A Guidebook* (IAEA 1980). These curves represent a generic pattern rather than exact values for the work force requirements for construction of a nuclear reactor and are based on historical experience. The work force includes the preconstruction design, engineering, and procurement effort that is necessary before actual construction begins. Training and trainees are not included; workers are assumed to be skilled or professional and to have been specially recruited for a given task. A series of support facilities generally must be built to accompany the main reactor building. Each facility is assumed to follow the same general construction pattern. Consequently, the loading curves overlap. The overall program loading curve can be derived from the smaller overlapping curves. Since the work force peaks roughly three-quarters of the way through the project, the generic work force curves are skewed to the right.

We have divided the project into ten series. Each series number corresponds to one year of construction. Sixty percent of the work is done in the first three-quarters of the total time period, and 40% is completed in the last one-quarter, immediately after the peak. A sharp increase from the early years of the project to the peak is followed by a gradual decline. In each series, as construction time increases, the peak decreases, and the curve is "stretched." That is, the curves have the same shape, but as the schedule lengthens, peaks decrease. In planning, minimizing the peak work force (building support facilities before or after the reactor work force peaks) is one strategy to lessen socioeconomic impacts. With the given work force, schedule dates, and loading curves, the total project peak can be estimated.

N.2 ESTIMATION OF REGIONAL IMPACTS

Estimates of total employment and income impacts to each alternative site are based on the construction, preoperation, and operating and maintenance costs estimated within the region of influence of each reactor/facilities configuration. Site-specific engineering cost estimates for each technology were derived from a life-cycle cost analysis performed by Oak Ridge National Laboratory (ORNL 1988; Williams and Fuller 1990). Estimates of costs and expenditures by SIC and the resulting impacts for each technology configuration are given in Tables N.1 through N.9, for the HWR, LWR, and MHTGR at Hanford, INEL, and SRS, respectively.

Although operating costs would recur over the 40-year life of the facilities, the estimates used in this analysis are expressed in terms of present value, for 1988 prices. Present values are used as a standard of comparison for consistency and convenience. The output, earnings, and employment impact estimates are based on the present costs and are also expressed in terms of present values. The socioeconomic impacts of construction and preoperation expenditures will not occur in a single year but will be spread over the preconstruction, construction, testing and certification, and preoperation periods of the reactor.

Operation and maintenance costs are based upon a typical year's expenditure, taking into account the expected life of the facility. Therefore, the output, earnings, and employment impacts of the operation and maintenance costs should be interpreted as expected annual levels of activity.

TABLE N.1 Regional Socioeconomic Impacts of the HWR Technology at Hanford

Category	Economic Impact (millions of 1988 dollars)				Employment Effect (workers)
	SIC Number	Primary Cost	Output Effect	Income Effect	
Construction and preoperation ^a					
Mining	10	20	26	7	354
General contracting	15	552	812	270	12,600
Heavy construction	16	187	275	91	4,220
Special trade	17	479	706	235	10,800
Chemicals	28	68	105	19	668
Stone and clay	32	5	7	1	83
Primary metal industries	33	136	174	25	908
Fabricated metal products	34	284	361	97	4,140
Machinery, excluding electrical	35	84	113	36	1,500
Electronics	36	45	59	19	818
Instruments and related equipment	38	10	14	5	193
Electricity, gas, and sanitation	49	0	0	0	3
Wholesale trade, durables	50	71	100	36	1,590
Engineering and accounting	87	1,510	2,470	1,020	54,500
Administration, environmental	95	13	18	6	374
Total ^b		3,470	5,250	1,870	92,700
Annual operation and maintenance					
Mining	10	0	0	0	9
Chemicals	28	102	156	28	994
Primary metal industries	33	33	23	30	157
Electricity, gas, and sanitation	49	70	90	16	618
Wholesale trade	51	3	0	0	0
Eating and drinking establishments	58	1	2	0	72
Insurance carriers	63	0	1	0	22
Business services	73	35	50	20	1,110
Engineering and accounting	87	76	109	45	2,410
Miscellaneous services	89	15	22	9	485
Administration, environmental	95	6	8	3	173
Total ^b		335	471	129	6,050

^aConstruction and preoperation employment effects are given in total person-years of employment; employment values for operation and maintenance represent annual effects.

^bColumns do not add to totals because of rounding.

Source: Evans 1990.

TABLE N.2 Regional Socioeconomic Impacts of Completion of the WNP-1 at Hanford

Category	Economic Impact (millions of 1988 dollars)				Employment Effect (workers)
	SIC Number	Primary Cost	Output Effect	Income Effect	
Construction and preoperation ^a					
General contracting	15	198	292	97	4,480
Heavy construction	16	36	52	17	803
Special trade	17	129	190	63	2,910
Chemicals	28	181	276	50	1,760
Primary metal industries	33	15	19	2	101
Fabricated metal products	34	244	310	83	3,560
Machinery, excluding electrical	35	42	56	18	750
Electronics	36	14	19	6	264
Instruments and related equipment	38	4	6	2	88
Electricity, gas, and sanitation	49	0	0	0	2
Wholesale trade, durables	50	27	39	13	615
Engineering and accounting	87	341	488	201	10,800
Administration, environmental	95	12	17	6	346
Administration, other	96	57	0	0	0
Total ^b		1,300	1,770	562	26,500
Annual operation and maintenance					
Mining	10	0	0	0	9
Chemicals	28	103	156	28	994
Primary metal industries	33	33	24	31	161
Machinery, excluding electrical	35	18	23	7	319
Electricity, gas, and sanitation	49	51	66	12	454
Wholesale trade	51	7	0	0	0
Eating and drinking establishments	58	1	2	0	72
Insurance carriers	63	0	0	0	22
Business services	73	35	50	20	1,110
Engineering and accounting	87	76	109	45	2,410
Miscellaneous services	89	15	22	9	485
Administration, environmental	95	5	7	2	144
Total ^b		339	470	132	6,180

^aConstruction and preoperation employment effects are given in total person-years of employment; employment values for operation and maintenance represent annual effects.

^bColumns do not add to totals because of rounding.

Source: Evans 1990.

TABLE N.3 Regional Socioeconomic Impacts of the MHTGR Technology at Hanford

Category	Economic Impact (millions of 1988 dollars)				Employment Effect (workers)
	SIC Number	Primary Cost	Output Effect	Income Effect	
Construction and preoperation ^a					
Mining	10	60	79	22	1,060
General contracting	15	643	946	315	14,500
Heavy construction	16	254	374	124	5,740
Special trade	17	397	585	194	8,970
Chemicals	28	68	319	58	2,020
Stone and clay	32	0	0	0	0
Primary metal industries	33	14	18	2	94
Fabricated metal products	34	484	615	165	7,070
Machinery, excluding electrical	35	72	95	31	1,280
Electronics	36	26	34	11	473
Instruments and related equipment	38	9	13	4	183
Electricity, gas, and sanitation	49	0	0	0	0
Wholesale trade, durables	50	43	60	21	967
Wholesale trade, nondurables	51	50	69	25	1,110
Engineering and accounting	87	1,520	2,170	893	47,800
Administration, environmental	95	11	15	5	317
Administration, other	96	32	45	16	922
Total ^b		3,690	5,440	1,890	92,506
Annual operation and maintenance					
Mining	10	11	14	4	195
Chemicals	28	125	190	34	1,210
Primary metal industries	33	33	77	99	516
Electricity, gas, and sanitation	49	53	68	12	467
Wholesale trade	51	11	0	0	0
Eating and drinking establishments	58	1	2	0	72
Insurance carriers	63	0	1	0	22
Business services	73	35	50	20	1,110
Engineering and accounting	87	76	109	45	2,410
Miscellaneous services	89	15	22	9	485
Administration, environmental	95	6	8	3	173
Total ^b		413	565	145	6,660

^aConstruction and preoperation employment effects are given in total person-years of employment; employment values for operation and maintenance represent annual effects.

^bColumns do not add to totals because of rounding.

Source: Evans, 1990.

TABLE N.4 Regional Socioeconomic Impacts of the HWR Technology at INEL

Category	Economic Impact (millions of 1988 dollars)				Employment Effect (workers)
	SIC Number	Primary Cost	Output Effect	Income Effect	
Construction and preoperation ^a					
Mining	10	20	30	7	310
General contracting	15	468	813	283	14,300
Heavy construction	16	175	304	106	5,360
Special trade	17	403	701	244	12,300
Chemicals	28	66	114	22	953
Stone and clay	32	5	9	2	173
Primary metal industries	33	120	187	63	3,290
Fabricated metal products	34	242	343	95	5,030
Machinery, excluding electrical	35	69	105	31	1,410
Electronics	36	39	61	22	1,040
Instruments and related equipment	38	8	12	4	258
Electricity, gas, and sanitation	49	0	0	0	2
Wholesale trade, durables	50	57	91	34	1,910
Engineering and accounting	87	1,280	2,100	917	52,000
Administration, environmental	95	13	0	0	0
Administration, other	96	0	22	8	529
Total ^b		2,970	4,900	1,840	98,865
Annual operation and maintenance					
Mining	10	0	0	0	8
Chemicals	28	103	176	35	1,480
Primary metal industries	33	33	23	36	644
Electricity, gas, and sanitation	49	58	79	11	482
Wholesale trade	51	3	0	0	0
Eating and drinking establishments	58	1	2	0	98
Insurance carriers	63	0	0	0	33
Business services	73	35	57	25	1,430
Engineering and accounting	87	76	126	54	3,110
Miscellaneous services	89	15	25	11	625
Administration, environmental	95	4	6	2	164
Administration, other	96	0	0	0	0
Total ^b		322	512	154	8,074

^aConstruction and preoperation employment effects are given in total person-years of employment; employment values for operation and maintenance represent annual effects.

^bColumns do not add to totals because of rounding.

Source: Moor 1990.

TABLE N.5 Regional Socioeconomic Impacts of the LWR Technology at INEL

Category	Economic Impact (millions of 1988 dollars)				Employment Effect (workers)
	SIC Number	Primary Cost	Output Effect	Income Effect	
Construction and preoperation ^a					
Mining	10	90	139	32	1,400
General contracting	15	479	832	289	14,600
Heavy construction	16	148	257	89	4,530
Special trade	17	390	677	236	11,900
Chemicals	28	191	328	65	2,750
Stone and clay	32	9	16	4	296
Primary metal industries	33	183	285	96	5,020
Fabricated metal products	34	396	561	157	8,240
Machinery, excluding electrical	35	87	132	39	1,780
Electronics	36	35	55	19	935
Instruments and related equipment	38	5	8	3	183
Electricity, gas, and sanitation	49	0	0	0	2
Wholesale trade, durables	50	50	80	30	1,690
Engineering and accounting	87	918	1,510	657	37,300
Administration, environmental	95	12	20	7	488
Total ^b		3,000	4,900	1,730	91,114
Annual operation and maintenance					
Mining	10	0	0	0	8
Chemicals	28	102	174	34	1,460
Primary metal industries	33	33	24	37	658
Machinery, excluding electricity	35	18	27	8	365
Electricity, gas, and sanitation	49	46	62	8	384
Wholesale trade	51	7	0	0	0
Eating and drinking establishments	58	1	2	0	98
Insurance carriers	63	0	1	0	33
Business services	73	35	57	25	1,430
Engineering and accounting	87	76	126	54	3,110
Miscellaneous services	89	15	25	11	625
Administration, environmental	95	3	5	1	123
Total ^b		331	521	159	8,294

^aConstruction and preoperation employment effects are given in total person-years of employment; employment values for operation and maintenance represent annual effects.

^bColumns do not add to totals because of rounding.

Source: Moor 1990.

TABLE N.6 Regional Socioeconomic Impacts of the MHTGR Technology at INEL

Category	Economic Impact (millions of 1988 dollars)				Employment Effect (workers)
	SIC Number	Primary Cost	Output Effect	Income Effect	
Construction and preoperation ^a					
Mining	10	59	92	21	926
General contracting	15	508	882	307	15,500
Heavy construction	16	299	520	181	9,160
Special trade	17	272	473	164	8,320
Chemicals	28	208	358	71	3,000
Stone and clay	32	0	0	0	0
Primary metal industries	33	7	11	3	195
Fabricated metal products	34	426	603	168	8,860
Machinery, excluding electrical	35	52	78	23	1,060
Electronics	36	31	48	17	816
Instruments and related equipment	38	7	11	4	247
Electricity, gas, and sanitation	49	0	0	0	0
Wholesale trade, durables	50	44	71	26	1,500
Engineering and accounting	87	1,330	2,190	954	54,100
Administration, environmental	95	11	18	7	448
Administration, other	96	32	54	21	1,300
Total ^b		3,330	5,490	2,000	105,432
Annual operation and maintenance					
Mining	10	11	17	4	171
Chemicals	28	119	205	41	1,710
Primary metal industries	33	33	72	112	1,970
Electricity, gas, and sanitation	49	42	57	8	351
Wholesale trade	51	11	0	0	0
Eating and drinking establishments	58	1	2	0	98
Insurance carriers	63	0	1	0	33
Business services	73	35	57	25	1,430
Engineering and accounting	87	76	126	54	3,110
Miscellaneous services	89	15	25	11	625
Administration, environmental	95	4	6	2	164
Total ^b		389	612	186	9,662

^aConstruction and preoperation employment effects are given in total person-years of employment; employment values for operation and maintenance represent annual effects.

^bColumns do not add to totals because of rounding.

Source: Moor 1990.

TABLE N.7 Regional Socioeconomic Impacts of the HWR Technology at SRS

Category	Economic Impact (millions of 1988 dollars)				Employment Effect (workers)
	SIC Number	Primary Cost	Output Effect	Income Effect	
Construction and preoperation ^a					
Mining	10	20	29	8	348
General contracting	15	301	525	175	9,160
Heavy construction	16	147	257	85	4,480
Special trade	17	330	574	191	10,000
Chemicals	28	60	100	19	786
Stone and clay	32	6	11	3	155
Primary metal industries	33	121	180	54	2,180
Fabricated metal products	34	298	436	121	5,670
Machinery, excluding electrical	35	61	95	29	1,320
Electronics	36	29	45	12	557
Instruments and related equipment	38	4	7	2	117
Electricity, gas, and sanitation	49	0	0	0	2
Wholesale trade, durables	50	42	17	6	287
Engineering and accounting	87	1,070	1,750	790	45,100
Administration, environmental	95	13	21	8	410
Total ^b		2,500	4,050	1,510	80,572
Annual operation and maintenance					
Mining	10	0	0	0	9
Chemicals	28	100	168	32	1,320
Primary metal industries	33	33	23	35	425
Electricity, gas, and sanitation	49	74	97	15	608
Wholesale trade	51	3	0	0	0
Eating and drinking establishments	58	1	2	0	79
Insurance carriers	63	0	1	0	23
Business services	73	35	57	25	1,470
Engineering and accounting	87	76	124	56	3,220
Miscellaneous services	89	15	25	11	647
Administration, environmental	95	6	10	4	205
Total ^b		337	522	158	8,006

^aConstruction and preoperation employment effects are given in total person-years of employment; employment values for operation and maintenance represent annual effects.

^bColumns do not add to totals because of rounding.

Source: Bean 1990.

TABLE N.8 Regional Socioeconomic Impacts of the LWR Technology at SRS

Category	Economic Impact (millions of 1988 dollars)				Employment Effect (workers)
	SIC Number	Primary Cost	Output Effect	Income Effect	
Construction and preoperation ^a					
Mining	10	90	134	36	1,570
General contracting	15	430	749	249	13,000
Heavy construction	16	141	246	81	4,290
Special trade	17	368	642	213	11,200
Chemicals	28	191	319	62	2,500
Stone and clay	32	9	15	4	215
Primary metal industries	33	176	262	80	3,190
Fabricated metal products	34	393	575	160	7,470
Machinery, excluding electrical	35	92	145	44	2,010
Electronics	36	33	51	13	636
Instruments and related equipment	38	5	9	2	136
Electricity, gas, and sanitation	49	0	0	0	2
Wholesale trade, durables	50	49	78	28	1,280
Engineering and accounting	87	929	1,510	683	39,000
Administration, environmental	95	12	20	7	378
Total ^b		2,920	4,760	1,670	86,877
Annual operation and maintenance					
Mining	10	0	0	0	9
Chemicals	28	102	170	33	1,330
Primary metal industries	33	33	24	35	434
Machinery, excluding electricity	35	18	28	8	391
Electricity, gas, and sanitation	49	59	78	12	490
Wholesale trade	51	7	0	0	0
Eating and drinking establishments	58	1	2	0	79
Insurance carriers	63	0	1	0	23
Business services	73	35	57	25	1,470
Engineering and accounting	87	76	124	56	3,220
Miscellaneous services	89	15	0	0	0
Administration, environmental	95	5	8	3	158
Total ^b		346	507	152	7,604

^aConstruction and preoperation employment effects are given in total person-years of employment; employment values for operation and maintenance represent annual effects.

^bColumns do not add to totals because of rounding.

Source: Bean 1990.

TABLE N.9 Regional Socioeconomic Impacts of the MHTGR Technology at SRS

Category	Economic Impact (millions of 1988 dollars)				Employment Effect (workers)
	SIC Number	Primary Cost	Output Effect	Income Effect	
Construction and preoperation ^a					
Mining	10	59	88	24	1,040
General contracting	15	535	933	310	16,300
Heavy construction	16	217	379	126	6,610
Special trade	17	334	581	193	10,100
Chemicals	28	208	348	68	2,720
Stone and clay	32	0	0	0	0
Primary metal industries	33	6	10	3	125
Fabricated metal products	34	470	687	191	8,930
Machinery, excluding electrical	35	53	83	25	1,160
Electronics	36	32	50	13	623
Instruments and related equipment	38	7	12	3	188
Electricity, gas, and sanitation	49	0	0	0	0
Wholesale trade, durables	50	45	73	26	1,190
Wholesale trade, nondurables	51	45	73	26	1,190
Engineering and accounting	87	1,340	2,170	984	56,200
Administration, environmental	95	11	18	6	347
Administration, other	96	32	53	19	1,010
Total ^b		3,400	5,560	2,020	107,733
Annual operation and maintenance					
Mining	10	11	16	4	191
Chemicals	28	119	199	38	1,560
Primary metal industries	33	33	72	107	1,300
Electricity, gas, and sanitation	49	50	56	8	351
Wholesale trade	51	11	0	0	0
Eating and drinking establishments	58	1	2	0	79
Insurance carriers	63	0	1	0	23
Business services	73	35	57	25	1,470
Engineering and accounting	87	76	124	56	3,220
Miscellaneous services	89	15	25	11	647
Administration, environmental	95	6	6	2	127
Total ^b		399	596	182	8,968

^aConstruction and preoperation employment effects are given in total person-years of employment; employment values for operation and maintenance represent annual effects.

^bColumns do not add to totals because of rounding.

Source: Bean 1990.

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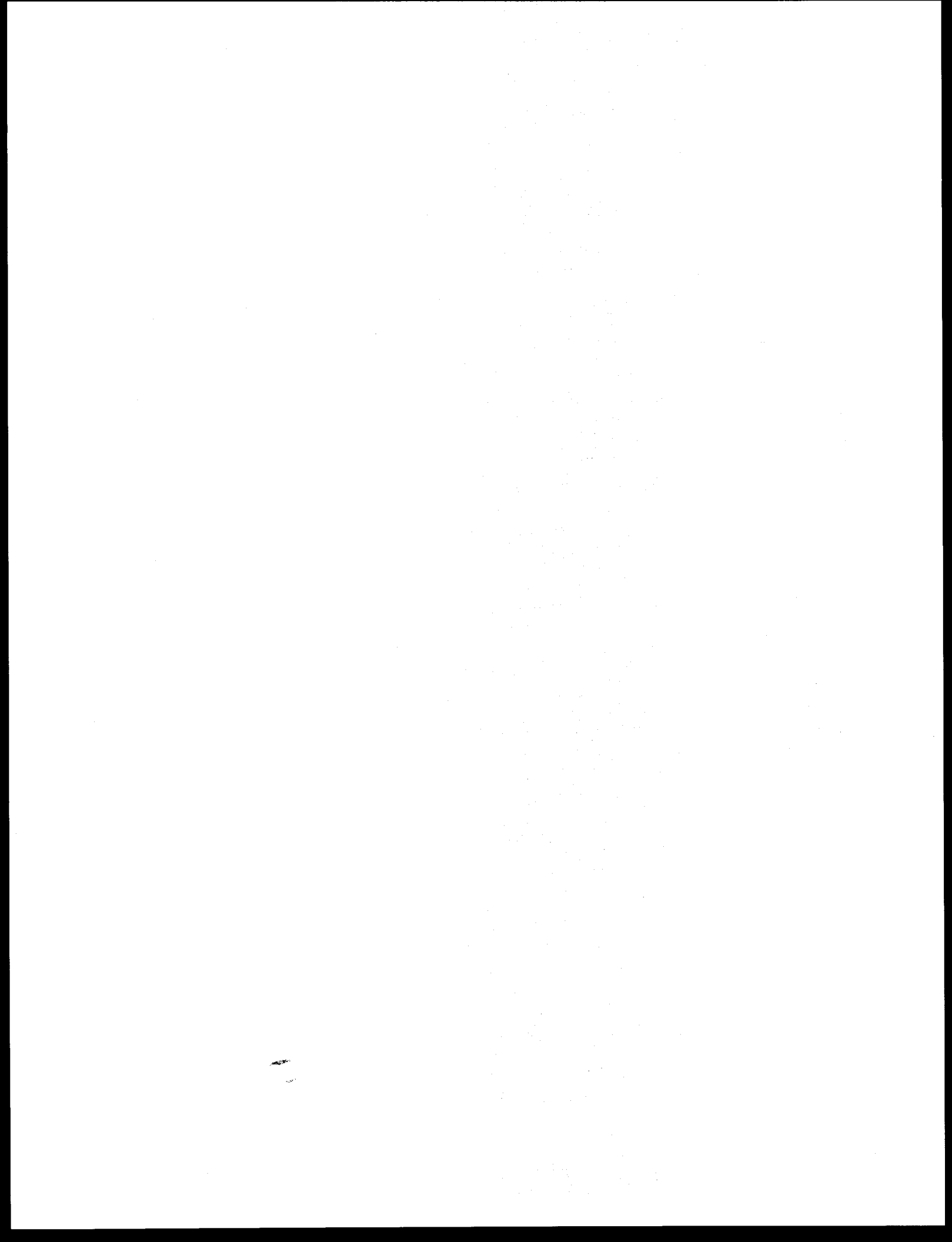
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APPENDIX O:
DECONTAMINATION AND DECOMMISSIONING



APPENDIX O: DECONTAMINATION AND DECOMMISSIONING

O.1 INTRODUCTION

Appendix O provides a summary description of the impacts of the decontamination and decommissioning (D&D) of the New Production Reactor (NPR) and its support facilities, and the resultant impacts in terms of occupational dose, volume of wastes generated, cost, and schedule. The analysis was prepared to determine whether the impacts of D&D would preclude the siting, construction, and operation of any of the alternative reactor technologies at any of the alternative sites. D&D activities are carried out at the end of a nuclear facility's life to permit the facility to be removed safely from service and to enable the property to be used for other purposes. D&D activities can range in the scope of work required, from performing a simple radiological survey to completely dismantling and removing a radioactively contaminated facility.

It is impossible to provide exact D&D plans or impacts at this time because the detailed NPR design has not been completed and D&D will not be needed for about 50 years. However, preliminary estimates of impacts can be prepared, using a combination of reported experience with D&D in the nuclear industry and the current design of NPRs and their support facilities. Toward the end of the useful life of each NPR facility, additional environmental review would be conducted that would fully describe D&D options and their impacts.

