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## 4. ENVIRONMENTAL CONSEQUENCES

Chapter 4 describes the environmental consequences of the proposed action and alternatives to treat and manage the sodium-bonded spent nuclear fuel. It begins with a general discussion of the expected environmental impacts; the product and waste forms that would be generated from the proposed action; and the methodology for assessing health effects from radiological and chemical effluents. It follows with a detailed description of the environmental consequences for the No Action and the reasonable alternatives. The chapter provides separate discussions on the environmental consequences of the intersite transportation of the sodium-bonded spent nuclear fuel; the cumulative impacts at each of the proposed sites; and the programmatic considerations associated with the proposed action. The chapter concludes with a look at several issues under the proposed action, such as unavoidable, adverse environmental impacts; relationships between local, short-term uses of the environment and the enhancement of long-term productivity; and irretrievable commitments of resources.

### 4.1 OVERVIEW OF ENVIRONMENTAL IMPACTS

This *Draft Environmental Impact Statement for the Treatment and Management of Sodium-Bonded Spent Nuclear Fuel* (SBSNF EIS) is in compliance with Council on Environmental Quality regulations that require the affected environment of proposed Federal actions to be “interpreted comprehensively to include the natural and physical environment and the relationship of people with the environment” (40 CFR 1508.14).

The environmental consequence analysis focused on potentially affected areas. These areas are discussed in detail: air quality, water resources, socioeconomics, public and occupational health and safety (normal operations and accident conditions), environmental justice, waste management, and transportation. For the remaining areas (i.e., land resources, visual resources, noise, geology and soils, ecological resources, and cultural and paleontological resources), analyses show that the proposed treatment activities would have minimal or no impact at the candidate sites regardless of the alternatives being considered. This is because existing facilities within developed areas would be used; no new land disturbance would take place and proposed activities would be consistent with current operations. Since none of the alternatives analyzed in detail involve construction other than internal building modifications for installing new equipment, the effects of construction on any of the resources would be negligible and are not evaluated in this chapter.

The specific assumptions associated with the impact analysis common to all alternatives are provided in the appendices. The results of the assessment of environmental consequences are presented in this chapter. More detailed descriptions of the development of the impacts for some resource areas are presented in Appendices E through H, as follows:

- Appendix E, Evaluation of Human Health Effects From Normal Operations
- Appendix F, Evaluation of Human Health Effects From Facility Accidents
- Appendix G, Evaluation of Human Health Effects From Overland Transportation
- Appendix H, Analysis of Environmental Justice

#### 4.1.1 Presentation of the Environmental Impacts

The primary impacts of concern are products and wastes, impacts on the public, and occupational health and safety associated with the various sodium-bonded spent nuclear fuel treatment processes. Additional impacts and topics covered in Chapter 4 include the following:

- Air Quality
- Water Quality
- Environmental Justice
- Socioeconomics
- Waste Management
- Transportation
- Short-term versus Long-term Resource Commitments
- Irreversible and Irrecoverable Resource Commitments
- Cumulative Impacts

Several kinds of impacts are not discussed in Chapter 4 because they will not occur, they will be extremely small, and/or they are covered by other analyses:

**Land**—The treatment and management of sodium-bonded spent nuclear fuel would not require the construction of new facilities on previously undisturbed land at Argonne National Laboratory-West (ANL-W) or the Savannah River Site (SRS).

**Intrasite Transportation**—The incident-free impacts of intrasite transportation are limited to radiation exposure to workers loading and unloading trucks and are included in the overall worker dose values presented for each process. The accident risks are bounded by the site accident risk analysis. Strict site safety procedures and short travel distances limit the impacts to workers.

**Noise**—Noise impacts at the management sites should be minor and limited to noises generated during operations. No offsite noise impacts are expected except for minor changes in traffic noise levels.

**Ecological Resources**—Because no new construction in undisturbed areas would be required for the treatment and management of sodium-bonded fuel, there would be no disturbance to terrestrial and aquatic habitats or wetlands. Thus, there would be no negative impacts from construction on terrestrial or aquatic plants or animals, including threatened and endangered species.

Scientific evidence indicates that chronic radiation doses below 0.1 rad per day do not harm animal or plant populations (IAEA 1992). This is equivalent to 100 millirem per day for direct radiation and greater than 100 millirem per day for ingestion of plutonium. Compliance with DOE Order 5400.5 to limit the exposure of the most exposed member of the public to 100 millirem per year (i.e., about 0.3 millirem per day) makes it highly probable that dose rates to plants and animals in the same area would be less than 0.1 rad per day. Therefore, no radiological damage to plant and animal populations would be expected as the result of the sodium-bonded spent nuclear fuel treatment processes.

Chemicals emitted to the environment during routine processing activities from F-Canyon at SRS are presented in Section 4.5.1. In addition, Sections 4.5.4.1 and 4.5.4.2 contain modeled airborne concentrations for the chemicals emitted that have the potential to impact plants or animals. These chemicals would not impact plants or animals because either the amounts emitted are very low or the chemicals have little potential for causing negative effects.

For the reasons discussed above, no adverse impacts to ecological resources would be expected to occur due to DOE's treatment and management of sodium-bonded spent nuclear fuel.

**Cultural and Paleontological Resources**—No new facilities would be needed or constructed, therefore, there would be no impacts on cultural or paleontological resources.

**Geology and Soils**—No new facilities would be needed or constructed. Therefore, there would be no disturbance to either geologic or soil resources at the management sites. Hazards from large-scale geologic conditions were analyzed in detail in various DOE programmatic environmental impact statements and site-specific facility safety analysis reports. The impacts from these hazards (e.g., earthquakes) on the management facilities and treatment processes are evaluated in this environmental impact statement (EIS).

#### 4.1.2 Products and Wastes

**Generation**—All the treatment processing alternatives in this EIS, except for direct disposal in high-integrity cans, would change sodium-bonded spent nuclear fuel into other forms. Driver and blanket sodium-bonded spent nuclear fuels are inputs—products and wastes are the outputs. The products and wastes are better suited for storage, transportation, and disposal or other disposition than the existing sodium-bonded fuel. The products and wastes fall into several distinct categories:

Materials to be managed as high-level radioactive waste would be generated at SRS and ANL-W. The final form would be solid ceramic, metal, or borosilicate glass inside stainless steel canisters. This waste would be stored at SRS and/or ANL-W until a geologic repository is ready to receive it.

Transuranic waste refers to processed materials that contain alpha-emitting material concentrations (such as plutonium) above 100 nanocuries per gram of waste. Transuranic waste would be generated from all treatment technologies. This waste could be disposed of in the Waste Isolation Pilot Plant.

The separated uranium resulting from the electrometallurgical treatment process at ANL-W would be made into solid metal ingots. The separated uranium resulting from processing the driver spent nuclear fuels would be made into low-enriched uranium ingots. The uranium products would be stored in secure facilities along with other uranium already in storage at ANL-W.

Separated depleted uranium from plutonium-uranium extraction (PUREX) processing of declad and cleaned blanket spent nuclear fuel at SRS would be made into uranium oxides and stored in drums along with other depleted uranium at SRS (more than 27,000 metric tons of depleted uranium is currently stored at SRS). The 57 metric tons of depleted uranium are a small fraction of what is currently stored.

Separated plutonium resulting from PUREX processing of declad and cleaned blanket spent nuclear fuel at F-Canyon would be in a metal form. The separated plutonium would be stored in secure facilities along with the plutonium already in storage at SRS until decisions are made about its disposition. DOE would not use this plutonium for nuclear explosive purposes (DOE 1994a).

Low-level radioactive waste would be generated from all treatment technology alternatives considered. This waste would be disposed of in existing facilities using routine procedures.

Saltstone would be generated only at SRS. Saltstone is a form of concrete containing low levels of radioactivity and would be disposed of on site.

**Waste Minimization**—DOE would incorporate the best available practices into all the processing technologies at the two management sites to generate the smallest possible amount of waste. The DOE sites managing the sodium-bonded spent nuclear fuel would comply with DOE's waste minimization and pollution prevention goals. The following summarizes recent achievements in pollution prevention and waste minimization at ANL-W and SRS.

ANL-W conducted pollution prevention projects in 1997 that reduced waste generation by an estimated 1,700 cubic meters (61,100 cubic feet) at a cost savings of \$154,000. Radioactive waste generation in 1997

was reduced by 61 percent compared to 1993 baseline levels. Mixed waste generation was increased by 67 percent, hazardous waste generation was reduced by 44 percent, and sanitary waste generation was reduced by 32 percent compared to baseline levels. Fifty-six percent of sanitary waste was recycled in 1997. ANL-W affirmative procurement purchases are not tracked separately, and are included in the Idaho National Engineering and Environmental Laboratory (INEEL) totals. For INEEL, 72 percent of the materials purchased were U.S. Environmental Protection Agency (EPA)-designated recycled products (DOE 1998f).