The radiological impacts from D&D activities to the general population are expected to be insignificant. Radiological doses from D&D would result from two paths: exposure to airborne contaminants (D&D activities release small amounts of radiologically contaminated fugitive dust into the atmosphere) and exposure to the off-site transportation of D&D wastes. For the first path, all D&D activities will be regulated by DOE orders. Exposure limits to the general population would be similar to exposure limits for facility operations (Sections 5.2.7, 5.3.7, and 5.4.7). For the second path, it is likely that most, if not all, of the low-level waste from D&D will be disposed of on-site; however, any high-level or transuranic wastes recovered during D&D will be disposed of off-site. The radiological impacts of the off-site shipments would be similar to the impacts of operational waste shipments (Sections 5.2.7, 5.3.7, and 5.4.7).

Although D&D activities do not begin until the end of a facility's life, planning for D&D begins in the design phase. The NPR facilities will be designed to facilitate D&D activities, as required by DOE Order 5820.2A, *Radioactive Waste Management*, and DOE Order 6430.1A, *General Design Criteria*. In addition, the NPR requirements document (DOE 1989a) specifies design requirements that would facilitate D&D (Table O.1). These requirements include architectural and structural features that facilitate dismantling reactor components and demolishing plant structures; techniques for removing components and materials; techniques for preventing or reducing radioactive contamination; and techniques for removing, handling, segregating, and minimizing wastes. The NPR D&D design is expected to meet or exceed U.S. Nuclear Regulatory Commission (NRC) requirements for commercial nuclear facilities.

The NPR design will include a plan that addresses eventual decommissioning. The plan will establish a basis for ensuring that the proposed reactors and their support facilities would be decommissioned safely and efficiently within acceptable levels of radioactivity. The

TABLE O.1 D&D Design Features

-
- Modular, separable confinements for radioactive and other hazardous materials that preclude contamination of fixed portions of the structure.
 - Localized liquid transfer systems that avoid long runs of buried contaminated piping, including special provisions that ensure the integrity of joints in buried pipelines.
 - Exhaust filtration components of the ventilation systems at or near individual enclosures to minimize long runs of internally contaminated ductwork.
 - Equipment, including effluent decontamination equipment, that precludes the accumulation of radioactive or other hazardous materials in relatively inaccessible areas, including curves and turns in piping and ductwork.
 - Easily decontaminated materials that reduce the amount of radioactive and other hazardous materials requiring disposal.
 - Designs that ease cutup, dismantlement, removal, and packaging of contaminated equipment for the facility.
 - Modular radiation shielding in lieu of or in addition to monolithic shielding walls.
 - Lifting lugs on large tanks and equipment.
 - Fully drainable piping systems that carry contaminated or potentially contaminated liquids.
-

Source: DOE 1989a.

design-phase D&D plan will address all regulatory compliance, decommissioning options and activities, controls and limits on procedures and equipment to protect occupational and public health and safety, final radiation surveys, quality assurance, safeguards, mitigation of environmental impacts, radioactive and other hazardous wastes, disposal sites, occupational and public exposure to radiation, costs, and scheduling. The plan will specifically address the disposal of low-level, high-level, and transuranic radioactive wastes; hazardous waste storage requirements; and compliance with RCRA and CERCLA. The transportation of wastes for off-site disposal will also be addressed.

Near the end of the useful life of NPR, a final D&D plan will be prepared, using then-current technology and applicable regulations and addressing any waste disposal requirements. The final plan will address, compare, and update all issues discussed in the design phase and will be subject to NEPA review.

O.2 DECONTAMINATION AND DECOMMISSIONING OPTIONS

Both the U.S. Department of Energy (DOE) and NRC have planned and conducted D&D activities. An NRC document on decommissioning nuclear facilities (NRC 1988) summarizes NRC's experience in D&D activities and identifies and describes various D&D options. The NRC

document addresses D&D of both nuclear reactors and their support facilities, including fuel fabrication and spent fuel processing facilities. The NRC also has described ways to facilitate reactor decommissioning (Moore 1979). On June 27, 1988, NRC published its final rule on decommissioning (53 FR 24018). The final rule is intended to ensure that all licensed facilities will be decommissioned in a safe and timely manner and that adequate funds will be available for D&D activities. Although NRC requirements are not applicable to DOE facilities, the NRC documents are useful in evaluating potential D&D options for NPR facilities.

The U.S. Department of Energy performed the D&D for the Shippingport Atomic Power Station and prepared a final decommissioning report on the activities (DOE 1982). DOE also studied the decommissioning of eight reactors at Hanford (DOE 1989b) and provided a comprehensive description of all the decommissioning steps, showing how to prepare estimates of radioactivity, occupational doses, and costs (NES 1980).

The NRC and DOE documents discuss three types of decommissioning options: decontamination (removal of radioactive materials to allow decay to acceptable levels); storage to allow the radionuclides to decay for as long as 60 to 100 years, followed by decontamination; and permanent encasement of the radioactive facility in a long-lived material such as concrete to allow decay of the radioactive material to safe levels. Variations exist within these options; however, both NRC and DOE recognize the main concept. The terms, "acceptable levels of radioactivity" and "unrestricted use," denote levels of radioactivity at which the facility and surrounding land could be used by the general public with no restrictions on use or duration of occupancy.

The NPR requirements document (DOE 1989a) states that three NRC D&D options will be considered for the NPR:

- *DECON (DECONtamination)* -- the equipment, structures, and portions of the facility and site that contain radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after operations cease. Personnel radiation exposures in this option are generally higher than for other options because all work is completed within a few months or years following shutdown. In addition, more money and waste disposal space are required in a relatively short time. DECON enables the facility and site to be available for unrestricted use in a short time and reduces requirements for long-term security, maintenance, and surveillance.
- *SAFSTOR (SAFe STORage)* -- the facility is placed and maintained in a condition that allows it to be safely stored for a number of years before decontamination begins. SAFSTOR consists of a short preparation period for safe storage, a variable period of safe storage (up to 60 years), and a short decontamination period. SAFSTOR is well suited for facilities where a significant amount of radioactivity is from radionuclides with half-lives of the order of a few years. Deferring the final D&D allows most of the contamination to decay, lowering both costs and occupational exposures. However, continued maintenance and surveillance are required, and original plant personnel may not be available to provide D&D personnel with an intimate knowledge of the facility.

- **ENTOMB (ENTOMBment)** -- radioactive contaminants are encased (entombed) in a structurally sound, long-lived material such as concrete. The entombed structure is maintained, and surveillance continues until the radionuclides decay to levels that permit the property to be released for unrestricted use. ENTOMB is used when residual radionuclides will decay to low levels of radioactivity within a reasonable time (within the time of the continued structural integrity, i.e., of the order of 100 years), allowing lower personnel exposures and lower maintenance and surveillance costs. Disadvantages include the difficulty of demonstrating that the entombed radioactivity has decayed to acceptable levels after entombment, and long-term dedication of the site.

In the final generic EIS on decommissioning (NRC 1983), NRC determined that either the DECON or the SAFSTOR option with decommissioning after as long as 60 years' storage is acceptable for power reactors. For other commercial nuclear facilities, NRC determined that the DECON option is the only acceptable option unless another D&D option is necessary to protect public health and safety.

O.3 METHODOLOGY

Following the direction of the NPR requirements document (DOE 1989a), DOE developed a methodology to estimate occupational exposure, the volume of waste generated, and the cost of D&D. The final generic decommissioning document (NRC 1988) provides D&D information for several facility types, based on a consistent set of assumptions. Because facility volume is the only consistent engineering information available for all reactors and support facilities at this early phase of NPR design, information from the NRC document (NRC 1988) was used to generate volume-normalized estimation factors for occupational doses, waste volumes, and costs (Blond 1989). The duration of the D&D schedule was also based on the NRC report (NRC 1988).

Volume-normalized estimation factors were derived for three types of contaminated NPR facilities: production reactor (Table O.2), spent fuel processing and plutonium-target processing (Table O.3), and fuel fabrication facilities (Table O.4). Uncontaminated structures included in the volume totals in the tables are indicated as an "uncontaminated" estimation factor. Equipment and areas within the NPR facilities used for handling and treating waste are included in the D&D impact analysis. Facilities used for sitewide waste management, waste storage, and waste disposal activities are not included. The NPR activities would only increase the activities at these facilities by a small increment, and these facilities could be used to support sitewide waste management activities beyond the life of NPR. In addition, the sitewide waste management facilities will be addressed in DOE's Programmatic Waste Management Environmental Impact Statement. The estimation factors for the production reactor were derived from light-water reactor D&D experience and include reactor containment and attached buildings.

Spent fuel and plutonium-target processing estimation factors were based on D&D experience at commercial spent fuel processing facilities. Operations at a commercial spent fuel processing facility include receiving and storing irradiated fuel assemblies, disassembling fuel

TABLE O.2 Volume-Normalized Estimation Factors Used in D&D of Reactor Facilities

Estimation Factor for	SAFSTOR ^a				ENTOMB ^a	
	DECON	10 yr	30 yr	100 yr	Internals Included ^b	Internals Removed ^c
Cost (\$/m ³)	270	290	300	240	170	190
Occupational dose (person-rem/m ³)	3.3×10^{-3}	1.8×10^{-3}	9.0×10^{-4}	8.3×10^{-4}	2.5×10^{-3}	2.8×10^{-3}
Waste volume (m ³ /m ³)	5.0×10^{-2}	5.0×10^{-2}	5.0×10^{-2}	4.8×10^{-3}	4.7×10^{-3}	NA ^d

^aAfter the SAFSTOR period, D&D would take 6 years.

^bPressure vessel internals entombed inside containment.

^cPressure vessel internals removed from containment and disposed of off-site.

^dNA = not available.

Source: Blond 1989.

TABLE O.3 Volume-Normalized Estimation Factors Used in D&D of Spent Fuel and Plutonium-Target Processing Facilities

Estimation Factor for	DECON	SAFSTOR (Passive) ^{a,b}			SAFSTOR (Custodial) ^{a,c}		
		10 yr	30 yr	100 yr	10 yr	30 yr	100 yr
Cost (\$/m ³)	1,200	1,300	1,300	1,500	1,300	1,500	2,000
Occupational dose (person-rem/m ³)	3.4×10^{-3}	2.9×10^{-3}	2.0×10^{-3}	8.9×10^{-4}	2.9×10^{-3}	2.1×10^{-3}	1.2×10^{-3}
Waste volume (m ³ /m ³)							
HLW + TRU	3.0×10^{-2}	1.3×10^{-3}	1.3×10^{-3}	1.3×10^{-3}	1.3×10^{-3}	1.3×10^{-3}	1.3×10^{-3}
LLW	2.0×10^{-2}	1.2×10^{-3}	1.2×10^{-3}	1.2×10^{-3}	1.2×10^{-3}	1.2×10^{-3}	1.2×10^{-3}
Total	5.0×10^{-2}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}	2.5×10^{-3}

^aAfter the SAFSTOR period, D&D would take 5 years.^bPassive SAFSTOR involves a comprehensive initial decontamination effort. Structures must be secured so that electronic surveillance and inspection are adequate to ensure safety and security of the facility until final decommissioning.^cCustodial SAFSTOR involves a minimal initial decontamination effort, but requires maintenance of active systems and full-time maintenance and security personnel to ensure safety of the public and the environment until final decommissioning.

Source: Blond 1989.

**TABLE O.4 Volume-Normalized Estimation Factors
Used in D&D of Fuel Fabrication Facilities**

Estimation Factor for	DECON	SAFSTOR (Custodial) ^a	
		10 yr	30 yr
Cost (\$/m ³)	46	800	130
Occupational dose (person-rem/m ³)	8.7×10^{-5}	1.2×10^{-4}	1.7×10^{-4}
Waste volume (m ³ /m ³)	1.7×10^{-13}	1.7×10^{-14}	1.7×10^{-14}

^aAfter the SAFSTOR period, D&D would take 9 months.

Source: Blond 1989.

assemblies, chopping fuel rods into small sections, dissolving fuel rod pieces in nitric acid, dissolving and purifying uranium and plutonium, converting uranium to uranium hexafluoride and plutonium to plutonium oxide for off-site shipment, and solidifying the high-level waste for off-site storage or disposal. Commercial spent fuel processing estimation factors are also used for tritium extraction cell estimates.

Fuel fabrication estimation factors were based on the D&D experience of commercial uranium fuel fabrication facilities. Operations at a commercial uranium fuel fabrication facility include converting low-enriched uranium hexafluoride to uranium dioxide, compacting and sintering the uranium dioxide into fuel pellets, incorporating the fuel pellets into fuel clad assemblies, and recovering uranium from scrap wastes and off-standard pellets. The D&D activities at a highly enriched uranium fuel fabrication facility (such as proposed for the HWR and MHTGR) would be similar to the D&D activities at the low-enriched uranium fuel fabrication facility described by NRC. The Fuel Fabrication Facility estimation factors are also used for Tritium Purification Facility estimation factors.

To calculate the consequences of decontamination and decommissioning of NPR facilities, the volume-normalized estimation factors, summarized in Tables O.2, O.3, and O.4, were applied to the volume of each NPR facility. The volumes were obtained from the external dimensions of each facility. The structure volumes include uncontaminated areas such as offices, lunchrooms, restrooms, and clothes-changing rooms within the contaminated structure. In applying the estimation factors for the production reactor, the facility volume included the volumes of the reactor containment building, attached buildings, and other contaminated structures. For facilities in which radioactive materials were not handled, the estimation factors for nonradioactive facilities were based on the least costly DECON estimation factors for the three types of radioactive facility estimation factors. For the uncontaminated facility, the occupational dose and radioactive waste volumes were assumed to be zero.

O.4 FACILITY VOLUMES, COST, OCCUPATIONAL DOSES, AND WASTE VOLUMES

Appendixes A, B, and C discuss in detail the NPR fuel cycle and how each reactor technology would be implemented at each proposed site. The following sections present the volume requirements for each of the three reactor technologies and the facility volumes associated with each reactor technology for each alternative site.

O.4.1 Reactor Facility Volumes

Table O.5 gives the volumes of potentially contaminated buildings for specific facilities associated with each reactor technology. The total volumes of radiologically contaminated buildings vary from 140,000 m³ for the heavy-water reactor to about 520,000 m³ for the modular high-temperature gas-cooled reactor. These facility volumes will be used at each site in the appropriate technology.

O.4.2 Hanford Site

This section discusses the D&D of the three alternative reactor technologies and their support facilities at the Hanford Site. The facility volumes used by NPR, the processes in each facility, and the resultant D&D costs, occupational doses, and waste volumes generated by D&D are summarized. The LWR and HWR technologies would use the existing Fuels and Materials Examination Facility, which has a volume of 140,000 m³ (Appendix B, Section B.2.3.1).

Tables O.6, O.7, and O.8 present lists of potentially radiologically contaminated buildings and their respective volumes at Hanford. The tables also include the estimation factors used for each building. The information presented in Tables O.6, O.7, and O.8 is based mainly on data in Appendixes A, B, and C and is supplemented as indicated.

As shown in Tables O.6, O.7, and O.8 and in Figure O.1, the total volumes for tritium production facilities range from 350,000 to 700,000 m³, and for tritium and plutonium production and spent fuel processing facilities from 790,000 to 1,200,000 m³. For the Hanford LWR technology, the volume of facilities subject to D&D would be the same for the tritium production mode with spent fuel processing, and the combined tritium and plutonium production mode with spent fuel processing because both options would use the same facilities.

Figures O.2, O.3, and O.4 and Tables O.9, O.10, and O.11 show the impacts of D&D activities at Hanford in terms of cost, occupational dose, and volume of wastes. For all three reactor technologies, the tritium production mode has the lowest impacts (cost, occupational dose, and waste volumes), but impacts would increase with the addition of the spent fuel processing and plutonium production options. For the tritium production mode, the estimated cost varies from \$210 million to \$320 million; the occupational dose varies from 610 to 940 person-rem; and the estimated waste volume varies from 26,000 to 38,000 m³. For the tritium and plutonium production mode with spent fuel processing, the estimated cost ranges from \$700 million to \$830 million; the occupational dose varies from 2,000 to 2,400 person-rem. Little variation occurs between the LWR and MHTGR technologies for the tritium production mode or for the combined tritium and plutonium production mode with spent fuel processing

TABLE O.5 Radiologically Contaminated Buildings in the Reactor Facility

Facility/Technology	Approximate Volume (m ³)	Reference
Heavy-Water Reactor		
Reactor Building	57,000	WSRC 1989
Control Building	32,000	WSRC 1989
Heavy-Water Processing Facility	19,000	Kirkland 1990
Radioactive Waste Management Complex	27,000	WSRC 1991
Hot Machine Shop ^a	8,700	WSRC 1991
Total	140,000	
Light-Water Reactor		
General Service Building	350,000	Section B.2.2
Containment Building ^a	130,000	Section B.2.2
Total	480,000	
Modular High-Temperature Gas-Cooled Reactor		
Reactor buildings ^b	84,000	EG&G 1989
Silos ^b	560	EG&G 1989
Auxiliary Building ^c	6,700	EG&G 1989
Reactor Maintenance Building	370,000	EG&G 1989
Reactor Service Building	14,000	EG&G 1989
Radioactive Waste Management Complex	31,000	EG&G 1989
Hot Machine Shop	17,000	EG&G 1989
Total	520,000	

^aA 0.9-m-thick containment wall is assumed.

^bThe reactor building volume is based on eight reactor modules.

^cExcludes 84,000-m³ containment volume for each nuclear island.

because of the small variation in facility volumes. For cost and occupational dose, essentially no differences exist between the tritium production mode with spent fuel processing and the tritium and plutonium production mode with spent fuel processing for the same reactor technology because the facility volumes are nearly identical. For the Hanford Site, the reactor technology and fuel cycle option with the least impact is the HWR for tritium production. The largest impact is associated with the MHTGR for tritium and plutonium production with spent fuel processing.

TABLE O.6 Potential Radiologically Contaminated HWR Facilities at Hanford

Facility/Mode	Length (m)	Width (m)	Height (m)	Approximate Volume (m ³)	Estimation Factor ^a	Reference
Tritium Production						
Tritium-Target Fabrication Facility	61	122	10 ^b	74,000	Uncontaminated and fuel fabrica- tion	Section A.3.3; Figure A.14
Reactor Facility	--	--	--	140,000 ^c	Reactor	--
Fuels and Materials Examination Facility	79	46	38	140,000	Fuel processing	Section B.2.3
Total (tritium production only)				350,000		
Tritium Production and Spent Fuel Processing						
Fuel Processing Building	142	76	37	400,000	Fuel processing	Section A.3.5
Uranium Recovery Building	30	27	14	11,000	Fuel processing	Section A.3.5
Total (tritium production plus spent fuel processing)				760,000		
Tritium and Plutonium Production and Spent Fuel Processing						
Plutonium-Target Fabrication Facility ^d	--	--	10 ^b	33,000	Fuel fabrication	Section A.3.4; WHC 1989
Total (tritium and plutonium production plus spent fuel processing)				790,000		

^aSee Table O.2.^bAssumed height.^cSee Table O.5.^dWHC (1989) states 3,252-m² area.

TABLE O.7 Potential Radiologically Contaminated LWR Facilities at Hanford

Facility/Mode	Length (m)	Width (m)	Height (m)	Approximate Volume (m ³)	Estimation Factor ^a	Reference
Tritium Production Fuels and Materials Examination Facility	79	46	38	140,000	Uncontaminated, fuel fabrication, and fuel processing Reactor	Section B.2.3
Reactor Facility Total (tritium production only)	--	--	--	480,000 ^b 620,000		--
Tritium Production and Spent Fuel Processing Main Processing Building	142	76	37	400,000	Fuel processing	Sections A.3.5 and B.2.5
Uranium Recovery Building	30	27	14	11,000	Fuel processing	Sections A.3.5 and B.2.5
Total (tritium production plus spent fuel processing)				1,000,000		

^aSee Table O.2.

^bSee Table O.5.

TABLE O.8 Potential Radiologically Contaminated MHTGR Facilities at Hanford

Facility/Mode	Length (m)	Width (m)	Height (m)	Approximate Volume (m ³)	Estimation Factor ^a	Reference
Tritium Production						
Tritium-Target Fabrication Facility	49	30	9	13,000	Uncontaminated	Sections C.2.3 and C.2.4
Fuel Fabrication Facility	70	46	9	29,000	Fuel fabrication	Sections C.2.3 and C.3.3
Reactor Facility ^b	--	--	--	520,000	Reactor	--
Fuels and Materials Examination Facility	79	46	38	140,000	Fuel processing	Sections B.2.3 and C.3.6
Total (tritium production only)				700,000		
Tritium Production and Spent Fuel Processing						
Fuel Processing Facility ^c	--	--	37	410,000	Fuel processing	Section C.3.5; Figure C.15
Uranium Recovery Building	30	27	14	11,000	Fuel processing	Section C.3.5
Total (tritium production plus spent fuel processing)				1,100,000		
Tritium and Plutonium Production and Spent Fuel Processing						
Plutonium-Target Fabrication Facility	76	46	9	31,000	Fuel fabrication	Sections C.2.4 and C.3.4
Total (tritium and plutonium production plus spent fuel processing)				1,200,000		

^aSee Table O.2.^bSee Table O.5.^cThe 11,000-m² floor area is calculated on the basis of Figure C.15.