SRS conducted pollution prevention projects in 1997 that reduced waste generation by an estimated 18,200 cubic meters (644,000 cubic feet) at a cost savings of \$18.5 million. Radioactive waste generation in 1997 was reduced by 57 percent compared to 1993 baseline levels. Mixed waste generation was increased by 115 percent, hazardous waste generation was reduced by 15 percent, and sanitary waste generation was reduced by 58 percent compared to baseline levels. Seventy-eight percent of sanitary waste was recycled in 1997, and 52 percent of the materials purchased under the affirmative procurement process were EPA-designated recycled products (DOE 1998f).

#### **4.1.3 General Radiological and Chemical Health Consequences**

The methodologies used to evaluate potential radiological and chemical health effects are described in Appendix E. This section provides information about the development and interpretation of the health risk estimates.

**Radiological**—The effect of radiation on people depends upon the kind of radiation exposure (alpha, beta, and neutron particles and gamma and x-rays), duration of exposure, and the total amount of tissue exposed to radiation. The amount of radiant energy imparted to tissue from exposure to ionizing radiation is referred to as “absorbed dose.” The sum of the absorbed dose to each tissue, when multiplied by certain quality and weighting factors that take into account radiation quality and different sensitivities of these various tissues, is referred to as “effective dose equivalent.”

An individual may be exposed to radiation from outside or inside the body, because radioactive materials may enter the body by ingestion or inhalation. External dose is different from internal dose in that it is delivered only during the actual time of exposure. An internal dose, however, continues to be delivered as long as the radioactive source is in the body (although both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time). The dose from internal exposure is calculated over 50 years following the initial exposure.

The regulatory annual radiation dose limits to the maximally exposed offsite individual from total operations at a DOE site are 10 millirem from atmospheric pathways, 4 millirem from drinking water pathways, and 100 millirem from all pathways combined (DOE Order 5400.5 and 40 CFR Part 61, Subpart H). The potential doses associated with the normal operation of various treatment technologies and storage of sodium-bonded spent nuclear fuel are very small fractions of these values, and total site doses will remain well within these DOE limits. For comparison, DOE estimates that the average individual in the United States receives a dose of approximately 350 millirem per year from all radiation sources combined, including natural and medical sources.

The collective or “population” dose to an exposed population is calculated by summing the estimated doses received by each member of the exposed population. The total population dose received by the exposed population is measured in person-rem. For example, if 1,000 people each received a dose of 0.001 rem, the population dose would be 1 person-rem (1,000 persons  $\times$  0.001 rem = 1 person-rem). The same population dose (1 person-rem) would result if 500 people each received a dose of 0.002 rem (500 persons  $\times$  0.002 rem = 1 person-rem).

Radiation can cause a variety of adverse health effects in people. A large dose of radiation can cause prompt death. At low doses of radiation, the most important adverse health effect from environmental and occupational radiation exposures (which are typically low doses) is the potential inducement of fatal cancers. This effect is referred to as “latent cancer fatalities” because the cancer may take years to develop and for death to occur.

In addition to latent cancer fatalities, other health effects could result from environmental and occupational exposures to radiation. These effects include nonfatal cancers among the exposed population and genetic effects in subsequent generations. Nonfatal cancers and genetic effects are less probable consequences of radiation exposure. For simplicity, this EIS presents estimated effects of radiation only in terms of latent cancer fatalities. Estimates of the total detriment (fatal cancers, nonfatal cancers, and genetic effects) due to radiation exposure may be obtained from the estimates of latent cancer fatalities presented in this EIS by multiplying by 1.4 for workers and by 1.46 for the general public. The dose-to-effect factors for fatal and nonfatal cancers are shown in **Table 4–1**.

**Table 4–1 Risk of Latent Cancer Fatalities and Other Health Effects  
From Exposure to 1 Rem of Radiation<sup>a</sup>**

<i>Population<sup>b</sup></i>	<i>Latent Cancer Fatalities</i>	<i>Nonfatal Cancers</i>	<i>Genetic Effects</i>	<i>Total Detriment</i>
Workers	0.0004	0.00008	0.00008	0.00056
Public	0.0005	0.0001	0.00013	0.00073

<sup>a</sup> When applied to an individual, units are lifetime probability of a latent cancer fatalities per rem of radiation dose. When applied to a population of individuals, units are excess number of cancers per person-rem of radiation dose. Genetic effects as used here apply to populations, not individuals.

<sup>b</sup> The difference between the worker risk and the general public risk is attributable to the fact that the general population includes more individuals in the more sensitive age group of less than 18 years of age.

Note: One rem equals 1,000 millirem.

The factors used in this EIS to relate a dose to its effect are 0.0004 latent cancer fatalities per person-rem for workers and 0.0005 latent cancer fatalities per person-rem for individuals among the general population. The latter factor is slightly higher because some individuals in the public, such as infants and children, are more sensitive to radiation than workers. These factors are based on the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP 1991) and are consistent with those used by the U.S. Nuclear Regulatory Commission (NRC) in its rulemaking *Standards for Protection Against Radiation* (10 CFR 20). The factors apply where the dose to an individual is less than 20 rem and the dose rate is less than 10 rem per hour. At higher doses and dose rates, the factors used to relate radiation doses to latent cancer fatalities are doubled. At much higher doses, prompt effects, rather than latent cancer fatalities risk, may be the primary concern.

These concepts may be applied to estimate the effects of exposing a population to radiation. For example, if 100,000 people were each exposed only to natural background radiation (0.3 rem per year), 15 latent cancer fatalities per year would be expected (100,000 persons × 0.3 rem per year × 0.0005 latent cancer fatalities per person-rem = 15 latent cancer fatalities per year).

Sometimes calculations of the number of latent cancer fatalities associated with radiation exposure do not yield whole numbers and, especially in environmental applications, may yield numbers less than 1. For example, if 100,000 people each were exposed to a total dose of only 1 millirem (0.001 rem), the population dose would be 100 person-rem, and the corresponding estimated number of excess latent cancer fatalities would be 0.05 (100,000 persons × 0.001 rem × 0.0005 latent cancer fatalities per person-rem = 0.05 latent cancer

fatalities). The latent cancer fatality rate of 0.05 is the *expected* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people. In most groups, nobody (0 people) would incur a latent cancer fatality from the 1 millirem dose each member would have received. In a small fraction of the groups, 1 latent cancer fatality would result; in exceptionally few groups, 2 or more latent cancer fatalities would occur. The average number of deaths for all the groups would be 0.05 latent cancer fatalities (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is 0 latent cancer fatalities.

These same concepts apply to estimating the effects of radiation exposure on a single individual. Consider the effects, for example, of exposure to natural background radiation over a lifetime. The latent cancer fatality risk corresponding to a single individual's exposure to 0.3 rem per year over a (presumed) 72-year lifetime is:

$$1 \text{ person} \times 0.3 \text{ rem per year} \times 72 \text{ years} \times 0.0005 \text{ latent cancer fatalities per person-rem} = 0.011$$

latent cancer fatalities, or slightly more than 1 chance in 100 of a latent cancer fatality.

Again, this is a statistical estimate. That is, the estimated effect of natural background radiation exposure on the exposed individual would produce a 1.1 percent chance that the individual would incur a latent cancer fatality. Presented another way, this method estimates that about 1 person in 91 would die of cancer induced by natural background radiation.

The estimates of health effects from radiation doses used in this EIS are based on the linear no-threshold theory of radiation carcinogenesis, which postulates that all radiation doses, even those close to 0, are harmful. A recent examination of low radiation studies has reported that no statistically significant low-dose radiation study was found to support the linear no-threshold theory (Pollycove 1997). This finding is supported by the National Council of Radiation Protection and Measurements in a report on collective dose that states “. . . essentially no human data can be said to prove or even to provide direct support for the concept of collective dose with its implicit uncertainties of nonthreshold, linearity, and dose-rate independence with respect to risk” (NCRP 1995). Accordingly, calculations of health impacts based on the linear no-threshold theory may overstate the actual impacts of low radiation doses and should be viewed as an upper bound on the potential health effects.

**Chemical**—The potential impacts of exposure to hazardous chemicals released to the atmosphere as a result of the processing of sodium-bonded spent nuclear fuel were evaluated for the incident-free operation and accident conditions at management facilities. No hazardous chemicals are expected to be released from incident-free operation of the treatment technologies at ANL-W. The receptors considered in these evaluations include the offsite population in the vicinity of the sites and noninvolved workers located on site at SRS. Impacts also were evaluated for the maximally exposed offsite individual. The health effect endpoints evaluated in this analysis include excess latent cancers and chemical-specific noncancer health effects. The maximally exposed individual is located in the region with the highest estimated concentration. The hazardous chemical impacts are evaluated in terms of comparison to Emergency Response Planning Guideline. Emergency Response Planning Guidelines values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects. (See Appendix F, Section F.3.1.2, for more detail.)