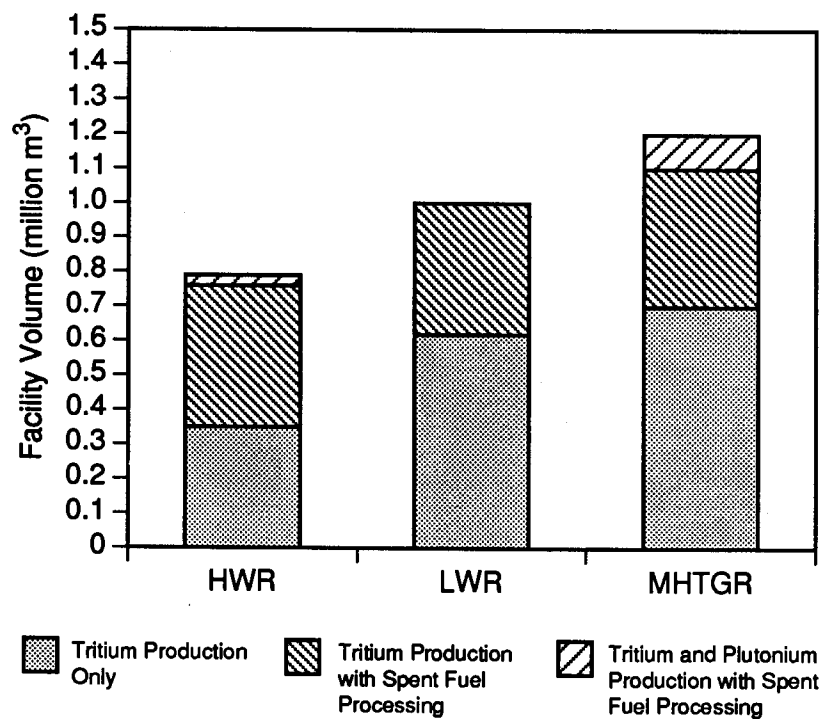


FIGURE O.1 Facility Volumes for D&D at Hanford for the HWR, LWR, and MHTGR Technologies

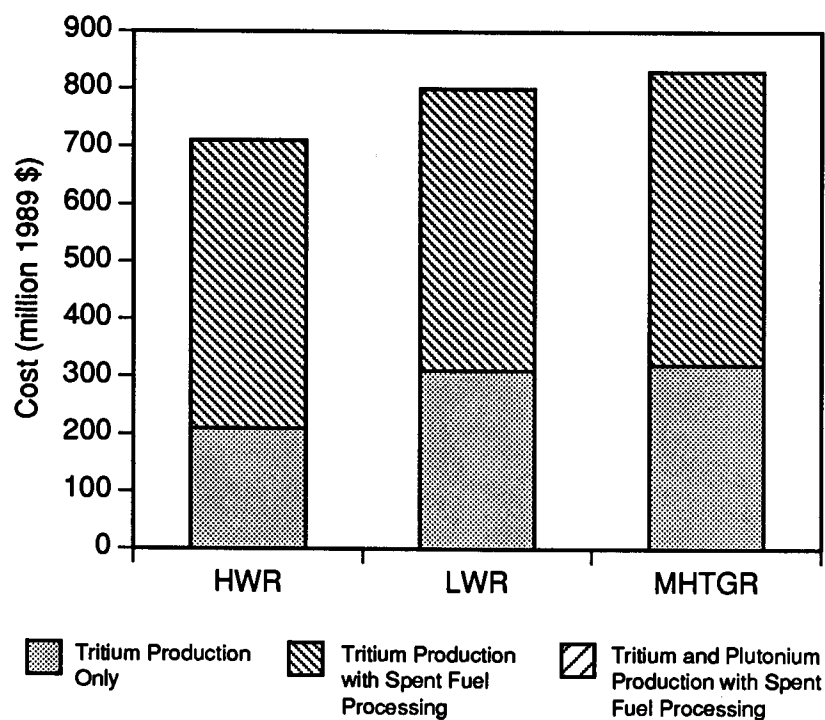


FIGURE O.2 D&D Cost at Hanford for the HWR, LWR, and MHTGR Technologies

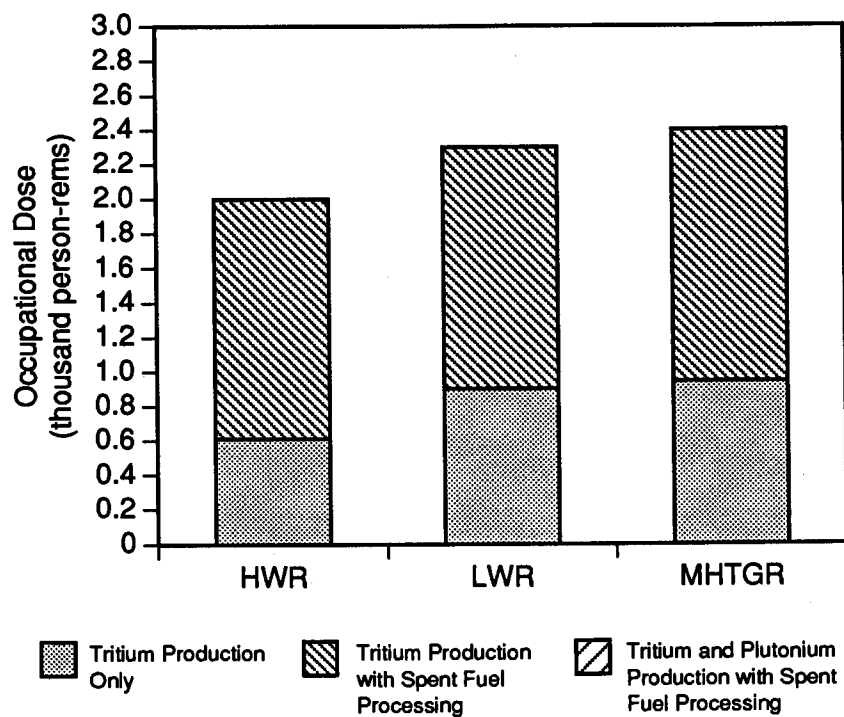


FIGURE O.3 D&D Occupational Dose at Hanford for the HWR, LWR, and MHTGR Technologies

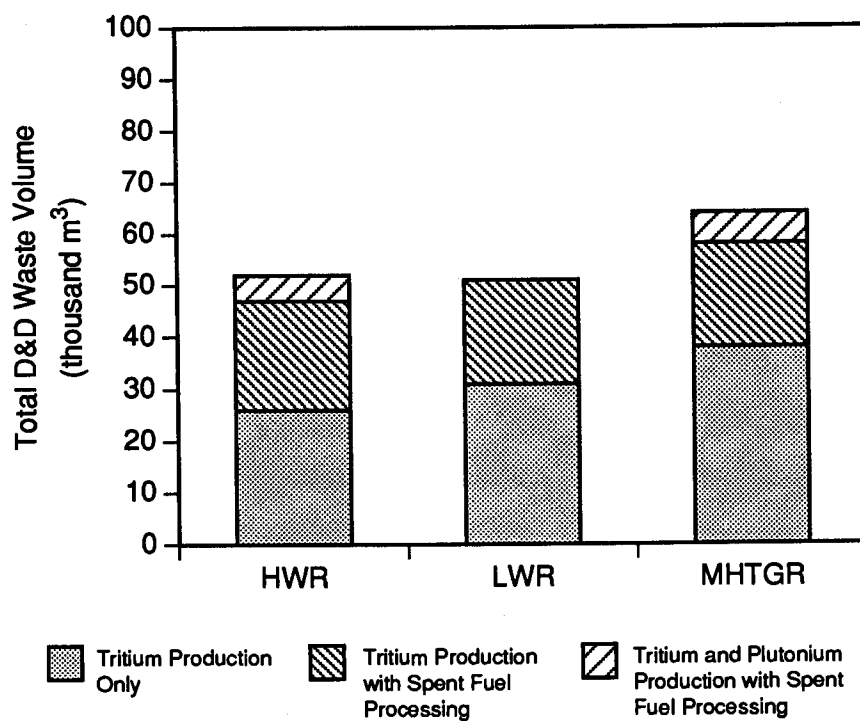


FIGURE O.4 D&D Total Waste Volume at Hanford for the HWR, LWR, and MHTGR Technologies

TABLE O.9 D&D Impacts of the HWR at Hanford

Facility/Mode	Estimation Factor ^a	Approximate Volume (m ³)	Cost (millions of 1989 dollars)	Occupational Dose (person-rem)	Waste Volume (m ³)		
					HLW + TRU	LLW	Total
Tritium Production							
Tritium-Target Fabrication Facility	Fuel fabrication	74,000	3.4	6.4	NA ^b	NA	13,000
Reactor Facility	Reactor	140,000	42	130	NA	NA	7,000
Fuels and Materials Examination Facility	Fuel processing	140,000	170	480	4,200	2,800	6,900
Total (tritium production only)		350,000	210	610	4,200	2,800	26,000
Tritium Production and Spent Fuel Processing							
Fuel Processing Building	Fuel processing	400,000	480	1,400	12,000	8,000	20,000
Uranium Recovery Building	Fuel processing	11,000	13	37	330	220	540
Total (tritium production plus spent fuel processing)		760,000	700	2,000	17,000	11,000	47,000
Tritium and Plutonium Production and Spent Fuel Processing							
Plutonium-Target Fabrication Facility	Fuel fabrication	33,000	1.5	2.9	NA	NA	5,600
Total (tritium and plutonium production plus spent fuel processing)		790,000	700	2,000	17,000	11,000	53,000

^aSee Table O.2.

^bNA = not available.

TABLE O.10 D&D Impacts of the LWR at Hanford

Facility/Mode	Estimation Factor ^a	Approximate Volume (m ³)	Cost (millions of 1989 dollars)	Occupational Dose (person-rem)	Waste Volume (m ³)		
					HLW + TRU	LLW	Total
Tritium Production							
Fuels and Materials Examination Facility	Fuel processing	140,000	170	470	4,100	2,800	6,900
Reactor Facility	Reactor	480,000	140	430	NA ^b	NA	24,000
Total (tritium production only)		620,000	310	900	4,100	2,800	31,000
Tritium Production and Spent Fuel Processing							
Main Processing Building	Fuel processing	400,000	480	1,400	12,000	8,000	20,000
Uranium Recovery Building	Fuel processing	11,000	14	39	340	230	560
Total (tritium production plus spent fuel processing)		1,000,000	800	2,300	16,000	11,000	51,000

^aSee Table O.2.^bNA = not available.

TABLE O.11 D&D Impacts of the MHTGR at Hanford

Facility/Mode	Estimation Factor ^a	Approximate Volume (m ³)	Cost (millions of 1989 dollars)	Occupational Dose (person-rem)	Waste Volume (m ³)		
					HLW + TRU	LLW	Total
Tritium Production							
Tritium-Target Fabrication Facility	Uncontaminated	13,000	0.61	0	0	0	0
Fuel Fabrication Facility	Fuel fabrication	29,000	1.3	2.5	NA ^b	NA	4,900
Reactor Facility	Reactor	520,000	160	470	NA	NA	26,000
Fuels and Materials Examination Facility	Fuel processing	140,000	170	470	4,100	2,800	6,900
Total (tritium production only)		700,000	320	940	4,100	2,800	38,000
Tritium Production and Spent Fuel Processing							
Fuel Processing Building	Fuel processing	410,000	490	1,400	12,000	8,100	20,000
Uranium Recovery Building	Fuel processing	11,000	14	39	340	230	560
Total (tritium production plus spent fuel processing)		1,100,000	830	2,400	16,000	16,000	59,000
Tritium and Plutonium Production and Spent Fuel Processing							
Plutonium-Target Fabrication Facility	Fuel fabrication	31,000	1.4	2.7	NA	NA	5,300
Total (tritium and plutonium production plus spent fuel processing)		1,200,000	830	2,400	16,000	11,000	64,000

^aSee Table O.2.

^bNA = not available.

O.4.3 Idaho National Engineering Laboratory

This section discusses the D&D of the three alternative reactor technologies and their support facilities at INEL. The facility volumes used by NPR, the processes in each facility, and the resultant D&D costs, occupational doses, and waste volumes generated by D&D are summarized. The HWR and MHTGR technologies would use the fuel processing building. In addition, the Fuel Storage Area of the Fluorinal Dissolution Process and Fuel Storage Facility would store spent fuel for the HWR and LWR technologies.

Tables O.12, O.13, and O.14 list potentially radiologically contaminated INEL buildings and their respective volumes; they also give the estimation factors used for each building. The information in the tables is based on data in Appendixes A, B, and C. As indicated in these tables and Figure O.5, the building volumes for the tritium production mode range from 240,000 to 600,000 m³, and for the combined tritium and plutonium production mode with spent fuel processing from 520,000 to 1,000,000 m³. The volume of facilities subject to D&D for the LWR technology at INEL is the same for the tritium production mode with spent fuel processing and the combined tritium and plutonium production mode with spent fuel processing because both options would use the same facilities.

Figures O.6, O.7, and O.8 and Tables O.15, O.16, and O.17 show the impacts of D&D activities at INEL in terms of cost, occupational dose, and waste volume. For the tritium production mode, the estimated cost varies from \$56 million to \$220 million; the occupational dose varies from 170 to 640 person-rem; and the estimated waste volume varies from 22,000 to 36,000 m³. For the combined tritium and plutonium production mode with spent fuel processing, the estimated cost ranges from \$340 million to \$710 million; the occupational dose varies from 970 to 2,000 person-rem; and the waste volume varies from 42,000 to 57,000 m³. At INEL for all three reactor technologies, the tritium production mode has the least impact; impacts would increase as spent fuel processing and plutonium production options are added. D&D impacts for the LWR would be the same for either tritium production with spent fuel processing or tritium and plutonium production with spent fuel processing because the facility volumes are the same for either option. For the INEL Site, the reactor technology and fuel cycle option with the least impact is the HWR for tritium production. The largest impact is associated with the LWR for tritium and plutonium production with spent fuel processing.

O.4.4 Savannah River Site

This section discusses the D&D of the three alternative reactor technologies and their support facilities at SRS. The facility volumes used by NPR, the processes in each facility, and the resultant D&D costs, occupational doses, and waste volumes generated by D&D are summarized. The three alternative reactors would use several existing or planned facilities at SRS, including buildings 313-M, 320-M, 321-M, 221-H (H-Canyon), 221-F (F-Canyon), and 233-H. D&D activities for these facilities are included in the EIS because they would be used by NPR.

Tables O.18, O.19, and O.20 present lists of potentially radiologically contaminated buildings and their respective volumes at SRS. The information is based on data contained in Appendixes A, B, and C. The tables also include the estimation factors used for each building. As indicated in the tables and in Figure O.9, the building volumes for the tritium production mode range from 300,000 to 570,000 m³, and for the combined tritium and plutonium production mode with spent fuel processing from 900,000 to 1,100,000 m³.

TABLE O.12 Potential Radiologically Contaminated HWR Facilities at INEL

Facility/Mode	Length (m)	Width (m)	Height (m)	Approximate Volume (m ³)	Estimation Factor ^a	Reference
Tritium Production						
Tritium-Target Fabrication Facility ^b	--	--	10 ^c	86,000	Uncontaminated and fuel fabrica- tion	Section A.4.3
Reactor Facility ^d	--	--	--	140,000	Reactor	--
Tritium-Target Processing Facility	--	--	--	8,100	Fuel processing	Section A.4.6; Kirkland 1990
Total (tritium production only)				230,000		
Tritium Production and Spent Fuel Processing						
Headend Fuel Processing Facility	66	24	24	38,000	Fuel processing	Section A.4.5
Fuel Processing Facility						
Above ground	75.6	65.5	18 ^c	89,000	Fuel processing	Section A.2.6
Below ground	54.9	75.6	18 ^c	75,000	Fuel processing	Section A.2.6
Total (tritium production plus spent fuel processing)				430,000		
Tritium and Plutonium Production and Spent Fuel Processing						
Plutonium-Target Fabrication Facility ^e	--	--	10 ^c	51,000	Fuel fabrication	Section A.4.4
Plutonium-Target Processing Facility	75	43	10 ^c	32,000	Fuel processing	Sections C.2.7 and 5.9.1; EG&G 1989
Total (tritium and plutonium production plus spent fuel processing)				510,000		

^aSee Table O.2.

^bSection A.4.3 states 8,600-m² area.

^cAssumed height.

^dSee Table O.5.

^eSection A.4.4 states 5,100-m² area.

TABLE O.13 Potential Radiologically Contaminated LWR Facilities at INEL

Facility/Mode	Length (m)	Width (m)	Height (m)	Approximate Volume (m ³)	Estimation Factor ^a	Reference
Tritium Production						
Tritium-Target Fabrication Facility ^b	--	--	10 ^c	4,600	Uncontaminated	Section B.3.3
Fuel Fabrication Facility ^d	--	--	10 ^c	56,000	Fuel fabrication	Section B.3.3
Reactor Facility ^e	--	--	--	480,000	Reactor	--
Tritium-Target Processing Facility ^f	--	--	21	59,000	Fuel processing	Section B.3.6
Total (tritium production only)				600,000		
Tritium Production and Spent Fuel Processing						
Fuel Processing Building	142	76	37	400,000	Fuel processing	Sections A.3.5 and B.3.6
Uranium Recovery Building	30	27	14	11,000	Fuel processing	Sections A.3.5 and B.3.6
Total (tritium production plus spent fuel processing)				1,000,000		

^aSee Table O.2.^bSection B.3.3 states 460-m² area.^cAssumed height.^dSection B.3.3 states 5,600-m² area.^eSee Table O.5.^fSection B.3.6 states 2,790-m² area.

TABLE O.14 Potential Radiologically Contaminated MHTGR Facilities at INEL

Facility/Mode	Length (m)	Width (m)	Height (m)	Approximate Volume (m ³)	Estimation Factor ^a	Reference
Tritium Production						
Tritium-Target Fabrication Facility	49	30	9	13,000	Uncontaminated	Section C.2.3
Fuel Fabrication Facility	70	46	9.1	29,000	Fuel fabrication	Section C.2.3
Reactor Facility ^{a,b}	-	-	-	520,000	Reactor	-
Tritium-Target Processing Facility	49	24	21	25,000	Fuel processing	Section C.2.6
Total (tritium production only)				590,000		
Tritium Production and Spent Fuel Processing						
Headend Fuel Processing Facility	76	26	24	47,000	Fuel processing	Section C.2.5
Fuel Processing Building						
Above ground ^c	75.6	65.5	18	89,000	Fuel processing	Section C.2.5
Below ground ^c	54.9	75.6	18	75,000	Fuel processing	Section C.2.5
Total (tritium production plus spent fuel processing)				800,000		
Tritium and Plutonium Production and Spent Fuel Processing						
Plutonium-Target Fabrication Facility	76	46	9	31,000	Fuel fabrication	Section C.2.4
Plutonium-Target Processing Facility	75	43	15 ^c	48,000	Fuel processing	Section C.2.7; EG&G 1989
Total (tritium and plutonium production plus spent fuel processing)				880,000		

^aSee Table O.2.

^bSee Table O.5.

^cAssumed height.

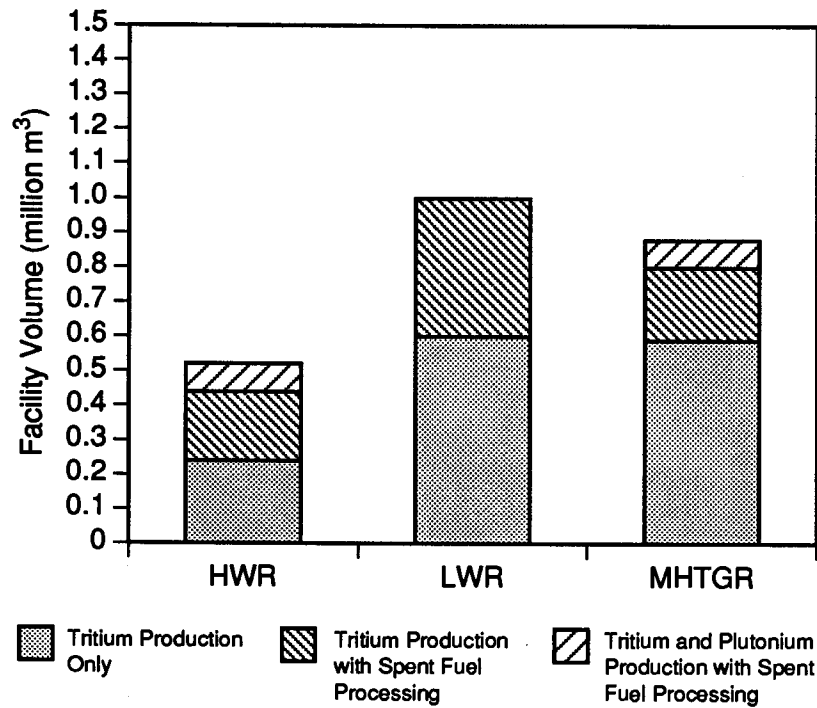


FIGURE O.5 Facility Volumes for D&D at INEL for the HWR, LWR, and MHTGR Technologies

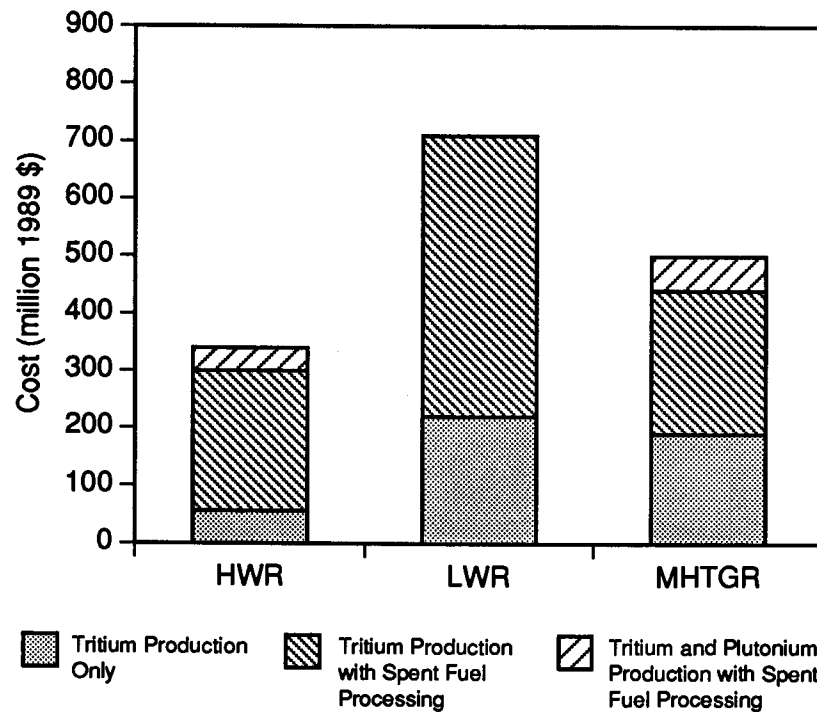


FIGURE O.6 D&D Cost at INEL for the HWR, LWR, and MHTGR Technologies

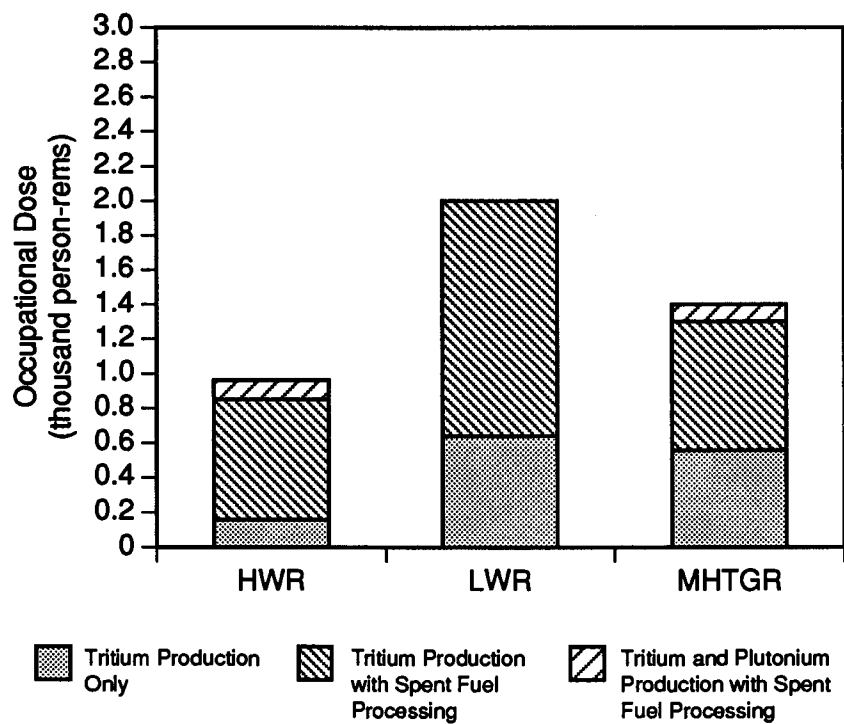


FIGURE O.7 D&D Occupational Dose at INEL for the HWR, LWR, and MHTGR Technologies

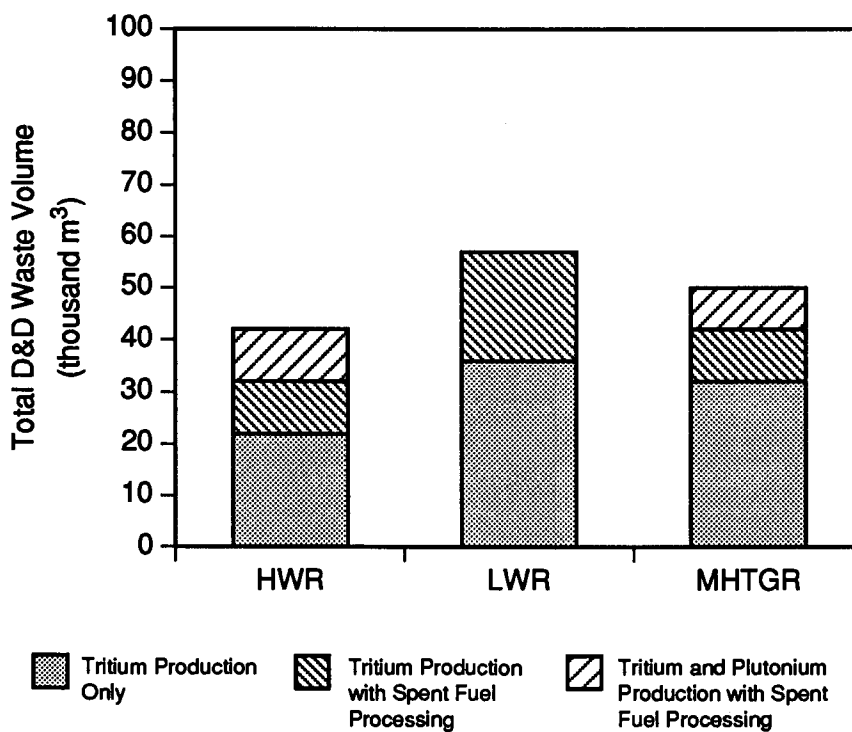


FIGURE O.8 D&D Total Waste Volume at INEL for the HWR, LWR, and MHTGR Technologies

TABLE O.15 D&D Impacts of the HWR at INEL

Facility/Mode	Estimation Factor ^a	Approximate Volume (m ³)	Cost (millions of 1989 dollars)	Occupational Dose (person-rem)	Waste Volume (m ³)		
					HLW + TRU	LLW	Total
Tritium Production							
Tritium-Target Fabrication Facility	Fuel fabrication	86,000	4.0	7.5	NA ^b	NA	15,000
Reactor Facility	Reactor	140,000	42	130	NA	NA	7,000
Tritium-Target Processing Facility	Fuel processing	8,100	9.7	28	240	160	400
Total (tritium production only)		240,000	56	170	240	160	22,000
Tritium Production and Spent Fuel Processing							
Headend Fuel Processing Facility	Fuel processing	38,000	46	130	1,100	760	1,900
Fuel Processing Facility	Fuel processing	160,000	200	560	4,900	3,300	8,000
Total (tritium production plus spent fuel processing)		440,000	300	860	6,300	4,200	32,000
Tritium and Plutonium Production and Spent Fuel Processing							
Plutonium-Target Fabrication Facility	Fuel fabrication	51,000	2.3	4.4	NA	NA	8,700
Plutonium-Target Processing Facility	Fuel processing	32,000	37	110	930	620	1,500
Total (tritium and plutonium production plus spent fuel processing)		520,000	340	970	7,200	4,800	42,000

^aSee Table O.2.^bNA = not available.