## 4.2 NO ACTION ALTERNATIVE

Under the No Action Alternative, the sodium-bonded spent nuclear fuel would not be treated (no sodium would be removed from the interior of the fuel elements). Under this alternative, two options are evaluated:

- a. The SBSNF EIS evaluates the impacts of the activities required to monitor and stabilize the sodium-bonded spent nuclear fuel as necessary for continued safe and secure storage indefinitely at current locations, or

until a new treatment technology, such as the glass material oxidation and dissolution system (GMODS) or plasma arc, is developed. (See Section 2.6 for more details on GMODS and plasma arc.)

- b. The SBSNF EIS evaluates the impacts of direct disposal of sodium-bonded spent nuclear fuel in a geologic repository by packaging the fuel in high-integrity cans with minimal preparation.

Under both options, the EIS evaluates the impacts associated with activities required to clean and stabilize the waste materials generated during the Electrometallurgical Treatment Demonstration Project at ANL-W. Under this demonstration project, a total of approximately 1.6 metric tons of heavy metal of Experimental Breeder Reactor II (EBR-II) fuels consisting of about 1 metric ton of blanket spent nuclear fuel and 0.6 metric tons of driver spent nuclear fuel would be processed. The waste materials generated in this project currently are being transformed to ceramic and metallic waste forms, but the majority of wastes still would need stabilization. In addition, at the completion of the demonstration project, any remaining sodium-bonded spent nuclear fuel in the treatment facilities would be packaged and transferred to dry storage in the Radioactive Scrap and Waste Facility. Spent nuclear fuel transfer activities and waste processing activities would be completed in about two years after the necessary waste stabilization equipment is installed.

DOE is transferring all INEEL spent nuclear fuel, including the sodium-bonded spent nuclear fuel currently stored at the Idaho Nuclear Technology and Engineering Center (INTEC) Building 603 (wet storage), to dry storage. During transfer, each fuel can containing sodium-bonded fuel would be nondestructively examined to determine the fuel and can conditions and their suitability for storage. If any fuel can was found to be degraded and causing water inleakage, it would be repackaged and transferred to ANL-W for stabilization and/or recanning for storage. As stated in the amended Record of Decision for the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, (Programmatic Spent Nuclear Fuel EIS) (61 FR 9442), future sodium-bonded spent nuclear fuel transfers to Idaho would be packaged and stored at INTEC. If direct disposal of the sodium-bonded fuel becomes feasible, the stored fuel at the Radioactive Scrap and Waste Facility and at INTEC would be transferred to a dry spent nuclear fuel storage facility (to be built at INTEC) to be repackaged for offsite transport and disposal at a geologic repository.

The activities associated with the preparation of sodium-bonded spent nuclear fuel for direct disposal would be similar to those needed to prepare the fuel for interim or indefinite storage. Both require that the fuel be transferred to a hot cell, examined (nondestructive examination) and characterized, and repackaged. The only difference between these two options is that for direct disposal, the sodium-bonded spent nuclear fuel would be placed in high-integrity cans in preparation for ultimate disposal, while for storage it would not be placed in high-integrity cans. Direct disposal also requires consideration of criticality safety, thereby limiting the amount of driver spent nuclear fuel that could be packaged in a canister, leading to higher repository volume needs.

The impacts presented below would be applicable to either option considered under the No Action Alternative. The only impact that is different between the two options is the volume of high-level radioactive waste presented in Table 4-8. All other impacts are identical for each of the two options.

#### **4.2.1 Air Quality**

##### *Nonradiological Gaseous Emissions*

It is expected that activities under either option of this alternative would have a small impact on existing air quality at ANL-W, as any nonradiological emissions would be very low and well below the regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