TABLE O.16 D&D Impacts of the LWR at INEL

Facility/Mode	Estimation Factor ^a	Approximate Volume (m ³)	Cost (millions of 1989 dollars)	Occupational Dose (person-rem)	Waste Volume (m ³)		
					HLW + TRU	LLW	Total
Tritium Production							
Tritium-Target Fabrication Facility	Uncontaminated	4,600	0.21	0	0	0	0
Fuel Fabrication Facility	Fuel fabrication	56,000	2.6	4.9	NA ^b	NA	9,500
Reactor Facility	Reactor	480,000	140	430	NA	NA	24,000
Tritium-Target Processing Facility	Fuel processing	59,000	71	200	1,800	1,200	2,900
Total (tritium production only)		600,000	220	640	1,800	1,200	36,000
Tritium Production and Spent Fuel Processing							
Fuel Processing Facility	Fuel processing	400,000	480	1,400	12,000	8,000	20,000
Uranium Recovery Building	Fuel processing	11,000	13	37	330	220	540
Total (tritium production plus spent fuel processing)		1,000,000	710	2,000	14,000	9,400	57,000

^aSee Table O.2.

^bNA = not available.

TABLE O.17 D&D Impacts of the MHTGR at INEL

Facility/Mode	Estimation Factor ^a	Approximate Volume (m ³)	Cost (millions of 1989 dollars)	Occupational Dose (person-rem)	Waste Volume (m ³)		
					HLW + TRU	LLW	Total
Tritium Production							
Tritium-Target Fabrication Facility	Uncontaminated	13,000	0.61	0	0	0	0
Fuel Fabrication Facility	Fuel fabrication	29,000	1.3	2.5	NA ^b	NA	4,900
Reactor Facility	Reactor	520,000	160	470	NA	NA	26,000
Tritium-Target Processing Facility	Fuel processing	25,000	30	85	750	500	1,200
Total (tritium production only)		590,000	190	560	750	500	32,000
Tritium Production and Spent Fuel Processing							
Headend Fuel Processing Facility	Fuel processing	47,000	56	160	1,400	940	2,300
Fuel Processing Facility	Fuel processing	160,000	200	560	4,900	3,300	8,000
Total (tritium production plus spent fuel processing)		800,000	440	1,300	7,100	4,700	42,000
Tritium and Plutonium Production and Spent Fuel Processing							
Plutonium-Target Fabrication Facility	Fuel fabrication	31,000	1.4	2.7	NA	NA	5,300
Plutonium-Target Processing Facility	Fuel processing	48,000	58	160	1,400	960	2,400
Total (tritium and plutonium production plus spent fuel processing)		880,000	500	1,400	8,500	5,700	50,000

^aSee Table O.2.^bNA = not available.

TABLE O.18 Potential Radiologically Contaminated HWR Facilities at SRS

Facility/Mode	Length (m)	Width (m)	Height (m)	Approximate Volume (m ³)	Estimation Factor ^a	Reference
Tritium Production						
Building 320-M	64	48	20 ^b	61,000	Uncontaminated	Section A.2.3
Building 321-M	122	38	20 ^b	93,000	Fuel fabrication	Section A.2.3
Reactor Facility ^c	--	--	--	140,000	Reactor	--
Building 233-H	--	--	--	8,100	Fuel processing	Kirkland 1990
Total (tritium production only)				300,000		
Tritium Production and Spent Fuel Processing						
Building 221-H	255	37	20	190,000	Fuel processing	Section A.2.5
Total (tritium production plus spent fuel processing)				490,000		
Tritium and Plutonium Production and Spent Fuel Processing						
Building 313-M	134	73	20 ^b	200,000	Fuel fabrication	Section A.2.4
Building 221-F	255	40.3	20	210,000	Fuel processing	Section A.2.7
Total (tritium and plutonium production plus spent fuel processing)				900,000		

^aSee Table O.2.

^bAssumed height.

^cSee Table O.5.

TABLE O.19 Potential Radiologically Contaminated LWR Facilities at SRS

Facility/Mode	Length (m)	Width (m)	Height (m)	Approximate Volume (m ³)	Estimation Factor ^a	Reference
Tritium Production						
Tritium-Target Fabrication Facility ^b	--	--	10 ^c	4,600	Uncontaminated	Section B.3.3
Fuel Fabrication Facility ^d	--	--	10 ^c	56,000	Fuel fabrication	Section B.3.3
Reactor Facility ^e	--	--	--	480,000	Reactor	--
Building 233-H	--	--	--	8,100	Fuel processing	Kirkland 1990
Total (tritium production only)				550,000		
Tritium Production and Spent Fuel Processing						
Headend Fuel Processing Facility	55	30	37	61,000	Fuel processing	Section B.4.5
Building 221-H	255	37	20	190,000	Fuel processing	Sections A.2.5 and B.4.5
Uranium Recovery Building	30	14	14	5,900	Fuel processing	Section B.4.5
Total (tritium production plus spent fuel processing)				810,000		
Tritium and Plutonium Production and Spent Fuel Processing						
Headend Fuel Processing Facility	55	30	37	61,000	Fuel processing	Sections B.4.5 and B.4.7
Building 221-F	255	40.3	20	210,000	Fuel processing	Sections A.2.7 and B.4.7
Uranium Recovery Building	30	14	14	5,900	Fuel processing	Sections B.4.5 and B.4.7
Total (tritium and plutonium production plus spent fuel processing)				1,100,000		

^aSee Table O.2.^bSection B.3.3 states 460-m² area.^cAssumed height.^dSection B.3.3 states 5,600-m² area.^eSee Table O.5.

TABLE O.20 Potential Radiologically Contaminated MHTGR Facilities at SRS

Facility/Mode	Length (m)	Width (m)	Height (m)	Approximate Volume (m ³)	Estimation Factor ^a	Reference
Tritium Production						
Tritium-Target Fabrication Facility	49	30	9	13,000	Uncontaminated	Sections C.2.3 and C.4.3
Fuel Fabrication Facility	70	46	9	29,000	Fuel fabrication	Sections C.2.3 and C.4.3
Reactor Facility ^b	--	--	--	520,000	Reactor	--
Building 233-H	--	--	--	8,100	Fuel processing	Kirkland 1990
Total (tritium production only)				570,000		
Tritium Production and Spent Fuel Processing						
Headend Fuel Processing Facility	76	26	24	47,000	Fuel processing	Sections C.2.5 and C.4.5
Building 221-H	255	37	20	190,000	Fuel processing	Sections A.2.5 and C.4.5
Total (tritium production plus spent fuel processing)				810,000		
Tritium and Plutonium Production and Spent Fuel Processing						
Plutonium-Target Fabrication Facility	76	46	9	31,000	Fuel fabrication	Sections C.2.4 and C.4.4
Headend Fuel Processing Facility	73	27	24	47,000	Fuel processing	Section C.4.7
Building 221-F	255	40.3	20	210,000	Fuel processing	Section A.2.7
Total (tritium and plutonium production plus spent fuel processing)				1,100,000		

^aSee Table O.2.

^bSee Table O.5.

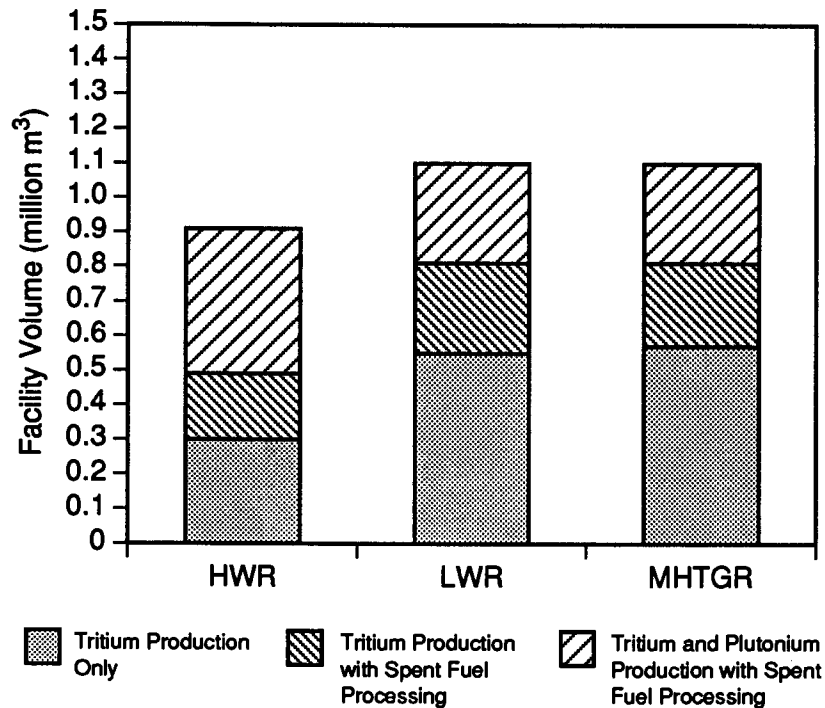


FIGURE O.9 Facility Volumes for D&D at SRS for the HWR, LWR, and MHTGR Technologies

Figures O.10, O.11, and O.12 and Tables O.21, O.22, and O.23 show the impacts of D&D activities at SRS in terms of cost, occupational dose, and waste volume. For the tritium production mode, the estimated cost varies from \$60 million to \$170 million; the occupational dose varies from 160 to 500 person-rem; and the estimated waste volume varies from 23,000 to 34,000 m³. For the combined tritium and plutonium production mode with spent fuel processing, the estimated cost varies from \$560 million to \$800 million; the occupational dose varies from 1,600 to 2,300 person-rem; and the waste volume varies from 60,000 to 77,000 m³. At SRS for all three reactor technologies, the tritium production mode has the least impact; impacts increase as spent fuel processing and plutonium production options are added. For any given fuel cycle option, the HWR has the least impact, and the LWR has the greatest impact. The MHTGR impacts are slightly less than the LWR impacts. For SRS, the reactor technology and fuel cycle option with the least impact is the HWR for the tritium production mode, while the largest cost and occupational dose is the LWR in the tritium and plutonium production mode with spent fuel processing. The highest waste volume occurs in the HWR combined tritium and plutonium production mode with spent fuel processing.

O.4.5 Summary of D&D Impacts

When comparing all reactor technologies and site combinations for the tritium production mode, the heavy-water reactor at INEL has the lowest cost (\$57 million), followed by the heavy-water reactor at SRS (\$60 million). The light-water reactor and modular high-temperature gas-cooled reactor at SRS would cost \$160 million to \$170 million. The remaining five combinations would cost between \$210 million and \$320 million. Similar trends are demonstrated for the occupational dose and total waste volume.

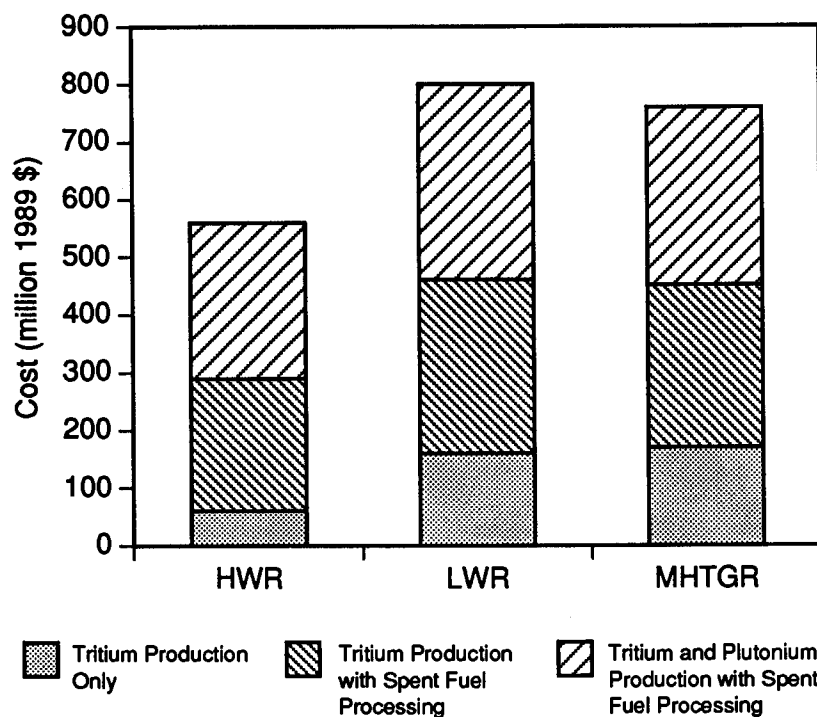


FIGURE O.10 D&D Cost at SRS for the HWR, LWR, and MHTGR Technologies

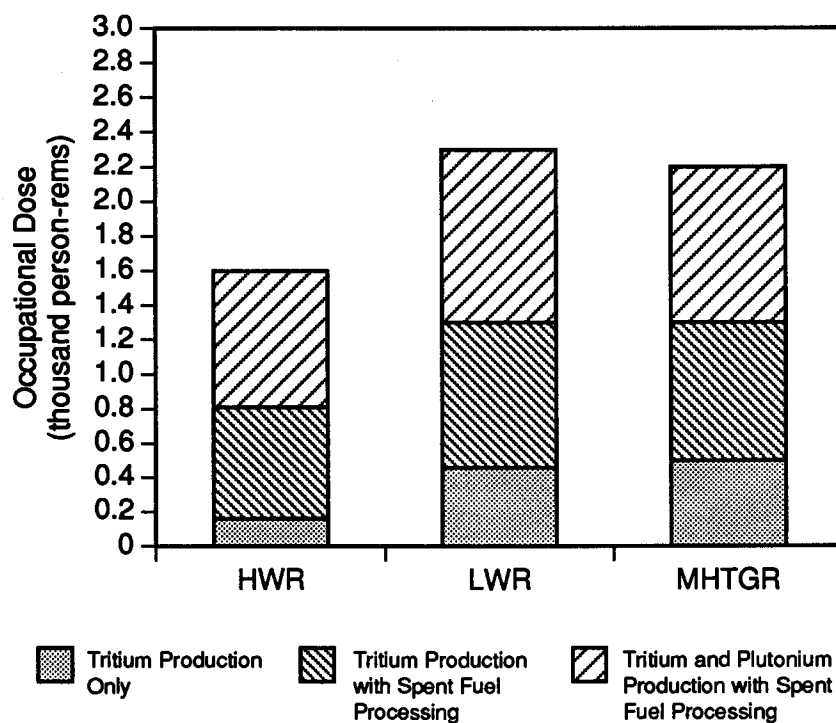


FIGURE O.11 D&D Occupational Dose at SRS for the HWR, LWR, and MHTGR Technologies

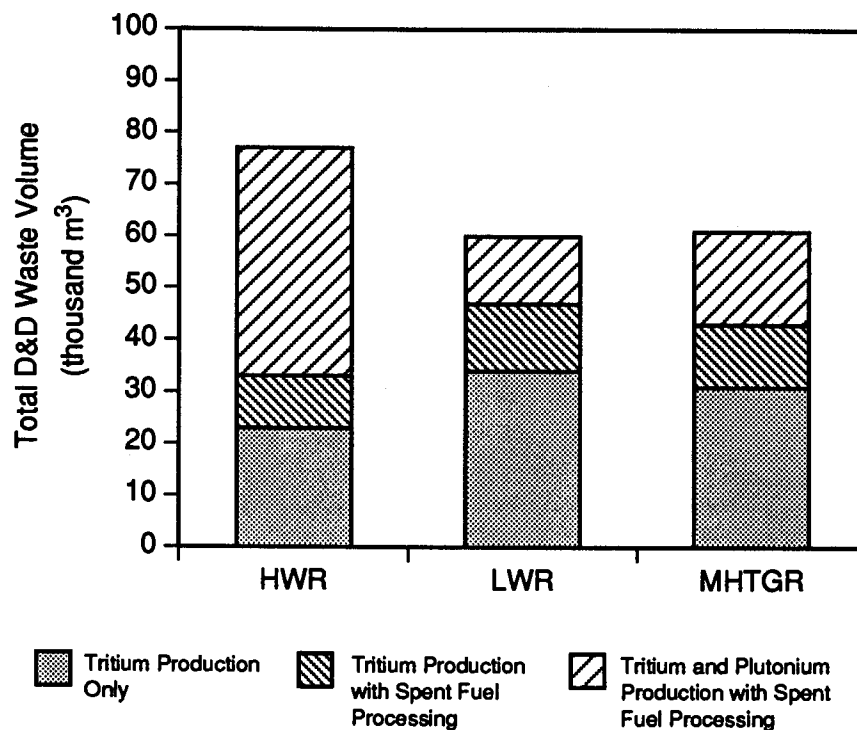


FIGURE O.12 D&D Total Waste Volume at SRS for the HWR, LWR, and MHTGR Technologies

For the combined tritium and plutonium production mode with spent fuel processing, the heavy-water reactor at INEL would have the lowest cost (\$340 million), followed by the modular high-temperature gas-cooled reactor at INEL (\$500 million). The heavy-water reactor at SRS would cost \$560 million. The costs of the remaining six combinations range from \$710 million to \$830 million. Similar trends occur for the occupational dose and total waste volume.

O.5 SCHEDULES FOR DECONTAMINATION AND DECOMMISSIONING

This section discusses estimates of the duration of decontamination activities for the various facilities (Table O.5) at each site. Decontamination activities would begin as soon as possible after the facility closes. It is assumed that fuel and target fabrication facilities would cease operation 1 year before reactor shutdown. The reactor and its support facilities would shut down after 40 years of operation. It is also assumed that fuel and target processing would cease after processing the last batch of targets or fuel. Processing would begin after the reactor has been shut down for 1 year (the spent fuel must be stored for 1 year after removal from the reactor to allow decay of short-lived isotopes). Therefore, after the reactor has been shut down for 2 years, these facilities would begin D&D activities that would last 5 years.

The assumed schedule of D&D activities is shown in Figure O.13 for Hanford, in Figure O.14 for INEL, and in Figure O.15 for SRS. The decontamination activities (active removal of radiological contamination) at the three sites would occur in two distinct time frames: from 1 year before reactor shutdown to 5 years after reactor shutdown, and from 30 to 36 years after reactor shutdown.

TABLE O.21 D&D Impacts of the HWR at SRS

Facility/Mode	Estimation Factor ^a	Approximate Volume (m ³)	Cost (millions of 1989 dollars)	Occupational Dose (person-rem)	Waste Volume (m ³)		
					HLW + TRU	LLW	Total
Tritium Production							
Building 320-M	Uncontaminated	61,000	2.8	0	0	0	0
Building 321-M	Fuel fabrication	93,000	4.3	8.1	NA ^b	NA	16,000
Reactor Facility	Reactor	140,000	43	130	NA	NA	7,200
Building 233-H	Fuel processing	8,100	9.7	2.8	240	160	400
Total (tritium production only)		300,000	60	160	240	160	24,000
Tritium Production and Spent Fuel Processing							
Building 221-H	Fuel processing	190,000	230	650	5,700	3,800	9,300
Total (tritium production plus spent fuel processing)		490,000	290	810	5,900	4,000	33,000
Tritium and Plutonium Production and Spent Fuel Processing							
Building 313-M	Fuel fabrication	200,000	9.2	17	NA	NA	34,000
Building 221-F	Fuel processing	220,000	260	750	NA	NA	11,000
Total (tritium and plutonium production plus spent fuel processing)		910,000	560	1,600	5,900	4,000	78,000

^aSee Table O.2.

^bNA = not available.

TABLE O.22 D&D Impacts of the LWR at SRS

Facility/Mode	Estimation Factor ^a	Approximate Volume (m ³)	Cost (millions of 1989 dollars)	Occupational Dose (person-rem)	Waste Volume (m ³)		
					HLW + TRU	LLW	Total
Tritium Production							
Tritium-Target Fabrication Facility	Uncontaminated	4,600	0.21	0	0	0	0
Fuel Fabrication Facility	Fuel fabrication	56,000	2.6	4.9	NA ^b	NA	9,500
Reactor Facility	Reactor	480,000	140	430	NA	NA	24,000
Building 233-M	Fuel processing	8,100	9.7	28	240	160	400
Total (tritium production only)		550,000	160	460	240	160	34,000
Tritium Production and Spent Fuel Processing							
Headend Fuel Processing Facility	Fuel processing	61,000	73	210	1,800	1,200	3,000
Building 221-H	Fuel processing	190,000	230	650	5,700	3,800	9,300
Uranium Recovery Building	Fuel processing	5,900	7.1	20	180	120	290
Total (tritium production plus spent fuel processing)		810,000	460	1,300	8,000	5,300	47,000
Tritium and Plutonium Production and Spent Fuel Processing							
Headend Fuel Processing Facility	Fuel processing	61,000	73	210	1,800	1,200	3,000
Building 221-F	Fuel processing	210,000	250	710	6,300	4,200	10,000
Uranium Recovery Building	Fuel processing	5,900	7.1	20	180	120	290
Total (tritium and plutonium production plus spent fuel processing)		1,100,000	800	2,300	16,000	11,000	61,000

^aSee Table O.2.^bNA = not available.

TABLE O.23 D&D Impacts of the MHTGR at SRS

Facility/Mode	Estimation Factor ^a	Approximate Volume (m ³)	Cost (millions of 1989 dollars)	Occupational Dose (person-rem)	Waste Volume (m ³)		
					HLW + TRU	LLW	Total
Tritium Production							
Tritium-Target Fabrication Facility	Uncontaminated	13,000	0.61	0	0	0	0
Fuel Fabrication Facility	Fuel fabrication	29,000	1.3	2.5	NA ^b	NA	4,900
Reactor Facility	Reactor	520,000	160	470	NA	NA	26,000
Building 233-M	Fuel processing	8,100	9.7	28	240	160	400
Total (tritium production only)		570,000	170	500	240	160	31,000
Tritium Production and Spent Fuel Processing							
Headend Fuel Processing Facility	Fuel processing	47,000	56	160	1,400	940	2,300
Building 221-H	Fuel processing	190,000	230	650	5,700	3,800	9,300
Total (tritium production plus spent fuel processing)		810,000	450	1,300	7,300	4,900	43,000
Tritium and Plutonium Production and Spent Fuel Processing							
Plutonium-Target Fabrication Facility	Fuel fabrication	31,000	1.4	2.7	NA	NA	5,300
Headend Fuel Processing Facility	Fuel processing	47,000	57	160	1,400	950	2,300
Building 221-F	Fuel processing	210,000	250	710	6,300	4,200	10,000
Total (tritium and plutonium production plus spent fuel processing)		1,100,000	760	2,200	15,000	10,000	61,000

^aSee Table O.2.

^bNA = not available.

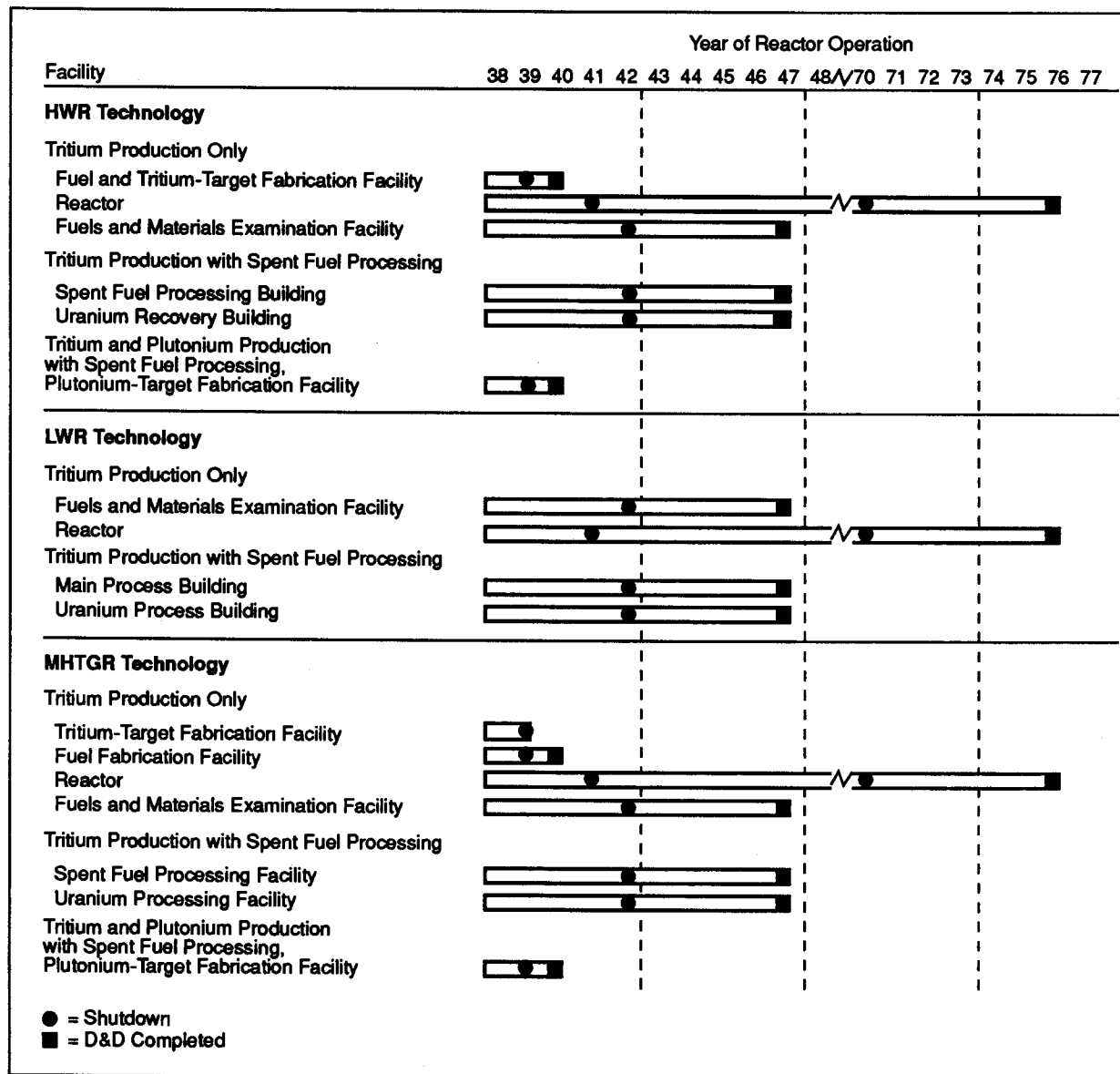


FIGURE O.13 Schedule of D&D at Hanford for the HWR, LWR, and MHTGR Technologies

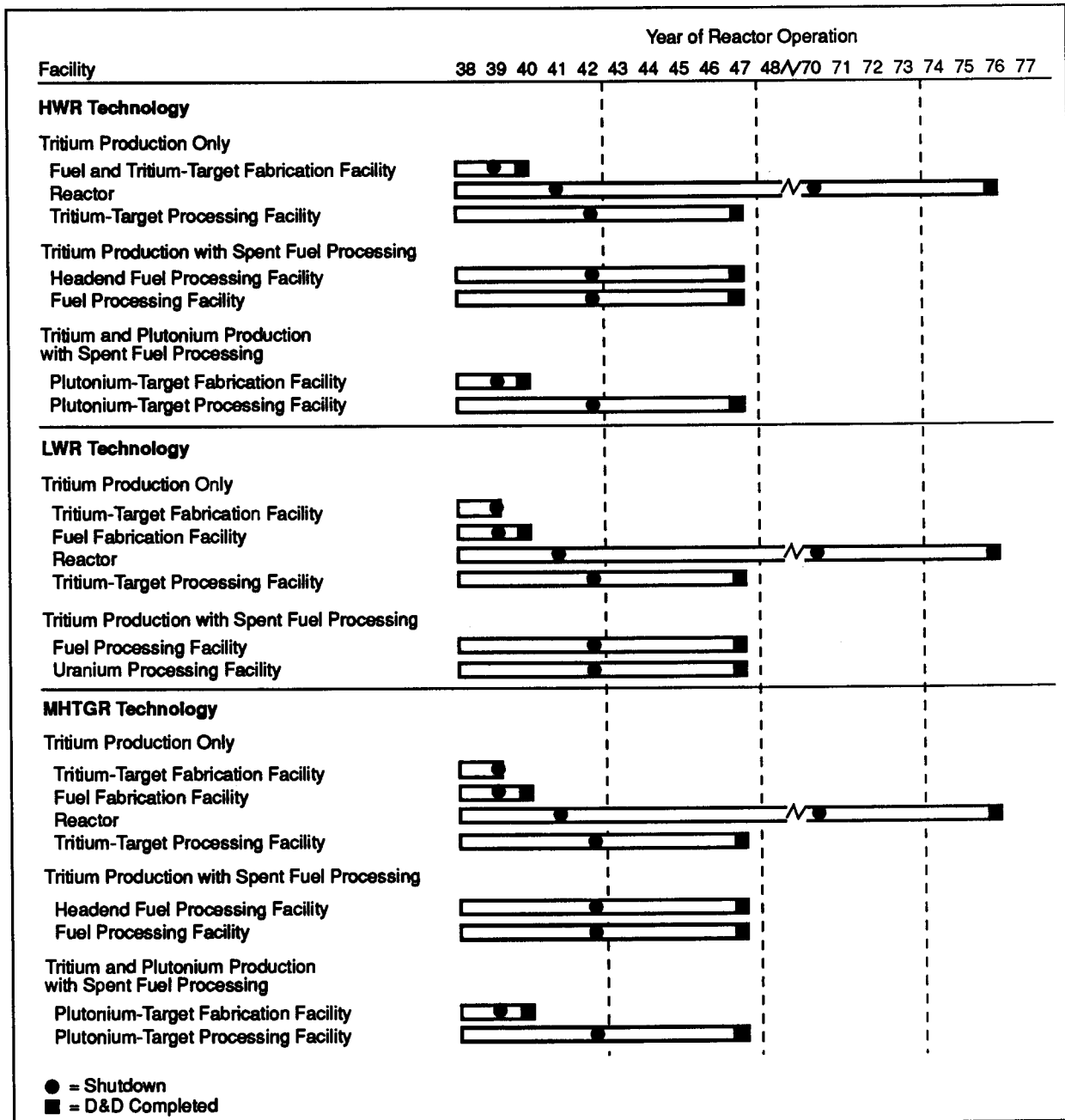


FIGURE O.14 Schedule of D&D at INEL for the HWR, LWR, and MHTGR Technologies

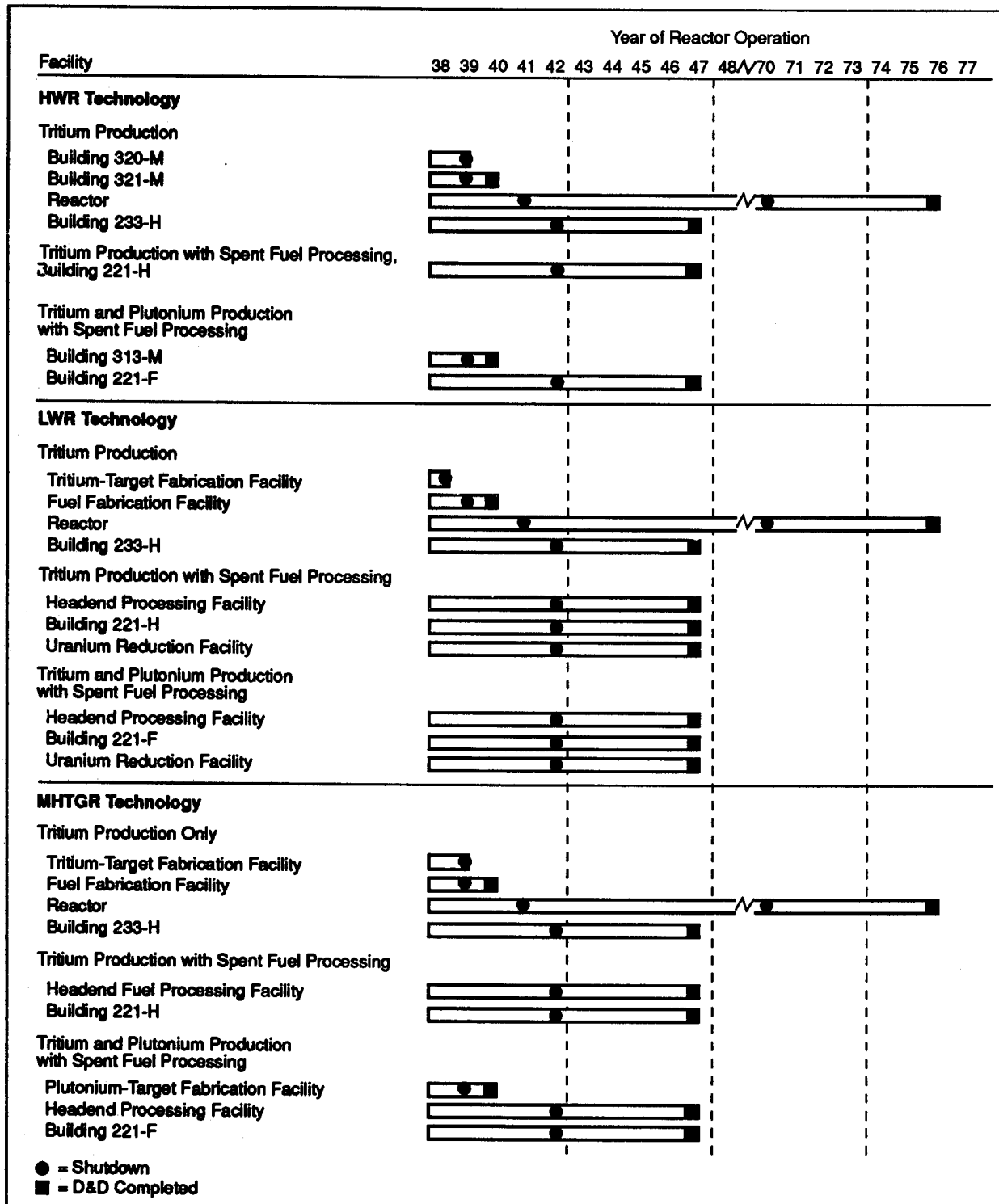


FIGURE O.15 Schedule of D&D at SRS for the HWR, LWR, and MHTGR Technologies

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*Copies of correspondence and other unpublished material cited in this list are available in U.S. Department of Energy Reading Rooms.

APPENDIX P:
WASTE STREAM COMPOSITION DATA

1. The first part of the document discusses the importance of maintaining accurate records of all transactions and activities. It emphasizes that proper record-keeping is essential for transparency and accountability, particularly in financial matters. The text suggests that organizations should implement robust systems to track and document every aspect of their operations.

2. The second part of the document addresses the challenges associated with data management and security. It highlights the need for organizations to protect sensitive information from unauthorized access and ensure the integrity of their data. The text recommends the use of secure storage solutions and regular security audits to mitigate risks.

3. The third part of the document focuses on the importance of communication and collaboration within an organization. It stresses that effective communication is key to achieving organizational goals and resolving conflicts. The text encourages the use of various communication channels, including meetings, reports, and digital tools, to facilitate information exchange.

4. The fourth part of the document discusses the role of leadership in driving organizational success. It emphasizes that leaders should provide clear vision, direction, and support to their teams. The text suggests that leaders should also foster a culture of innovation and continuous improvement, encouraging employees to take initiative and seek out new opportunities.

5. The fifth part of the document concludes by summarizing the key points discussed and reiterating the importance of the principles outlined. It states that by adhering to these principles, organizations can enhance their operational efficiency, maintain high standards of integrity, and achieve long-term success.

APPENDIX P:
WASTE STREAM COMPOSITION DATA

Appendix P presents pertinent site-specific information on liquid waste stream compositions for the light-water reactor (LWR), heavy-water reactor (HWR), and modular high-temperature gas-cooled reactor (MHTGR) at the Hanford Site, Idaho National Engineering Laboratory (INEL), and Savannah River Site (SRS). Tables P.1 through P.3 present information for the LWR and MHTGR at Hanford; Tables P.4 and P.5 present information for all three reactor technologies at INEL and Tables P.6 and P.7 present information for the HWR and MHTGR at SRS. The data in these tables are used in water resources and biotic resources assessments.

TABLE P.1 Chemical Composition of Cooling Tower Blowdown for an LWR at Hanford

Constituent	Concentrations in Blowdown (mg/L)		Mass Discharged (kg/d)				Drinking Water Standard (mg/L)
			Power Production Mode		Steam Condensing Mode		
			Average at 0.18 m ³ /s	Maximum ^a at 0.062 m ³ /s	Average at 0.29 m ³ /s	Maximum ^a at 0.11 m ³ /s	
Ammonia, as N	0.05	0.28	0.79	1.5	1.3	2.7	NS ^b
Calcium ^c	92.5	204	1,460	1,090	2,350	1,960	NS
Chloride	10.5	33.4	166	179	267	321	250 ^d
Chlorine, total residual	0	0.1	0.00	0.54	0.0	0.96	NS
Fluoride ^e	0.85	2.9	13.4	15.5	21.6	27.9	4.0 ^e
Hardness, ^c as CaCO ₃	343	800	5,420	4,280	8,720	7,690	NS
Magnesium ^c	20	49	316	262	508	471	NS
Nitrate, ^c as N	0.64	2.9	10.2	15.5	16.4	27.9	10 ^e
Nitrogen, ^c total organic	2.5	5	39.5	26.8	63.6	48.2	NS
Oil and grease	7.55	20	119	107	190	192	NS
pH units	7.9	8.5	—	—	—	—	6.5-8.5 ^d
Phosphorus	0.138	0.44	2.2	2.4	3.5	4.2	NS
Potassium ^c	3.85	9.1	60.9	48.6	97.7	87.3	NS
Silica, ^c as SiO ₂	22.3	62	353	331	567	596	NS
Sodium	10	24	158	128	254	230	NS
Solids, total dissolved	849	2,040	13,400	10,900	21,600	19,600	500 ^d
Solids, total suspended	20	100	316	535	508	961	NS
Sulfate	334	781	5,280	4,170	8,490	7,500	250 ^d

^aMaximum concentration of materials in cooling tower blowdown occurs at minimum flow (10 cycles of concentration).^bNS = no standard established.^cThese materials are concentrated but not added by operation of the plant.^dNational secondary drinking water regulations (40 CFR 143, July 1989).^eNational primary drinking water regulations (40 CFR 141, July 1989).

Source: WHC 1990.

TABLE P.2 Metal Concentration in Cooling Tower Blowdown for an LWR at Hanford

Constituent	Concentrations in Blowdown (mg/L)		Mass Discharged (kg/d)				Drinking Water Standard (mg/L)
	Average	Maximum	Power Production Mode		Steam Condensing Mode		
			Average at 0.18 m ³ /s	Maximum ^a at 0.062 m ³ /s	Average at 0.29 m ³ /s	Maximum ^a at 0.11 m ³ /s	
Barium ^b	0.5	1.0	7.9	5.3	12.7	9.6	1.0 ^c
Boron ^b	0.05	0.1	0.79	0.54	1.3	0.96	NS ^d
Cadmium ^b	0.0027	0.084	0.041	0.45	0.068	0.81	0.01 ^c
Chromium	0.0084	0.036	0.13	0.19	0.21	0.35	0.05 ^c
Colbalt ^b	0.0075	0.11	0.12	0.59	0.19	1.1	NS
Copper	0.22	0.61	3.4	3.2	5.5	5.8	1.0 ^e
Iron	0.31	1.5	4.9	7.8	7.8	14.0	0.3 ^e
Lead ^b	0.009	0.24	0.14	1.3	0.23	2.3	0.05 ^c
Manganese ^b	0.050	0.15	0.78	0.80	1.3	1.4	0.05 ^e
Mercury ^b	0.0026	0.041	0.041	0.22	0.068	0.40	0.002 ^c
Nickel	0.045	0.18	0.70	0.96	1.1	1.7	NS
Zinc	0.095	0.47	1.5	2.5	2.4	4.5	5.0 ^e

^aMaximum concentration of materials in cooling tower blowdown occurs at minimum flow (10 cycles of concentration).

^bThese materials are concentrated but not added by operation of the plant.

^cNational primary drinking water regulations (40 CFR 141, July 1989).

^dNS = no standard established.

^eNational secondary drinking water regulations (40 CFR 143, July 1989).

Source: WHC 1990.

TABLE P.3 Chemical Composition of Cooling Tower Blowdown for an MHTGR at Hanford^a

Constituent	Concentration ^b in Blowdown (mg/L)		Mass Discharged (kg/d) ^c		Drinking Water Standard (mg/L)
	Average	Maximum	At Average Concentration	At Maximum Concentration	
Ammonia, as N	0.05	0.28	0.9	2.3	NS ^d
Calcium ^e	92.5	204	1,740	1,640	NS
Chloride	10.5	33.4	197	269	250 ^f
Chlorine, total residual	0	0.1	0	1	NS
Fluoride ^e	0.85	2.9	16.0	23.4	4.0 ^g
Hardness, ^e as CaCO ₃	343	800	6,440	6,440	NS
Magnesium ^e	20	49	376	395	NS
Nitrate, ^e as N	0.645	2.9	12	23	10 ^g
Nitrogen, ^e total organic	2.5	5	47	40	NS
Oil and grease	7.5	20	141	161	NS
pH (pH units)	7.9	8.5	—	—	6.5-8.5 ^f
Phosphorus	0.138	0.440	2.6	3.5	NS
Potassium ^e	3.85	9.1	72	73	NS
Silica, ^e as SiO ₂	22.3	62	419	499	NS
Sodium	10	24	188	193	NS
Sulfate	376	888	7,050	7,150	250 ^f

^aData based on eight reactor modules operating at 100% capacity in the steam condensing mode.

^bThe normal average concentration is five cycles. Maximum concentration of materials in cooling tower blowdown occurs at minimum blowdown flow (10 cycles of concentration).

^cAverage mass discharge values are based on an average blowdown of 0.22 m³/s at average concentration. At maximum concentration, the blowdown is 0.093 m³/s.

^dNS = no standard established.

^eThese materials are concentrated but not added by operation of the plant.

^fNational secondary drinking water regulations (40 CFR 143, July 1989)

^gNational primary drinking water regulations (40 CFR 141, July 1989)

Source: WHC 1989.

**TABLE P.4 Estimated Values of Cooling Tower
Blowdown Parameters at INEL**

Parameters	Estimated Values
Temperature ^a (°C)	
HWR	32.2
LWR	29.4
MHTGR	32.2
Tritium ^b (pCi/L)	
HWR	25,000
LWR	74,000
MHTGR	14,000
Total Dissolved Solids ^c (mg/L)	1,410
Sulfate ^c (mg/L as SO ₄)	175
Hardness ^c (mg/L as CaCO ₃)	885
Silica ^c (mg/L as SiO ₂)	140
Iron ^c (mg/L)	0.29
Nitrogen (NO ₂ + NO ₃) ^c (mg/L as N)	6.5

^aData from United Engineers and Constructors 1990; temperature of cooling tower outlet does not account for cooling in evaporation pond.

^bCalculated from United Engineers and Constructors 1990; and Arnett, Martineau, and Lehto 1990. Estimated values are based on annual values of 210 Ci in 8,400,000 m³ of blowdown for an HWR, 640 Ci in 8,700,000 m³ of blowdown for an LWR, and 95 Ci in 6,700,000 m³ of blowdown for an MHTGR. Detritiation was assumed to reduce tritium releases by 90% for all three reactor alternatives.

^cFive times the mean concentrations in Table F.13.

TABLE P.5 Estimated Annual Liquid Radiological Releases from the Liquid Radwaste Treatment System for an MHTGR at INEL

Radionuclide	Half-Life	Radionuclide Release	
		Quantity (Ci/yr)	Concentration ^a (pCi/L)
Sr-90	28.9 yr	1.43×10^{-5}	38
Y-90	64.0 h	1.43×10^{-5}	38
Ag-110m	253.0 d	7.51×10^{-7}	2.0
I-131	8.06 d	2.66×10^{-7}	0.71
Cs-137	2.06 yr	3.39×10^{-7}	91
Cs-134	30.2 yr	1.48×10^{-4}	400
Ba-137m	2.55 min	1.40×10^{-4}	370
Others	--	2.25×10^{-5}	60

^aBased on an average liquid discharge of 370 m³/yr.