### *Radiological Gaseous Emissions*

Potential radiological releases from spent nuclear fuels during storage periods at INEEL were estimated based on the information provided in the No Action Alternative in the Programmatic Spent Nuclear Fuel EIS (DOE 1995a). Normal spent nuclear fuel storage activities would produce radiological air emissions that are small compared to radiological air emissions from other activities at INEEL, such as calciner operations at INTEC and reactor operations at the Test Reactor Area and ANL-W. The current estimates of radiological emissions are significantly lower than those used for the evaluation of impacts in the Programmatic Spent Nuclear Fuel EIS. For example, in 1997, the ANL-W facilities released only 1.14 curies of krypton-85 (DOE 1998c) as compared to the estimate of 13,000 curies used earlier in the Programmatic Spent Nuclear Fuel EIS. However, degradation of sodium-bonded spent nuclear fuel during storage cannot be ruled out. It is expected that a percentage of fuel would be degraded during storage, allowing the gaseous fission products to enter the storage environment. During fuel handling for examination and repackaging, these fission gases would be released to the environment. Since the extent of fuel degradation would not be known in advance, for the purposes of this EIS the estimates of air emissions during handling operations are conservatively based on the radiological gaseous emissions provided in the Programmatic Spent Nuclear Fuel EIS and adjusted for the percentage of sodium-bonded fuel to that of the total spent nuclear fuel inventory at INEEL. Therefore, annual radiological gaseous emissions are estimated to be between 0 and 460 curies of tritium/carbon-14 and between 0 and 7,120 curies of krypton-85 (DOE 1995a). These estimates of air emissions are conservative and would bound any potential releases that may occur during handling operation, (these releases correspond to an inventory of about 10 percent of degraded driver spent nuclear fuels). The handling operation for repackaging is estimated to last about two years.

#### **4.2.2 Water Resources**

##### *Surface Water*

No surface water is used at ANL-W. Flood waters from the Big Lost River are not expected to reach the facilities at ANL-W, as shown in Figure 3-3.

##### *Nonradiological Liquid Effluent*

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Current operating and monitoring practices would continue for National Pollutant Discharge Elimination System (NPDES) stormwater and liquid effluent discharges associated with facilities at ANL-W.

##### *Radiological Liquid Effluent*

No radiological liquid effluents would be discharged to the surface water.

##### *Groundwater*

Under either option of this alternative, there would be some reduction in groundwater consumption for domestic uses if the number of workers at ANL-W were to decrease. The current water use at ANL-W is 188 million liters (49.6 million gallons per year) per year .

### *Nonradiological Liquid Effluent*

For either option of this alternative, no nonradiological liquid effluents or wastes would be discharged to groundwater.

### *Radiological Liquid Effluent*

For either option of this alternative, no radiological liquid effluents would be discharged to groundwater.

## **4.2.3 Socioeconomics**

Under either option of the No Action Alternative, there could be a reduction of approximately 350 workers at ANL-W if a treatment technology is not selected or the decision is delayed. If all of these workers were to leave the regional economic area, this could result in the loss of an additional 623 indirect jobs. The total potential loss of 973 represents a 1 percent decrease in the regional economic area civilian labor force, which was estimated to be 150,835 in 1996 (DOE 1998d).

Since any reduction in the ANL-W labor force under the No Action Alternative would take place over time, combined with the fact that many of these workers could also support missions at INEEL, the effects are expected to be gradual. By 2010, the contributory effect of this and the potential for beneficial effects from other industrial and economic sectors within the regional economic area would serve to reduce or mask any effect on the regional economy.

Both options of the No Action Alternative would therefore most likely not result in any noticeable change in the existing regional economy, population and housing characteristics, and community services within the region of influence at ANL-W (see Section 3.2.8). Overall expenditures and employment at INEEL should remain relatively constant through 2010, which would, in turn, tend to maintain economic and demographic characteristics within the region of influence.

## **4.2.4 Public and Occupational Health and Safety**

The assessments of potential radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in **Tables 4-2** and **4-3** for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in **Tables 4-4** and **4-5**. The impacts from hazardous chemical releases during accident conditions are presented in **Table 4-6**. Background information on the effects of radiation on human health and safety is presented in Section 4.1.3 and Appendix E, Section E.2.

### **4.2.4.1 Normal Operations**

#### *Radiological Impacts*

Under either option of this alternative, radioactive releases from normal operations associated with spent nuclear fuel storage activities at ANL-W and INTEC would be small. Annual radiation doses to the public from these activities were calculated for the No Action Alternative in the Programmatic Spent Nuclear Fuel EIS based on a total INEEL spent nuclear fuel inventory of approximately 274 metric tons of heavy metal. The inventory of sodium-bonded spent nuclear fuel in storage at ANL-W and INTEC represents about 60 metric tons of heavy metal, or about 22 percent of the INEEL inventory identified in the Programmatic Spent Nuclear Fuel EIS. For this SBSNF EIS, radiological impacts from normal operations associated with storage of sodium-bonded spent nuclear fuel are estimated to be about 22 percent of the impacts calculated for the No Action Alternative in the Programmatic Spent Nuclear Fuel EIS.

Calculated maximum annual radiological impacts to the public are given in Table 4–2. The impacts are calculated for two receptor groups: the general public living within 80 kilometers (50 miles) of INEEL in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table.