Source: EG&G 1989.

TABLE P.6 Chemical Composition of Savannah River Water and Assumed Composition of Cooling Tower Blowdown for an HWR at SRS

Constituent	Ambient Savannah River Water Quality (mg/L)		2.5 Cycle Blowdown ^a 0.78 m ³ /s (mg/L)		2.5 Cycle Blowdown 0.78 m ³ /s (kg/d)		2.5 Cycle Raw Water Thermal Mixing 1.18 m ³ /s (kg/d)	
	Avg. ^b	Max.	Avg.	Max.	Avg.	Max.	Avg. ^c	Max.
Alkalinity	19.7	29	49	72	3,300	4,860	10,000	14,800
Chloride	7.22	14	18	35	1,030	2,350	3,680	7,140
Chlorine, residual ^d	-	-	0.011	-	0.19	-	-	-
Hardness	7.55	17	19	42	1,270	2,805	3,850	8,660
Suspended solids	9.81	40	24	100	1,640	6,700	5,000	20,400
Dissolved solids	62.8	86	160	220	10,500	14,400	32,000	43,800
Total solids	72.6	100	180	250	12,200	16,800	37,000	50,900
NO ₂ + NO ₃	0.44	1.7	1.1	4.2	73.6	285	224	866
Sulfate ^e	6.35	10	77	86	5,180	5,800	3,240	5,100
Total phosphate	0.09	0.19	0.22	0.48	15	31.8	65.9	96.8
Aluminum	0.12	0.24	0.3	0.6	20	40.5	61.4	122
Ammonia (NH ₃)	0.13	0.24	0.32	0.6	22	40.5	66.4	122
Calcium	4.14	5.2	10	13	694	872	2,110	2,650
Copper	0.02	0.05	0.05	0.12	3.2	8.2	10	25.5
Cadmium	0.01	0.01	0.025	0.025	1.8	1.8	5.0	5.0
Magnesium	1.27	1.5	3.2	3.8	213	251	647	765
Manganese	0.07	0.19	0.18	0.48	11.8	31.8	35.5	96.8
Mercury	0.00018	0.0002	0.00045	0.0005	0.032	0.032	0.091	0.10
Nickel ^f	0.02	0.05	-	0.81	-	54.1	10	25.5
Sodium	9.5	14	24	35	1,590	2,350	4,840	7,140

TABLE P.6 (Cont'd)

Constituent	Ambient Savannah River Water Quality (mg/L)		2.5 Cycle Blowdown ^a 0.78 m ³ /s (mg/L)		2.5 Cycle Blowdown 0.78 m ³ /s (kg/d)		2.5 Cycle Raw Water Thermal Mixing 1.18 m ³ /s (kg/d)	
	Avg. ^b	Max.	Avg.	Max.	Avg.	Max.	Avg. ^c	Max.
Total iron ^f	0.31	0.87	-	6.5	-	436	158	443
Lead	0.01	0.05	0.025	0.12	1.8	8.2	5.0	25.5
Chromium ^f	0.02	0.05	-	1.7	-	115	10.0	25.5
Zinc	0.03	0.1	0.075	0.25	5.0	16.8	15.5	50.9

^aBlowdown concentrations are assumed to be 2.5 times river water concentrations.^bUpstream Savannah River concentrations.^cRaw river water added to blowdown discharge canal at a rate of 1.18 m³/s to meet site limit of 2.8°C maximum temperature rise above ambient stream temperature.^dResidual chlorine levels set to 0.011 mg/L for 2 hours per day at Savannah River.^eSulfate addition of 2.9 kg/min assumed.^fIron, chromium, and nickel are assumed to be added to the system due to erosion of stainless steel heat exchangers. Based on an assumed concentration of chromium at the site discharge limit of 1.71 mg/L, with iron and nickel based on weight percent in 304L stainless steel (19% chromium, 9% nickel, <0.03% carbon, balance iron).

Source: WSRC 1991a.

TABLE P.7 Chemical Composition of Savannah River Water and Assumed Composition of Cooling Tower Blowdown Effluents for an MHTGR at SRS

Constituent	Ambient Savannah River Water Quality (mg/L)		2.5 Cycle Blowdown ^b 0.63 m ³ /s (mg/L)		2.5 Cycle Blowdown 0.63 m ³ /s (kg/d)		2.5 Cycle Raw Water Thermal Mixing 0.96 m ³ /s (kg/d)	
	Avg. ^a	Max.	Avg.	Max.	Avg. ^c	Max.	Avg. ^d	Max.
Alkalinity	19.7	29	49	72	2,680	3,940	8,130	12,000
Chloride	7.22	14	18.0	35	981	1,900	2,980	5,790
Chlorine, residual ^e	-	-	0.011	-	0.15	-	-	-
Hardness	7.55	17	19	42	1,030	2,310	3,120	7,030
Suspended solids	9.81	40	24	100	1,330	5,440	4,050	16,500
Dissolved solids	62.8	86	160	220	8,530	11,700	25,900	35,500
Total solids	72.6	100	180	250	9,870	13,600	30,000	41,300
NO ₂ + NO ₃	0.44	1.7	1.1	4.2	60	231	182	703
Sulfate ^f	6.35	10	72	81	3,890	4,380	2,620	4,130
Total phosphate	0.09	0.19	0.22	0.48	12.3	25.9	37.3	78.6
Aluminum	0.12	0.24	0.3	0.6	16.4	32.7	49.5	99.1
Ammonia (NH ₃)	0.13	0.24	0.32	0.6	17.7	32.7	53.6	99.1
Calcium	4.14	5.2	10	13	563	707	1,710	2,150
Copper	0.02	0.05	0.09	0.17	5.0	9.09	8.18	20.4
Cadmium	0.01	0.01	0.025	0.025	1.4	1.36	4.09	4.09
Magnesium	1.27	1.5	3.2	3.75	173	204	525	620
Manganese	0.07	0.19	0.18	0.48	9.55	25.9	29.1	78.6
Mercury	0.00018	0.0002	0.00045	0.0005	0.023	0.027	0.073	0.082
Nickel ^g	0.02	0.05	0.06	0.13	3.18	7.27	8.18	20.5
Sodium	9.5	14	24	35	1,290	1,900	3,930	5,790

TABLE P.7 (Cont'd)

Constituent	Ambient Savannah River Water Quality (mg/L)		2.5 Cycle Blowdown ^b 0.63 m ³ /s (mg/L)		2.5 Cycle Blowdown 0.63 m ³ /s (kg/d)		2.5 Cycle Raw Water Thermal Mixing 0.96 m ³ /s (kg/d)	
	Avg. ^a	Max.	Avg.	Max.	Avg. ^c	Max.	Avg. ^d	Max.
Total iron ^e	0.31	0.87	0.78	6.5	423	354	128	360
Lead	0.01	0.05	0.025	0.12	1.36	6.82	4.09	20.5
Chromium ^g	0.02	0.05	0.05	0.13	2.73	7.27	8.18	20.5
Zinc	0.03	0.1	0.075	0.25	4.09	13.6	12.3	41.4

^aUpstream Savannah River concentrations.^bBlowdown concentrations are assumed to be 2.5 times river water concentrations.^cDaily releases calculated for 0.63 m³/s blowdown rate, 24 hours per day.^dRaw river water added to blowdown discharge canal at a rate of 0.96 m³/s to meet site limit of 2.8°C maximum temperature rise above ambient stream temperature.^eResidual chlorine levels set to 0.011 mg/L for 2 hours per day at Savannah River.^fSulfate addition of 2.9 kg/min assumed.^gChromium, nickel, and iron are assumed to be added to system due to erosion of condenser coils at a rate comparable to WNP-1 at 41, 270, and 2,100 mg/min, respectively.

Source: WSRC 1991b.

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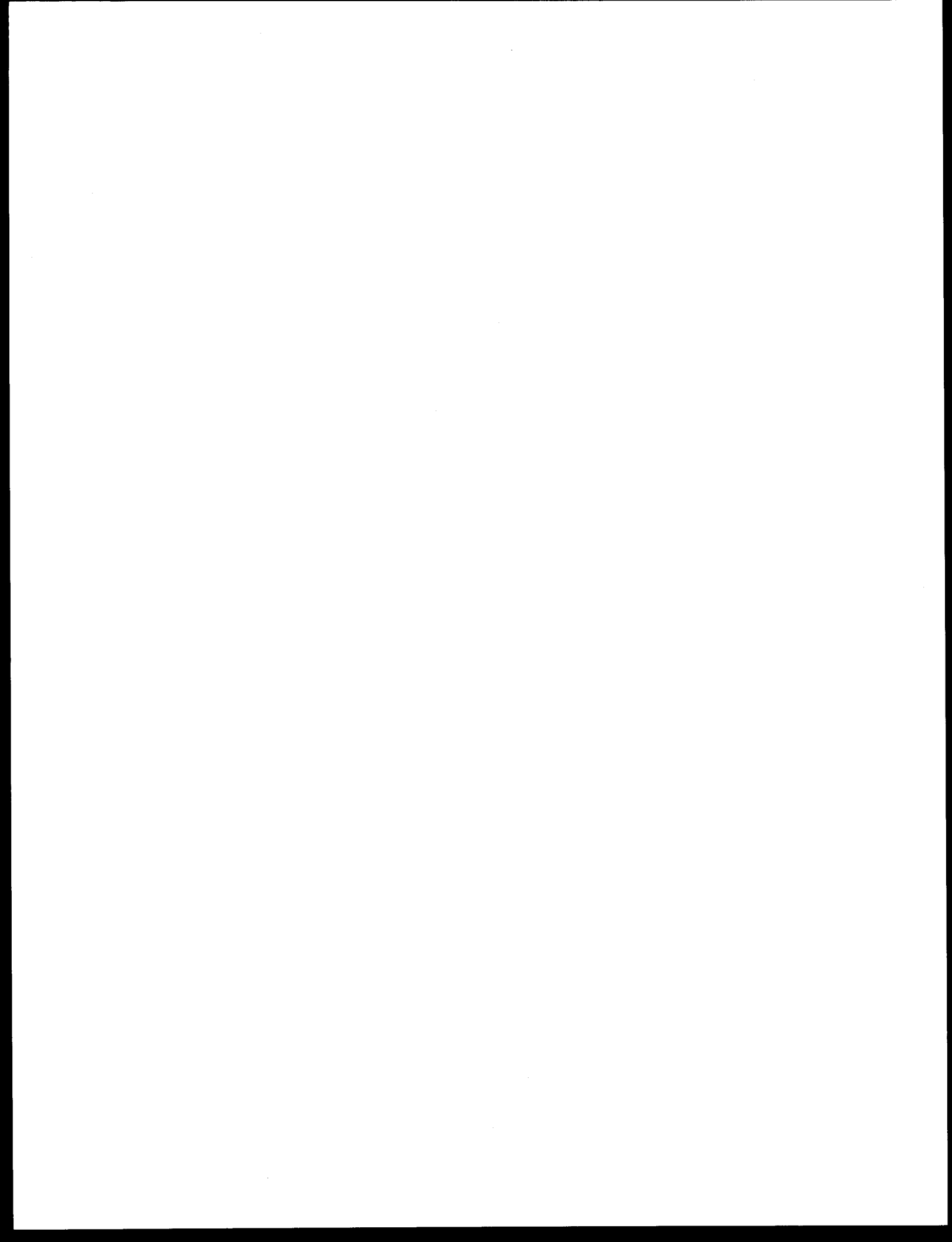
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*Copies of correspondence and other unpublished material cited in this list are available in U.S. Department of Energy Reading Rooms.



APPENDIX Q:
CULTURAL RESOURCES



APPENDIX Q:
CULTURAL RESOURCES

This appendix contains additional information (in tabular form) regarding archaeological surveys in the affected areas at Hanford (Table Q.1) and archaeological sites and historic structures in the affected areas at Hanford (Table Q.2), INEL (Table Q.3), and SRS (Table Q.4).

TABLE Q.1 Results of Archaeological Surveys Conducted in the Potentially Affected Area of the Hanford Site

Project Name	Year of Survey	Area Surveyed (km ²)	Transect Interval ^a	Cultural Resources Observed	Location of Collection	References
Skagit/Hanford Power	1981	2.20	50 ft	None	None	ERTEC 1982
Hanford Recon.	1968	6.73	NS ^b	None	NS	Rice 1968
WPPSS-WNP-2	1972		NS	None	None	Rice 1972
WNP-1 relocated	1974	9.48	NS	None	NS	Rice 1974
Warm water irrigation	1978		100 m ^c	None	None	Rice 1978, 1983
Support facilities	1980		5 m ^c	None	None	Rice 1980
Interior dunes	1981	2.5	65 yd ^d	One prehistoric site (45BN266)	HCRL	Rice 1981
Potential reactor	1983		65 yd	None	None	Rice 1984
FFTF	1978	0.12	Unknown	None	None	Rice, Stratton, and Lindeman, 1978
Ashe Substation	1976	0.18	25-50 m	Late period projectile point and ungulate tooth (45BN229)	Collected, Central Washington State College	Smith et al. 1977
Skagit/Hanford	1981	2.20	50 ft	None	None	ERTEC 1982
HCRC# 87-400-001	1987	0.01	20 m	None	None	Cadore and Chatters 1988
HCRC# 87-400-002	1987	0.02	20 m	Basalt-notched projectile point (200-1200 AD)	Collected, HCRL (HI-87-018)	Chatters and Hoover 1988
HCRC# 87-400-007	1988	0.12	20 m	None	None	Jackson and Chatters 1988
HCRC# 88-600-002	1988	0.02	20 m	None	None	Jackson and Chatters 1988

TABLE Q.1 (Cont'd)

Project Name	Year of Survey	Area Surveyed (km ²)	Transect Interval ^a	Cultural Resources Observed	Location of Collection	References
BWIP borehole pads DC-7 and DC-8	1988	<0.01	20 m	None	None	HCRL 1988

^aAll were pedestrian surveys; transect intervals are given in units reported by original investigators.

^bNS = not specified.

^c"Judgment sample."

^dShovel test down to 18 in. every 100 ft.

Conversion factors: 1 km² = 0.3861 mi²; 1 m = 3.281 ft.

TABLE Q.2 Archaeological Resources at the NPR Site on Hanford

Site	Area (m ²)	Period	Potential NRHP Eligibility
Hanford Townsite	a	Early twentieth century	Yes
45BN298H	a	Late nineteenth century	No
45BN305	a	Prehistoric	Yes
45BN305H	a	Historic	No
Building 271-B	a	Twentieth century	Yes
Building 224-B	a	Twentieth century	Yes
Building 272-E	a	Twentieth century	Yes
Building 272-EA	a	Twentieth century	Yes
Building 248-E	a	Twentieth century	Yes

^aData to be provided.

Source: Chatters 1989.

TABLE Q.3 Archaeological Resources at the NPR Site
on INEL

Site	Area (m ²)	Period	Potential NRHP Eligibility
10-BT-371	1,178	Prehistoric lithic	Yes ^a
10-BT-372	100	Prehistoric lithic	Yes ^a
10-BT-373	13,352	Prehistoric lithic	Yes
10-BT-375	3,600	Prehistoric lithic	Yes ^a
10-BT-377	3,000	Prehistoric lithic	Yes ^a
10-BT-378	1,414	Prehistoric lithic	Yes ^a
10-BT-383	1,408	Prehistoric lithic	Yes ^a
10-BT-385	188	Prehistoric lithic	Yes ^a
10-BT-388	4.9	Prehistoric feature	Yes ^a
10-BT-390	180	Prehistoric lithic; historic	Yes ^a
10-BT-395	144	Prehistoric lithic	Yes ^a
10-BT-396	628	Prehistoric lithic	Yes ^a
10-BT-397	600	Prehistoric lithic	Yes ^a
10-BT-398	942	Prehistoric lithic	Yes ^a
10-BT-400	754	Prehistoric lithic	Yes ^a
10-BT-401	400	Prehistoric lithic	Yes ^a
10-BT-404	3,600	Prehistoric lithic	No
10-BT-405	25	Prehistoric lithic	No
10-BT-406	1,885	Prehistoric lithic	No
10-BT-407	5,891	Prehistoric lithic	No
10-BT-413	400	Prehistoric lithic	No
10-BT-415/447	400	Prehistoric lithic	No
10-BT-417	1,178	Prehistoric lithic	No
10-BT-420	200	Prehistoric lithic	Yes ^a
10-BT-422	10	Prehistoric lithic	Yes ^a
10-BT-423	2,000	Prehistoric lithic	Yes ^a
10-BT-424	225	Prehistoric lithic	Yes ^a
10-BT-425	300	Prehistoric lithic	Yes
10-BT-426	4,712	Prehistoric lithic	No
10-BT-427	3,142	Prehistoric lithic	Yes ^a
10-BT-428	2,262	Prehistoric lithic	Yes
10-BT-429	550	Prehistoric lithic	Yes ^a
10-BT-430	39	Prehistoric lithic	Yes ^a
10-BT-436	11,781	Prehistoric lithic	Yes
10-BT-437	942	Prehistoric lithic	Yes ^a

TABLE Q.3 (Cont'd)

Site	Area (m ²)	Inventory	Potential NRHP Eligibility
10-BT-438	3.1	Prehistoric feature	Yes
10-BT-439	7,540	Prehistoric	Yes ^a
10-BT-440	500	Prehistoric lithic	No
10-BT-441	100	Prehistoric lithic	No
10-BT-443	3.1	Prehistoric feature	Yes
10-BT-444	10	Prehistoric feature	No
10-BT-445	1,131	Prehistoric lithic	No
10-BT-446	100	Prehistoric lithic	No
10-BT-448	17,250	Prehistoric lithic	Yes
10-BT-450	4,500	Prehistoric lithic	No
10-BT-452	353	Prehistoric lithic	No
10-BT-453	1.8	Prehistoric feature	Yes
10-BT-1486	9,900	Prehistoric lithic	Yes ^a
10-BT-1487	80	Prehistoric lithic	Yes ^a
10-BT-1488	353	Prehistoric lithic	Yes ^a
10-BT-1489	5,950	Prehistoric lithic	Yes ^a
10-BT-1490	11,781	Prehistoric lithic	Yes ^a
10-BT-1494	5,498	Prehistoric lithic	Yes ^a
10-BT-1497	1,950	Prehistoric lithic	Yes ^a
10-BT-1498	840	Prehistoric lithic	No
10-BT-1499	330	Prehistoric lithic	No
10-BT-1500	182	Prehistoric lithic	Yes ^a
10-BT-1502	1,280	Prehistoric lithic	Yes ^a
10-BT-1503	4,087	Prehistoric lithic	Yes ^a
10-BT-1505	810	Prehistoric lithic	Yes ^a
10-BT-1509	188	Prehistoric lithic	No
10-BT-1513	2,029	Prehistoric lithic	Yes ^a
10-BT-1515	1,050	Prehistoric lithic	Yes ^a
10-BT-1585	9,425	Prehistoric lithic	Yes ^a
10-BT-1593	100	Prehistoric lithic	Yes ^a

^aEligibility to be determined on the basis of results from field investigations during 1990-1991.

Source: Data derived from Ringe 1990.

TABLE Q.4 Archaeological Resources at the NPR Site on SRS

Site	Area (m ²)	Period	NRHP Eligibility
38BR256	2,500	Late nineteenth century-twentieth century	No
38BR258	3,000	Middle-Late Archaic Middle-Late Woodland	No
38BR287	1	Early Woodland (isolated find)	No
38BR309	2,500	Prehistoric lithic	No
38BR522	200	Twentieth century	Yes
38BR579	2,800	Late nineteenth century-twentieth century Prehistoric lithic	No
38BR619	20,000	Late nineteenth century-twentieth century	Yes
38BR620	2,400	Late nineteenth century-twentieth century Prehistoric lithic	No
38BR622	1	Prehistoric lithic	No
38BR623	1	Prehistoric lithic	No
38BR624	25	Prehistoric lithic	No
38BR625	50	Prehistoric lithic	No
38BR626	10	Late nineteenth century-twentieth century	No
38BR627	25	Prehistoric lithic	No
38BR628	500	Prehistoric lithic	No
38BR629	3,000	Late nineteenth century-twentieth century	Yes

Source: Data derived from Brooks et al. 1989.

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APPENDIX R:

**SUPPLEMENTAL EVALUATION OF ENVIRONMENTAL IMPACTS
FOR A SIX-MODULE 450-MWt MHTGR PRODUCTION COMPLEX**

APPENDIX R:

SUPPLEMENTAL EVALUATION OF ENVIRONMENTAL IMPACTS FOR A SIX-MODULE 450-MWt MHTGR PRODUCTION COMPLEX

R.1 INTRODUCTION

As described in Section 3 and in Appendix C, the reference modular high-temperature gas-cooled reactor (MHTGR) production complex evaluated in this environmental impact statement (EIS) would consist of eight reactor modules, each rated at 350-MWt power. The conceptual design effort, completed in 1990, confirmed that a 350-MWt reactor module is viable, considering environmental, safety, and technological aspects. Preliminary engineering work, initiated in the latter half of 1990, disclosed that optimizing the plant design could potentially allow higher reactor module thermal ratings without affecting the general overall plant design or the inherent MHTGR safety characteristics.

Appendix R presents an assessment of the potential environmental impacts of constructing, operating, and decommissioning a six-module 450-MWt MHTGR complex in comparison with the reference design, i.e., an eight-module 350-MWt MHTGR complex. All issues evaluated in the EIS are considered. Where the two concepts show significant construction and operating differences, impacts of the changes are discussed.

In this appendix, the MHTGR complex includes the Reactor Facility, Energy Conversion Area (ECA), cooling towers, fuel and target fabrication and processing facilities, and other support facilities. The Reactor Facility includes the reactor buildings, operations centers, and support buildings containing radioactive waste systems. The reactor building includes the reactor modules and associated containments, steam bypass systems, refueling equipment, and fuel and target assembly and storage areas. The reactor module includes the core, steam generator and associated systems, and structures within a containment.

Table R.1 is a matrix that identifies the issues evaluated, summarizes the differences relative to the reference design, and provides the bases for the assessment. This matrix provides a guide to issues addressed in detail as well as issues that did not require further consideration. Issues were not subjected to further consideration if no important differences were identified between potential impacts (real or perceived societal or economic effects) associated with the operation of a six-module 450-MWt MHTGR complex compared with an eight-module 350-MWt MHTGR complex. Areas meriting further assessment and their relative potential impacts to the Idaho National Engineering Laboratory (INEL) are discussed in Sections R.2 through R.9. The relative impacts for Hanford and the Savannah River Site (SRS) are discussed in Section R.10.

TABLE R.1 Matrix of Issues for Environmental Evaluation of a 450-MWt MHTGR Compared with a 350-MWt MHTGR

Issue	Relative Difference ^a	Remarks	Reference Section
Site selection	None	Site selection is independent of the reactor design.	None
Facility descriptions			
Reactor Facility and ECA	Changes	Power per module is increased.	R.2.1
Support facilities	None	The minor increase in product storage area has no overall effect.	R.2.2
Affected environment	None	The environmental baseline is independent of the reactor design.	R.3
Construction resources			
Reactor Facility and ECA			
Work force	Minor decrease	The minor decrease in work force has no overall effect.	R.4.1.1
Materials	Minor decrease	The materials decrease is small and has no overall effect.	R.4.1.2
Land use	Minor decrease	The minor decrease in land use has no discernible effect.	R.4.1.3
Schedule	None	The overall schedule is essentially the same.	R.4.1.4
Design criteria	None	The design criteria are dependent on site characteristics, not on reactor design.	None
Transportation	Minor decrease	The minor decrease has no overall effect.	R.4.1.5
Support facilities	None	The minor increase in product storage has no overall effect.	R.4.2
Operational resources			
Reactor Facility and ECA			
Work force	Minor decrease	The minor decrease in work force has no significant overall impact.	R.5.1.1
Materials	None	Except for a decrease in uranium, there are no discernible changes.	R.5.1.2
Transportation	None	There are no discernible changes because the material balances are the same.	R.5.1.3
Support facilities	Minor decrease	The minor decrease in uranium has no significant overall effect.	R.5.2

TABLE R.1 (Cont'd)

Issue	Relative Difference ^a	Remarks	Reference Section
Consequences			
Air resources			
Air quality			
Construction	Minor decrease	The slight decrease in excavation has no overall effect.	R.6.1.1
Operations	Minor decrease	The minor decrease in vehicle emissions has no significant effect.	R.6.1.1
Cooling towers	Minor decrease	Slightly less power is dissipated.	R.6.1.2
Noise	Minor decrease	The minor decrease in construction traffic, construction blasting, and operations traffic has no overall effect.	R.6.1.3
Water resources			
Construction	Minor decrease	Slightly lower reactor construction requirements have no overall effect.	R.6.2
Operations	Minor decrease	Slightly less power is dissipated.	R.6.2
Land use and resources	Minor decrease	The minor decrease in land disturbance has no discernible effect.	None
Biota	Minor decrease	Minor decreases in land disturbance, releases, and exposures for construction and operation have no overall effect.	None
Cultural resources	None	There are no discernible changes.	None
Health and safety			
Reactor Facility and ECA			
Normal operations	Minor decrease	Slightly lower releases have no significant impact and are well within regulatory guidelines, i.e., DOE Order 5400.5 and 40 CFR 61.	R.6.3.1
Nonsevere accidents	Increase	Increases in releases are well within regulatory guidelines, i.e., DOE Order 5480.6 and 10 CFR 100.	R.6.3.1
Severe accidents	Increase	Increases in releases are well within NPR safety goals.	R.6.3.1
Nonradiological	None	The same types, severities of hazards, and quantities are involved.	R.6.3.1

TABLE R.1 (Cont'd)

Issue	Relative Difference ^a	Remarks	Reference Section
Support facilities	Minor decrease	The minor reduction in fuel fabrication and processing normal operating releases has no overall effect and is well within regulatory limits.	R.6.3.2
Socioeconomics	Minor decrease	The minor reduction in work force has no overall effect.	R.6.4
Transportation	Minor decrease	The minor decrease in quantities being transported has no overall effect.	R.6.5
Waste management Construction	Minor decrease	The minor decrease in construction wastes has no discernible effect.	R.6.6.1
Operations	Decrease	The 5% reduction in HLW significantly reduces costs. Reduction in other wastes has no overall effect.	R.6.6.2
Cumulative impacts	None	Impacts of the 450-MWt design are comparable to the 350-MWt design; therefore, there is no significant increase in cumulative impacts.	R.7
Decontamination and decommissioning	Minor decrease	There is a minor decrease in waste volumes, occupational doses, and costs.	R.8
Environmental regulations	None	There is no change in applicable regulations or compliance.	R.9

^aIncrease or decrease in impact for the six-module 450-MWt MHTGR complex relative to the eight-module 350-MWt MHTGR complex.

Source: Modified from EG&G 1991.

R.2 FACILITY DESCRIPTIONS

R.2.1 Reactor Facility and Energy Conversion Area

R.2.1.1 Introduction

The plant structures, reactor systems, and components for the 450-MWt reactor would be similar in concept and design to those for the 350-MWt reactor. Consequently, much of the descriptive information provided in Appendix C for the reference design 350-MWt MHTGR complex would also apply to the 450-MWt MHTGR complex. Information given in Appendix C is not repeated here; however, similarities are cited, and design differences are discussed.

R.2.1.2 Reactor Arrangement and Plant Structures

The layout and design of the Reactor Facility would be similar for the two reactor concepts. The primary difference is that each reactor building would house three reactor modules in the 450-MWt Reactor Facility instead of four. This change would impact the ECA, where the energy conversion equipment consists of six 200-MWe (for the 450-MWt design) versus eight 150-MWe (for the 350-MWt design) turbine generators.

Each reactor module would be housed in a vertical cylindrical concrete enclosure fully embedded in the earth. The reactor module arrangement would be similar for both the 350-MWt (Figure C.3) and the 450-MWt designs. The reactor and steam generator vessels would be vertically positioned and connected by a coaxial cross-vessel. The reactor core, reactor metallic internals, shutdown circulator, and shutdown heat exchanger would be located inside the reactor vessel. The main circulator would be inside and at the top of the steam generator vessel. The arrangement and operation of these systems would be the same for both units.

R.2.1.3 Reactor Core

A plan view of the 450-MWt reactor core is shown in Figure R.1. The core array for the 450-MWt reactor would incorporate an additional ring of inner reflector columns, increasing the number from 19 to 37. The active core would consist of three rings of fuel columns, so the width across the annular active core of the 450-MWt design would be the same as the 350-MWt design. The outer reflector design would also be similar, consisting of two rings of hexagonal reflector blocks and a ring of permanent side reflector blocks. Overall, the height of the 450-MWt design core array would be the same as the 350-MWt design, but the diameter of the 450-MWt design would be 6.77 m compared with 6.10 m for the 350-MW design.

The number of fuel columns would increase from 66 in the 350-MWt design to 84 in the 450-MWt design. Each design would have ten fuel blocks per fuel column plus top and bottom reflector blocks. The ratio of fuel columns in the 450- and 350-MWt cores (84 compared with 66) would be nearly equal to the ratio of their power ratings; therefore, the reactor cores would have essentially the same power density.

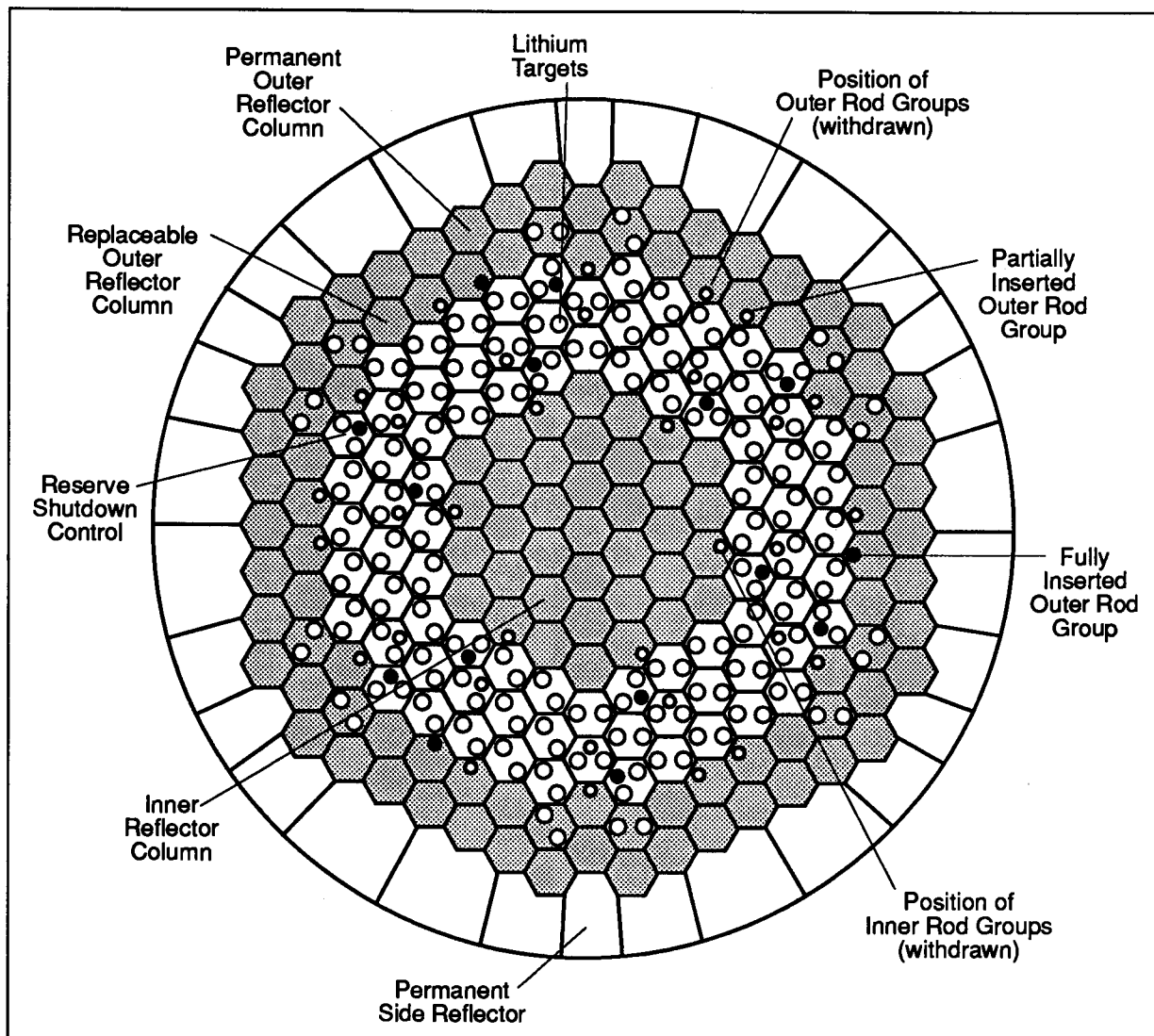


FIGURE R.1 Plan View of the 450-MWt Reactor Core (Source: Modified from EG&G 1991)

The 450-MWt reactivity control design would also be similar to the 350-MWt design, but it would incorporate control rods in the active core. The 450-MWt design would include 6 control rods in the inner reflector, 12 control rods in the active core, and 18 control rods in the outer reflector. The reserve shutdown control system would have six channels in the inner ring of fuel columns and six channels in the outer ring of fuel columns.

The 450-MWt design would be similar to the 350-MWt design because, for all normal operating conditions, the inner reflector and active core control rods would be fully withdrawn before achieving criticality. Consequently, the power level would be controlled with the outer reflector control rods only. Power operation on the outer reflector control rods (no control rods in the core) would provide the same assurance of a stable axial power profile for the 450-MWt core as for the 350-MWt core. The two designs also have similar reactivity insertions for rod withdrawal events. The control rod design for the 450-MWt reactor module, with the addition of in-core control rods, would considerably enhance the cold shutdown margin.

The fuel block designs in the 350- and 450-MWt cores would be identical, with the exception of fuel blocks in 12 columns of the 450-MWt core that contain control rod channels. The control rod channel in these fuel blocks would be slightly larger than the reserve shutdown channel. The fuel particle and fuel element designs would be identical as would the lithium-target particles, compacts, and assemblies.

The 350- and 450-MWt reactor modules would refuel on the same schedule. Each design has a nominal 3-year residence time for the fuel, and one-third of the fuel blocks would be replaced annually. The target assemblies in all the fuel and reflector blocks would be replaced semiannually for the tritium production mode and quarterly for the plutonium production mode.

Circulating activity inventories in the 450-MWt primary loop would be higher than in the 350-MWt primary loop by approximately the ratio of the power levels for the following reasons:

1. Fuel and target operating temperatures would be nearly equal (similar power distributions, core power density, and core coolant temperature rise). Core power distributions would be similar because the core configurations are similar, power operation is controlled by the outer reflector rods in both designs, similar loadings are used, and the power density and cycle lengths are similar.
2. Fuel and target exposures and burnups would be nearly equal (similar core power density and cycle length).
3. Given items 1 and 2, fuel and target particle performance and release to the primary coolant would be essentially equal on a per-megawatt basis.
4. The primary coolant purification flow rate would be proportional to the power level.

The 450-MWt MHTGR complex would maintain the flexibility to convert from a tritium production mode to either a plutonium production mode or a power production only mode. The same considerations and changes in the fuel cycle for these modes would apply to both the 350- and the 450-MWt MHTGR complexes.

R.2.1.4 Fuel and Target Handling

Equipment and procedures for refueling the 450-MWt reactor module would be similar to refueling equipment and procedures used in the 350-MWt reactor module. Refueling would be accomplished in one-sixth core sectors, and the spent fuel blocks would be replaced with a combination of new blocks and retargeted blocks from previous exchanges. For the 450-MWt reactor module, minor modifications would be required for the equipment support structures for refueling operations. These modifications would slightly impact the retargeting and refueling times and, thus, the capacity factor. The reactor module storage capacity for fuel blocks, reflector blocks, and targets would need to be increased.

R.2.1.5 Steam Generator

The higher power rating of the 450-MWt steam generator would be accommodated by increasing the number of steam generator tubes. Operating conditions on the secondary side of the steam generator would be the same for the 350- and 450-MWt designs (feedwater temperature/pressure = 193°C/210 kg/cm²; main steam temperature/pressure = 540.5°C/176.6 kg/cm²). Operating conditions on the primary side of the steam generator would be only slightly different for the 450-MWt design (pressure would increase from 65 to 72 kg/cm²; hot helium temperatures would increase from 685 to 695°C; and cold helium temperatures would increase from 259.4 to 274°C). The increase in the cold-leg temperature would be a design improvement for the steam generator, allowing a reduction in the overall height of the bundle by about 2.1 m. With these differences, the operating performance of the 350- and 450-MWt steam generators would be nearly the same.

R.2.1.6 Helium Circulator

The higher primary coolant flow rate for the 450-MWt design would increase the helium circulator pumping power requirement, which would be partially offset by the increase in the primary system pressure from 65 to 72 kg/cm². The resulting circulator would be larger than in the 350-MWt module, with a power rating 20% greater. Like the 350-MWt module, the 450-MWt design would include oil-bath-lubricated bearings, a radial flow impeller, and a submerged induction motor drive. Furthermore, the operating parameters (i.e., power rating, rotational speed, and bearing loading) would be within the experience envelope of existing circulators, although not in the exact reactor configuration or orientation.

R.2.1.7 Reactor and Steam Generator Vessels

The diameter of the steam generator vessel would be increased for the 450-MWt design by 0.91 m to accommodate the larger steam generator bundle, and the diameter of the reactor vessel would be increased by 0.67 m to accommodate the larger reactor core. The cross-vessel diameter would also be increased by 0.23 m to maintain coolant flow conditions in the cold and hot legs of the cross-vessel within acceptable limits. Because of the larger diameter and the increase in primary coolant pressure, the walls of both vessels would need to be slightly thicker and the vessels heavier.

The outside diameter of the containment silo would be 32 m and 27.74 m for the 450-MWt and 350-MWt reactor modules, respectively. The silo for the 450-MWt reactor would be 50.29 m high compared with 51.51 m high for the 350-MWt reactor because the steam generator is shorter.

R.2.2 Support Facilities

The facility types and descriptions presented in the EIS for the 350-MWt MHTGR complex would also apply to the 450-MWt MHTGR complex with the exception of two minor issues. First, there would be no change in tritium- or plutonium-target throughput, but there would be a 5% decrease in uranium throughput. These minor throughput adjustments would not significantly impact the support facilities. Second, the fuel and target fabrication facilities

would require somewhat larger product storage capacity because the 450-MWt reactor module reload quantities would be about 130% of the 350-MWt reactor module reload quantities. This increased storage capacity would require an increase of less than 30% in the storage cell capacity. No discernible increase in the layout of the facilities would occur because the product storage cell areas represent only a small portion of the overall fabrication facilities.

R.3 AFFECTED ENVIRONMENT

A description of the affected environment at Hanford, INEL, and SRS is provided in the EIS and is independent of reactor design. Either complex would be sited at the same location described in the EIS. Therefore, descriptions of the affected environment in the EIS apply.

R.4 CONSTRUCTION RESOURCES

R.4.1 Reactor Facility and Energy Conversion Area

R.4.1.1 Work Force Requirements

The total number of construction work force would be the same for both the 350-MWt and 450-MWt reactor facilities. The peak construction work force would probably increase slightly, i.e., less than 10%.

R.4.1.2 Material Requirements

Construction of a 450-MWt Reactor Facility would require slightly fewer materials than construction of the 350-MWt Reactor Facility. While the 450-MWt units would require more materials, the increase in materials would be offset by the deletion of two modules. These differences in materials would not result in significant overall effects.

R.4.1.3 Land Use Requirements

Construction of the 450-MWt MHTGR complex would disturb the same, or slightly less, land as the 350-MWt MHTGR complex. The area for each 450-MWt reactor module would be slightly larger than the area for a 350-MWt reactor module, but this increase in land area would not have a significant effect because six modules would be built instead of eight.

R.4.1.4 Schedule

The design, construction, certification, and preoperational testing schedule for the 450-MWt MHTGR complex has been reviewed and would be essentially the same as for the 350-MWt MHTGR. The completion of the first 450-MWt module is estimated to take three months longer than construction of the first 350-MWt unit. Because of the flexibility associated

with the modular design, it is possible to complete construction of the six-module 450-MWt MHTGR as much as six months earlier than the schedule presented in the EIS.

R.4.1.5 Transportation System

Material transportation and construction equipment fuel use rates during construction of the 450-MWt Reactor Facility would be bounded by the estimates for the 350-MWt Reactor Facility because the construction time and effort would be approximately the same.

R.4.2 Support Facilities

The support facility construction requirements presented in the EIS would apply directly to the 450-MWt MHTGR complex. The fabrication facility product storage cells would be slightly larger for the 450-MWt MHTGR complex to accommodate the 30% larger reactor module reload requirements, but this would not significantly affect the overall facility sizes, construction personnel, or material requirements.

R.5 OPERATIONAL RESOURCES

R.5.1 Reactor Facility and Energy Conversion Area

R.5.1.1 Work Force Requirements

Estimates of the work force required for normal operation of a six-module 450-MWt Reactor Facility would be about 3% less than for the eight-module 350-MWt Reactor Facility. This minor decrease (1%) in overall production complex operating personnel would not significantly impact the overall project.

R.5.1.2 Material Requirements

The types and quantities of materials required for the 450-MWt Reactor Facility would be unchanged relative to the 350-MWt Reactor Facility, including hazardous and nonhazardous materials (e.g., oils, lubricants, and cleaning and decontamination solutions), as well as fuel for standby power generation. The total standby power generator requirements for the 350-MWt Reactor Facility (10,000 kWe) would bound the standby power requirements for the 450-MWt Reactor Facility.

R.5.1.3 Transportation System

Transportation system requirements for workers would decrease slightly because of the slight (approximately 1%) decrease in work force for the overall production complex. Therefore, the transportation requirements for the 350-MWt MHTGR complex would also be bounding for the 450-MWt complex.

R.5.2 Support Facilities

The support facilities operations described in the EIS would apply directly for the 450-MWt MHTGR complex. The processes, work force, material requirements, and transportation requirements for the 450-MWt support facilities would be the same as for the 350-MWt support facilities. The tritium- or plutonium-target throughput would be unchanged, but the uranium throughput would decrease by 5%, which would affect the material requirements correspondingly. This throughput adjustment would affect the 350-MWt support facility designs, but would not affect operations.

R.6 CONSEQUENCES

R.6.1 Air Resources

R.6.1.1 Air Quality

Air quality impacts resulting from the 450-MWt MHTGR complex would be equal to, or slightly less than, air quality impacts determined for 350-MWt MHTGR complex.

Construction

The following parameters were examined to determine potential changes in air quality impacts resulting from construction of the 450-MWt MHTGR complex configuration: construction site fugitive dust, construction equipment exhaust emissions, road dust from concrete hauling, and vehicle emissions from commuting construction workers.

For the 350-MWt MHTGR complex, construction site fugitive dust emissions were found to be a major contributor to the ambient air concentrations of total suspended particulate and particulate matter less than 10 μm in diameter. These emissions are dependent on the size of the construction area being worked during the excavation phase of construction. Site fugitive dust emissions for construction of the 450-MWt MHTGR complex would be bounded by those presented in the EIS for the 350-MWt MHTGR complex because excavation required for the 450-MWt Reactor Facility would be slightly less (because of the reduced number of reactor silos), and excavation required for 450-MWt MHTGR complex support facilities would remain unchanged.

Other sources of construction emissions would result from equipment exhaust emissions, road dust from concrete hauling, and vehicle emissions from commuting construction workers. For the 350-MWt MHTGR complex, these sources would produce ambient air concentrations substantially less than National Ambient Air Quality Standards. Construction of the 450-MWt MHTGR complex would use about the same number of vehicles for excavation, but would require slightly less concrete and fewer construction workers. Air quality impacts for these sources, therefore, would be bounded by the air quality impacts determined for the 350-MWt MHTGR complex.

Operations

Air quality impacts calculated for the 350-MWt MHTGR complex would be negligible. Emissions were based on operating four 2,500-kWe generators 2 hours per month. The standby power requirements for the 450-MWt Reactor Facility would be well within this 10,000-kWe specification. Therefore, no increase in air quality impacts from this source is anticipated.

Overall reactor staffing would be approximately 3% less for the 450-MWt MHTGR complex than the 350-MWt MHTGR complex because of the reduced number of modules. Ambient air quality impacts resulting from vehicle emissions during worker commuting, therefore, would be indistinguishable between the two reactor concepts.

No other changes in toxic or criteria pollutant emission rates would occur as a result of operating the 450-MWt MHTGR complex. The emissions from support and waste management facilities would not be affected.

R.6.1.2 Cooling Towers

The 450-MWt Reactor Facility would require slightly less waste heat dissipation than the 350-MWt Reactor Facility (2,700 vs. 2,800 MW). Cooling tower design parameters and, therefore, potential environmental impacts resulting from the cooling tower plume for the 450-MWt complex would be similar to the assessment for the 350-MWt complex.

R.6.1.3 Noise

Conservative sound power level calculations indicated that construction and operation of the 350-MWt MHTGR complex would produce a negligible impact at the nearest site boundary. Vehicular traffic would produce a slight but measurable impact along associated traffic routes.

The construction noise impact associated with a 450-MWt MHTGR complex would be slightly less than the reference design because of reductions in blasting and excavation requirements, concrete hauling, and construction worker vehicle traffic. No changes would be required for cooling towers and standby generators; therefore, any changes in noise impacts from these operations would be negligible. The reduction in operational work force requirements (1%) for the 450-MWt MHTGR complex would have a negligible effect on noise along main traffic routes.

R.6.2 Water Resources

Potential impacts to water resources for constructing and operating a 350-MWt MHTGR complex are provided in the EIS. These impacts would also apply to the 450-MWt MHTGR complex, except for two minor differences. First, there would be a slight decrease in excavation activities and slightly fewer construction workers during the construction period because fewer reactor silos would be built. As a result, less water would be required for dust control and worker needs. Second, slightly less water would be required for reactor cooling and the resultant blowdown because of the reduction in operating power from 2,800 to 2,700 MWt.

R.6.3 Health and Safety

Assessments for the 350-MWt MHTGR complex were reviewed to estimate the health and safety impacts of operating a 450-MWt MHTGR complex. In many cases, it was possible to demonstrate directly (i.e., without extensive analysis) that no significant differences in environmental impacts would occur between the two designs. In some cases, the 450-MWt Reactor Facility would show an increase in radionuclide releases.

R.6.3.1 Reactor Facility and Energy Conversion Area

For the Reactor Facility, the radiological impacts of normal operations, nonsevere accidents, and severe accidents were evaluated by calculating radionuclide inventories and releases for a 450-MWt reactor module. The tritium production mode would have greater impacts to normal operations, nonsevere accidents, and severe accidents.