**Table 4–2 Annual Radiological Impacts to the Public From Operational Activities Associated With the No Action Alternative**

<i>Receptor</i>	<i>No Action Alternative</i>
<b>Population Dose Within 80 Kilometers (50 Miles)</b>	
Dose (person-rem) <sup>a</sup>	0.022
Latent cancer fatalities	0.000011
<b>Annual Dose to the Maximally Exposed Offsite Individual</b>	
Dose (millirem) <sup>a</sup>	0.00077
Latent cancer fatality risk	$3.9 \times 10^{-10}$
Percent of natural background <sup>b</sup>	0.00021
<b>Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)<sup>c</sup></b>	
Dose (millirem)	0.000092
Latent cancer fatality risk	$4.6 \times 10^{-11}$

<sup>a</sup> Based on 22 percent of the dose reported in Volume 1, Appendix B, Programmatic Spent Nuclear Fuel EIS (DOE 1995a).

<sup>b</sup> The annual natural background radiation level at INEEL is 360 millirem for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,520 person-rem. The site population in 2010 was assumed to be representative of the population over the operational period evaluated.

<sup>c</sup> Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Occupational doses were also estimated based on worker doses calculated for the No Action Alternative in the Programmatic Spent Nuclear Fuel EIS. The average worker dose (for ANL-W and INTEC workers) under the No Action Alternative was estimated to be similar to that currently experienced at ANL-W; see Table 4–3 (see Section 4.3.4).

**Table 4–3 Annual Radiological Impacts to Workers From Operational Activities Associated With the No Action Alternative**

<i>Impact</i>	<i>No Action Alternative</i>
<b>Worker<sup>a</sup></b>	
Average worker dose (millirem per year)	60
Latent cancer fatality risk	0.000024
Total dose (person-rem per year)	22
Latent cancer fatalities	0.0088

<sup>a</sup> The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to an involved worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year as established for all DOE activities in DOE Order N441.1.

Source: ANL 1999.

As shown in Tables 4-2 and 4-3:

- The annual dose to the maximally exposed offsite individual would be 0.00077 millirem per year, with an associated  $3.9 \times 10^{-10}$  risk per year of developing a fatal cancer (or one in 2.5 billion years).

- The collective dose to the population within 80 kilometers (50 miles) of the storage facilities at INEEL would be 0.022 person-rem per year, with an associated 0.000011 latent cancer fatalities per year (or one in 90,900 years).
- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one in 113 years).

### *Hazardous Chemical Impacts*

It is expected that the hazardous chemical impacts associated with either option of this alternative at ANL-W would be negligible, because any emissions of hazardous chemicals from activities under this alternative would be very low. The existing chemical environment is presented in Section 3.2.10.2.

#### **4.2.4.2 Facility Accidents**

The potential radiological impacts to the public and noninvolved onsite workers due to accidents are summarized in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, an earthquake, and an aircraft crash.

Under either option of the No Action Alternative, spent nuclear fuel transfer and waste processing activities associated with cleaning and stabilizing residual wastes generated during the electrometallurgic treatment demonstration project at ANL-W have the potential to involve accident scenarios similar to those evaluated for Alternative 1. However, the consequences associated with these accident scenarios are lower because of the limited quantities of residual wastes to be stabilized. Accidents associated with spent nuclear fuel transfer activities also could occur during the time when spent nuclear fuel is removed from the Radioactive Scrap and Waste Facility to prepare it for packaging and offsite shipment to a repository. It is estimated that spent nuclear fuel transfer and the waste stabilization activities would occur over a two-year time period.

During the time that sodium-bonded spent nuclear fuel is in dry storage at the Radioactive Scrap and Waste Facility, it is in a very safe and stable configuration, and no reasonably foreseeable accident scenarios could be identified. Sodium-bonded spent nuclear fuel currently in wet storage at INTEC would be transferred to dry storage facilities at INTEC. Handling accidents could occur during transfer activities at INTEC, similar to the accident scenario evaluated for ANL-W. However, because INTEC is further away from the INEEL site boundary and major population centers compared to ANL-W, the accident impacts at INTEC would be less than those for ANL-W.