Normal Operations

Radionuclides would be released from the MHTGR during normal operations from two sources: leakage of the primary system coolant including circulating activity, and tritium migration across the steam generator tubes, which would be vented from the secondary system to the environment. For both sources, the normal releases associated with the 450-MWt Reactor Facility would be bounded by the normal releases from the 350-MWt Reactor Facility. The rationale for this conclusion is as follows. Release of the radionuclides in the helium primary coolant as a result of leakage would equal the product of the radionuclide inventory in the primary coolant and the fractional helium leakage rate. The inventory of the radionuclide in the primary coolant in a 450-MWt reactor module would be greater than that in a 350-MWt reactor module by about 29% (approximately the ratio of the power levels). The design requirement for helium leakage from the primary system would be the same for both designs. Factoring in the number of modules, the radionuclides released from the 450-MWt Reactor Facility would be bounded by those from the 350-MWt Reactor Facility.

The migration of tritium across the steam generator tubes is dependent on the partial pressure of tritium in the primary system and the surface area and temperature of the steam generator tubes. The surface area of the high-temperature portion of the steam generator tubes is proportional to the module power rating. The partial pressure of tritium in a 450-MWt reactor module would be similar to that in a 350-MWt reactor module because the purification flow rate would be increased in proportion to the power level. Because the tube temperatures would be nearly equal, the tritium migration across the steam generator tubes in a single module would be proportional to the module power rating. Considering the number of modules, the total release of tritium from the secondary side of the 450-MWt reactor facilities would be slightly less than the 350-MWt reactor facilities (2,700 MW compared with 2,800 MW).

Nonsevere and Severe Accidents

The bounding nonsevere and severe accidents for the 450-MWt reactor module were identified and analyzed. Results from the analyses were compared with the results for a 350-MWt reactor module. For the purpose of this analysis, nonsevere accidents have a frequency of occurrence of greater than one event in 1 million years; severe accidents have a frequency of occurrence in the range from one event per 1 million years to one event per 1 billion years.

Bounding Accident Selection. Bounding accident scenarios were selected by reviewing accident assessments performed by EG&G (EG&G 1991) on a 450-MWt reactor module, and evaluating the potential differences between the 450- and the 350-MWt reactor modules that could result in identifying new accident scenarios (Dinneen 1990). This assessment resulted in several significant findings.

First, the sizes of the primary coolant system penetration lines for the 450-MW design would be either larger than (e.g., purification line) or the same as (e.g., instrumentation line) those for the 350-MWt design. Also, the containment volume would be increased in proportion to the mass of helium in the primary coolant system. Therefore, the small break, moderate break, and air ingress accident scenarios would be similar.

Second, both designs would have similar inherent shutdown capabilities (negative reactivity temperature coefficient) and similar safety margins for establishing safety relief valve settings. Radionuclides would be released from the fuel and targets to the primary system during anticipated transient without scram (ATWS) accident scenarios for both designs. The primary system pressure would remain below the safety relief valve pressure settings for both designs during ATWS accidents; therefore, the radionuclides would be contained within the primary system.

Third, the amount of positive reactivity insertion during control rod withdrawal would be similar in both designs. Therefore, the control rod withdrawal accidents for the 450-MWt reactor design would be comparable to those for the 350-MWt reactor module, with both designs containing all of the radionuclides within the primary system.

Fourth, moisture ingress accidents would depend on the dimensions of the individual steam generator tubes, the water inventory of the steam generator, and primary coolant system design. These accidents would be similar for both designs because the diameters of the steam generator tubes would be the same, and the steam generator and primary coolant system volumes would be increased proportionally for the 450-MWt reactor design.

Finally, thermal analysis of the 450-MWt reactor design indicates that core temperatures would be slightly higher for the depressurized conduction cooldown transients with and without the reactor cavity cooling system (RCCS). This difference and the higher radionuclide inventory per module would require reassessment of the radionuclide releases during these accidents.

On the basis of the above assessment, no new accident scenarios have been postulated for the 450-MWt reactor design that were not postulated for the 350-MWt reactor design. The containment response for the 450- and 350-MWt accident scenarios would be similar. Therefore, the same bounding accident scenarios apply for both designs.

The bounding accidents for either design and their associated frequencies of occurrence are as follows:

- *Nonsevere accident* -- The depressurized conduction cooldown with RCCS operational and a containment leak rate of 100% per day for the first puff and 1% per day for the second puff (frequency of 4.5×10^{-5} per reactor-year for the six-module 450-MWt reactor facility and 6.0×10^{-5} per reactor-year for the eight-module 350-MWt reactor facility).

- *Severe accident* -- The depressurized conduction cooldown without RCCS and a containment leak rate of 100% per day for the first puff and 1% per day for the second puff (frequency of 4.5×10^{-8} per reactor-year for the 450-MWt reactor facility and 6.0×10^{-8} per reactor-year for the 350-MWt reactor facility).

Comparison of Accident Results. Because of the higher power level and slightly larger core, the temperature response of the 450-MWt core would be different from that for the 350-MWt core. In particular, calculations performed by EG&G (EG&G 1991) indicate higher temperatures and longer times at temperature for the 450-MWt core compared with the 350-MWt core, resulting in greater radionuclide releases during the two bounding accidents.

The effects of the more severe thermal transients on radionuclide releases during accident conditions were analyzed using the identical models used in the analysis of the reference 350-MWt core for the EIS. The temperatures used in the assessment of the 450-MW core were developed using the following procedure:

- A temperature increment was developed for each accident, based on the difference in the EG&G-calculated peak temperatures for the 350- and 450-MWt cores.
- This time-dependent peak temperature difference was added to the temporal and spatial 350-MWt temperature distribution used in the original EIS analysis.

This procedure was used to provide a common basis to compare the new 450-MWt source terms with the 350-MWt source terms previously generated for the EIS. The peak temperatures for the 350- and 450-MW designs for a depressurized conduction cooldown accident with and without RCCS are shown in Figures R.2 and R.3, respectively. A comparison of the resulting release fractions for the nonsevere and severe accidents is presented in Table R.2 for both designs. The results of the release calculations are contained in an Engineering Design File (Petti 1991).

Nonsevere Accidents. For the bounding nonsevere accident (depressurized conduction cooldown), the estimated doses and health effects for the off-site and colocated worker have been calculated and are presented in Tables R.3 and R.4. For the nonsevere accident scenario, the releases associated with the 450-MWt reactor module would be higher by 30% to 90%, depending on the specific radionuclide. When the increases associated with each specific radionuclide are combined, the resulting impact analyses show that the doses and health effects would increase approximately 60% to 80%. On the basis of these results, the radiological consequences for the 450-MWt Reactor Facility remain less than the regulatory guidelines (DOE Order 5480.6 and 10 CFR 100).

Severe Accidents. For severe accident scenarios, the releases associated with the 450-MWt reactor module would increase by a factor of 1.3 to 4.8 over the 350-MWt reactor module, depending on the radionuclide. Scoping analyses indicate that health effects would increase by a factor of 2 to 4 over the 350-MWt reactor module, depending on the accident

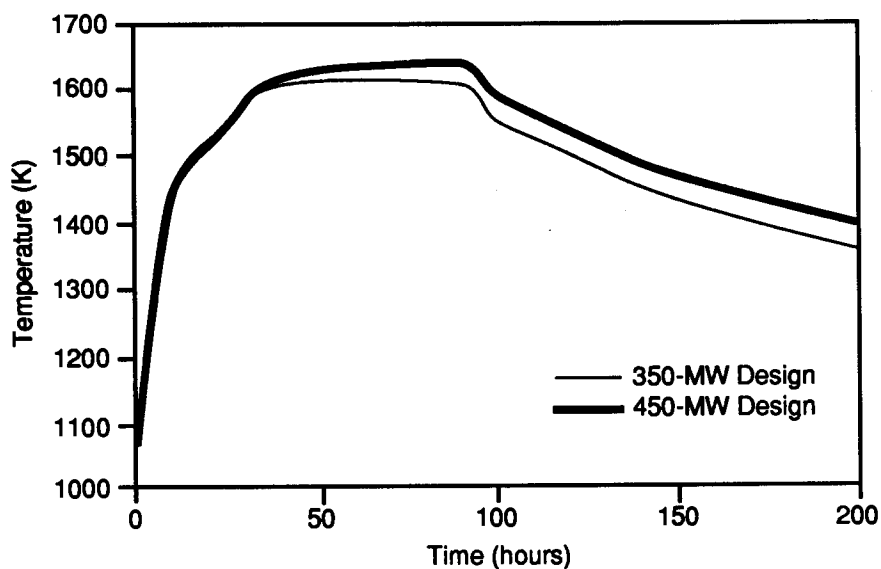


FIGURE R.2 Comparison of Peak Temperatures during a Depressurized Conduction Cooldown Accident with RCCS for the 350- and 450-MWt MHTGR Designs (Source: Modified from EG&G 1991)

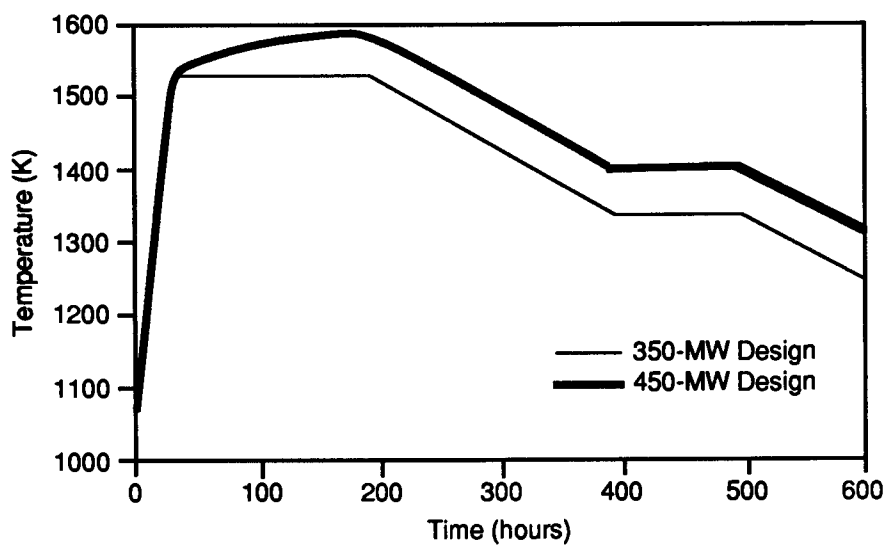


FIGURE R.3 Comparison of Peak Temperatures during a Depressurized Conduction Cooldown Accident without RCCS for the 350- and 450-MWt MHTGR Designs (Source: Modified from EG&G 1991)

TABLE R.2 Comparison of Release Fractions for the 350- and 450-MWt Designs

Element Group ^a	Release Fractions (RFs) from Fuel and Targets		Release Fractions from Graphite		Effective Ratio of RFs ^b
	450 MWt	350 MWt	450 MWt	350 MWt	
Nonsevere Accidents					
H-3	0.16	0.11	0 ^c	0 ^c	1.4
Cs	0	0	0.36	0.32	1.1
Sr	0	0	0.27	0.23	1.2
Ag	0	0	0 ^c	0 ^c	NA ^d
Kr	0	0	0 ^c	0 ^c	NA
Severe Accidents					
H-3	0.56	0.44	0 ^c	0 ^c	1.3
Cs	0	0	0.75	0.67	1.1
Sr	9.6×10^{-3}	2.2×10^{-3}	0.63	0.53	3.7 ^e
Ag	0.012	6.7×10^{-5}	0 ^c	0 ^c	1.8
Kr	0	0	0 ^c	0 ^c	NA

^aElement groups: H-3 = tritium; Cs = cesium; Ag = silver; Kr = noble gases, halogens, tellurium, and rubidium; Sr = strontium plus all other elements.

^bThe effective ratio of RFs is defined as $[(RF)_{450-MW}]/[(RF)_{350-MW}]$.

^cNo inventory of this element in the graphite.

^dNA = not applicable.

^eThe effective ratio of RFs for Sr is defined as $[(RF)_{fuel} + 0.001 \times (RF)_{graphite}]_{450-MW} / [(RF)_{fuel} + 0.001 \times (RF)_{graphite}]_{350-MW}$

Source: Modified from EG&G 1991.

scenario. On the basis of these results, the risk associated with the 450-MWt Reactor Facility would be less than the NPR safety goals by several orders of magnitude.

Nonradiological Accidents

Industrial accidents present the greatest hazard to NPR personnel. Industrial hazards would remain the same for the 450-MWt MHTGR. Any hazardous materials associated with the Reactor Facility would be insignificant in comparison with the materials involved with the support facilities; therefore, there would be no overall increase in risk to the public.

R.6.3.2 Support Facilities

The health and safety impacts associated with the 450-MWt MHTGR complex support facilities would be less than or equal to the impacts reported for the 350-MWt MHTGR complex. The radiological and nonradiological impacts associated with normal operation of fuel fabrication

TABLE R.3 Estimated Doses for the Postulated MHTGR Nonsevere Accident with Highest Consequences at INEL for the 350- and 450-MWt Designs^a

Ingestion of Contaminated Foodstuffs Considered	Maximum Off-site Individual (rem)		Worst-Sector Population for 0 - 80 km (person-rem)		Maximum Collocated Worker (rem)	
	350 MWt	450 MWt	350 MWt	450 MWt	350 MWt	450 MWt
No	3.4×10^{-2}	5.4×10^{-2}	3.7×10^2	5.8×10^2	4.4×10^{-3}	5.7×10^{-3}
Yes	1.0	1.8	8.6×10^3	1.5×10^4	--	--

^aDepressurized conduction cooldown with RCCS and 100% per day leak rate for the first puff and 1% per day leak rate for the second puff.

Source: Modified from EG&G 1991.

TABLE R.4 Estimated Health Effects for the Postulated MHTGR Nonsevere Accident with Highest Consequences at INEL for the 350- and 450-MWt Designs^a

Ingestion of Contaminated Foodstuffs Considered ^b	Maximum Off-site Individual Fatal Cancers (rem)		Worst-Sector Population for 0 - 80 km (person-rem)				Maximum Colocated Worker Fatal Cancers (rem)	
			Fatal Cancers		Genetic Effects			
	350 MWt	450 MWt	350 MWt	450 MWt	350 MWt	450 MWt	350 MWt	450 MWt
No	1×10^{-5}	2×10^{-5}	None	None ^c	None	None	2×10^{-6}	2×10^{-6}
Yes	4×10^{-4}	7×10^{-4}	3	6	2	4	--	--

^aDepressurized conduction cooldown with RCCS and 100% per day leak rate for the first puff and 1% per day leak rate for the second puff.

^bIngestion of contaminated food is only applicable to off-site populations or individuals.

^cNone means the calculated value is less than 0.5.

Source: Modified from EG&G 1991.

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and processing facilities would be somewhat lower for the 450-MWt MHTGR complex because of a 5% decrease in uranium requirements.

The 350-MWt MHTGR complex nonsevere and severe radiological and nonradiological accident impacts associated with fuel and target fabrication and processing would apply directly for the 450-MWt MHTGR complex. The larger product storage size (130%) in the fabrication facilities (Section 4.2) is not important because accidents involving product storage are insignificant relative to other fabrication accidents.

R.6.4 Socioeconomics

As discussed in Section R.4, the construction schedule would remain unchanged, and the total construction work force would be slightly reduced for the 450-MWt MHTGR complex. There would be a minor decrease (1%) in net operational work force (Section R.5). The net effect on overall socioeconomic impacts would be negligible.

R.6.5 Transportation System

The minor decreases in construction work force, construction materials, operating personnel, and operating materials would result in slight reductions in the transportation requirements and impacts. The differences would be insignificant, and the EIS estimates can be used directly.

R.6.6 Waste Management

R.6.6.1 Construction

The construction waste types and quantities presented in the EIS would also be applicable and bounding for the 450-MWt MHTGR complex. The wastes associated with reactor construction would be slightly lower for the 450-MWt MHTGR Reactor Facility, but this reduction would not be significant. There would be no changes in the waste estimates for the support facilities because there is no significant change in the footprint or layout of these facilities.

R.6.6.2 Operations

The operating waste types and quantities presented in the EIS would be applicable and bounding for the 450-MWt concept (Table R.5). The most significant difference would be a 5% reduction in the volume of high-level waste (HLW). This reduction in HLW would result directly from the 5% reduction in fuel (both the number of blocks and mass of uranium) required by the 450-MWt MHTGR concept. This 5% reduction in HLW would result in substantial cost reductions because of the extremely high cost projected for HLW disposal. All other wastes (low-level, transuranic, mixed, hazardous, and nonradiological/nonhazardous) would remain the same or decrease slightly.

TABLE R.5 Waste Volume Estimate Changes for the Tritium and Plutonium Production Modes (%)

Production Mode/ Facility	High- Level Waste	Low- Level Waste	TRU ^a Waste	Mixed Waste	Hazardous Waste	Other ^b Waste
Tritium Production Mode						
Reactor Facility	NA ^b	<1 ^c	NA	<1	<1	<1
Fuel Fabrication	NA	-5	NA	-5	-5	0
Tritium-Target Fabrication	NA	NA	NA	NA	0	0
Headend Fuel Processing	NA	-5	-5	-5	-5	0
Fuel Processing	-5	-5	-5	-5	-5	0
Tritium-Target Processing	NA	0	NA	0	0	0
Cumulative	-5	<1	-5	<1	<1	<1
Plutonium Production Mode						
Reactor Facility	NA	<1	NA	<1	<1	<1
Fuel Fabrication	NA	-5	NA	-5	-5	0
Plutonium-Target Processing	NA	<1	NA	<1	<1	<1
Headend Fuel Processing	NA	-5	-5	-5	-5	0
Fuel Processing	-5	-5	-5	-5	-5	0
Plutonium-Target Processing	-5	-5	-5	-5	-5	0
Cumulative	-5	-5	-5	<1	<1	<1

^aTRU = transuranic.

^bNonradiological nonhazardous wastes.

^cNA = not available.

^d<1% indicates a minor decrease in the estimate, but it is not significant enough to report.

Source: Modified from EG&G 1991.

R.7 CUMULATIVE IMPACTS

The changes in cumulative impacts for the 450-MWt MHTGR complex would be insignificant and would not increase relative to the 350-MWt MHTGR complex.

R.8 DECONTAMINATION AND DECOMMISSIONING

The decontamination and decommissioning impact to the 450-MWt MHTGR complex compared with the 350-MWt MHTGR complex would result in a reduction of some waste volumes, especially for the Reactor Facility. The reduced volumes effectively would reduce the estimation of occupational doses and costs. Although the volumes are reduced, the reductions would be less than 5% and would have a minimal impact on the overall cost of the program.

R.9 ENVIRONMENTAL REGULATIONS

The environmental regulations applicable to construction and operation of a 350-MWt MHTGR complex would apply equally to the 450-MWt MHTGR complex because the area of land disturbance during construction, the overall size of the facilities, and the emissions during operations for both concepts are similar.

R.10 MHTGR AT HANFORD AND SRS

The relative environmental impacts associated with construction, operation, and decommissioning the 450-MWt MHTGR complex at Hanford or SRS would be similar to those described in Sections R.2 through R.9 for the 450-MWt concept at INEL. The discussions presented in Sections R.2 through R.9 for INEL include all issues with similar impacts, except for severe and nonsevere accidents. Impacts associated with severe and nonsevere reactor accidents at Hanford and SRS are discussed below.

The same bounding nonsevere and severe accidents addressed for the 350-MWt reactor modules would also apply for the 450-MWt reactor modules at Hanford and SRS. The radionuclide releases at Hanford and SRS for nonsevere and severe accidents and resulting fractional increases in doses and health effects would be the same as for INEL (see the Support Facilities discussion presented in Section R.6.3.1). The resulting doses for the nonsevere accidents would increase by approximately 60% to 80% at each site (Tables R.6 and R.7). On the basis of Tables R.6 and R.7, radiological consequences for the 450-MWt Reactor Facility would be less than the regulatory guidelines (DOE Order 5480.6 and 40 CFR 61) by at least an order of magnitude.

As discussed in Section R.6.3.1, the risks associated with increased severe accident releases would increase by approximately a factor of 2 to 4 at either site, but the risks would remain orders of magnitude lower than the NPR safety goals.

TABLE R.6 Estimated Doses for the Postulated MHTGR Nonsevere Accident with Highest Consequences at Hanford and SRS for the 350- and 450-MWt Designs^a

Location	Ingestion of Contaminated Foodstuffs Considered ^b	Maximum Off-site Individual (rem)		Worst-Sector Population for 0 - 80 km (person-rem)		Maximum Colocated Worker (rem)	
		350 MWt	450 MWt	350 MWt	450 MWt	350 MWt	450 MWt
Hanford	No	2.1×10^{-2}	3.2×10^{-2}	6.7×10^2	1.1×10^3	3.7×10^{-2}	4.7×10^{-2}
	Yes	6.7×10^{-1}	1.2	1.4×10^4	2.4×10^4	--	--
SRS	No	8.9×10^{-3}	1.4×10^{-2}	3.0×10^2	4.7×10^2	2.3×10^{-3}	3.0×10^{-3}
	Yes	2.4×10^{-1}	4.0×10^{-1}	3.8×10^3	6.4×10^3	--	--

^aDepressurized conduction cooldown with RCCS and 100% per day leak rate for the first puff and 1% per day leak rate for the second puff.

^bIngestion of contaminated food is only applicable to off-site populations and individuals.

Source: Modified from EG&G 1991.

TABLE R.7 Estimated Health Effects for the Postulated MHTGR Nonsevere Accident with Highest Consequences at Hanford and SRS for the 350- and 450-MWt Designs^a

Location	Ingestion of Contami- nated Foodstuffs Considered ^b	Maximum Off-site Individual Fatal Cancers (rem)		Worst-Sector Population for 0 - 80 km (person-rem)		Genetic Effects		Maximum Colocated Worker Fatal Cancers (rem)	
		Fatal Cancers (rem)		Fatal Cancers		Genetic Effects		Fatal Cancers (rem)	
		350 MWt	450 MWt	350 MWt	450 MWt	350 MWt	450 MWt	350 MWt	450 MWt
Hanford	No	8×10^{-6}	1×10^{-5}	None	None ^c	None	None	2×10^{-5}	2×10^{-5}
	Yes	3×10^{-4}	5×10^{-4}	6	10	4	7	—	—
SRS	No	4×10^{-6}	6×10^{-6}	None	None	None	None	9×10^{-7}	1×10^{-6}
	Yes	1×10^{-4}	2×10^{-4}	2	3	1	2	—	—

^aDepressurized conduction cooldown with RCCS and 100% per day leak rate for the first puff and 1% per day leak rate for the second puff.^bIngestion of contaminated food is only applicable for off-site populations or workers.^cNone means the calculated value is less than 0.5.

Source: EG&G 1991.

R.11 REFERENCES*

Dinneen, G.A., 1990, *Bounding Accident Selection*, EG&G Engineering Design File No. NPR-MHTGR-0172, Idaho Falls, Idaho, Dec. 11.

EG&G, 1991, *Generic Reactor Plant Description and Source Terms*, EG&G Idaho, Inc., report EGG-NPR-8522, Vol. 1, Rev. B., Idaho Falls, Idaho.

Petti, D.A., 1991, *Source Term Results for the 450-MW NPR-MHTGR*, EG&G Engineering Design File NPR-MHTGR-0180, Idaho Falls, Idaho, Jan. 23.

*Copies of correspondence and other unpublished material cited in this list are available in U.S. Department of Energy Reading Rooms.