Table 4–4 presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker. The dose to the maximally exposed offsite individual was calculated for the 95<sup>th</sup> percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50<sup>th</sup> percentile meteorological conditions. The 50<sup>th</sup> percentile condition represents the median meteorological condition, and is defined as that for which more severe conditions occur 50 percent of the time. The 95<sup>th</sup> percentile condition represents relatively low probability meteorological conditions that produce higher calculated exposures, and is defined as that condition that is not exceeded more than 5 percent of the time. DOE did not quantitatively estimate the involved worker dose due to accidents. The consequences to involved workers are qualitatively assessed. This approach is used for two reasons: first, no adequate method exists for calculating meaningful consequences at or near the location where the accident occurs. Second, safety assurance for facility workers is demonstrated by both the workers' training and by the establishment of an Occupational Safety and Health Administration (OSHA) process safety management system (29 CFR 1910.119), the evaluations required by such a system, and the products derived from such evaluations

(e.g., procedures, programs, emergency plans). In any accident scenario, the individuals most likely to be injured are the involved workers. The risk to these workers would be due to both radiological and nonradiological effects. In a fire, the involved workers could be exposed to airborne radioactive material, in addition to the smoke and heat of the fire. In an explosion, there could be flying debris and containment barriers could be broken, exposing workers to airborne radioactive material. Most spills would not have a major effect on involved workers because they would clean up the spill wearing protective clothing and respirators as necessary. An accidental criticality could expose involved workers to large doses of prompt penetrating radiation, which could cause death in a short period of time. An earthquake accident presents very severe nonradiological effects to the involved workers. In such a scenario, the workers are likely to be hurt or killed from the collapse of the building before they could be evacuated. (See Appendix F, Section F.2.2.2, for more detail.) The accident risks for the same receptors are summarized in Table 4–5.

**Table 4–4 Accident Frequency and Consequences for the No Action Alternative**

Accident <sup>a</sup>	Frequency (event per year)	Maximally Exposed Offsite Individual		Population within 80 kilometers (50 miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality <sup>b</sup>	Dose (person- rem)	Latent Cancer Fatalities <sup>c</sup>	Dose (millirem)	Latent Cancer Fatality <sup>b</sup>
Salt powder spill in the Hot Fuel Examination Facility cell	0.01	0.00046	$2.3 \times 10^{-10}$	0.000098	$4.9 \times 10^{-8}$	$4.7 \times 10^{-7}$	$1.9 \times 10^{-13}$
Cask drop during spent nuclear fuel transfer	0.01	0.03	$1.5 \times 10^{-8}$	0.0035	$1.7 \times 10^{-6}$	0.00084	$3.4 \times 10^{-10}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.5 \times 10^{-6}$	0.22	$8.7 \times 10^{-8}$
Design-basis seismic event	0.008	12	$6.0 \times 10^{-6}$	1.4	0.00070	4.7	$1.9 \times 10^{-6}$
Salt spill during transfer	$1 \times 10^{-7}$	0.19	$9.5 \times 10^{-8}$	0.022	0.000011	0.073	$2.9 \times 10^{-8}$
Beyond-design-basis seismic event	0.00001	96	0.000048	11	0.0055	37	0.000015

<sup>a</sup> Only accidents involving EBR-II driver spent nuclear fuel, which maximizes the consequences, are presented.

<sup>b</sup> Increased likelihood of a latent cancer fatality.

<sup>c</sup> Increased number of latent cancer fatalities.

**Table 4–5 Annual Cancer Risks Due to Accidents for the No Action Alternative**

Accident	Maximally Exposed Offsite Individual <sup>a</sup>	Population within 80 kilometers (50 miles) <sup>b</sup>	Noninvolved Worker <sup>a</sup>
Salt powder spill in Hot Fuel Examination Facility cell	$2.3 \times 10^{-12}$	$4.9 \times 10^{-10}$	$1.9 \times 10^{-15}$
Cask drop during spent nuclear fuel transfer	$1.5 \times 10^{-10}$	$1.7 \times 10^{-8}$	$3.4 \times 10^{-12}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.5 \times 10^{-9}$	$8.7 \times 10^{-11}$
Design-basis seismic event	$4.8 \times 10^{-8}$	$5.6 \times 10^{-6}$	$1.5 \times 10^{-8}$
Salt spill during transfer	$9.5 \times 10^{-15}$	$1.1 \times 10^{-12}$	$2.9 \times 10^{-15}$
Beyond-design-basis seismic event	$4.8 \times 10^{-10}$	$5.5 \times 10^{-8}$	$1.5 \times 10^{-10}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

For the accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be  $4.8 \times 10^{-8}$  per year (or one in 20.8 million years) and  $1.5 \times 10^{-8}$  per year (or one in 66.7 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be  $5.6 \times 10^{-6}$  per year (or one in 178,600 years).

*Hazardous Chemical Impacts*

Nonradiological hazardous chemical impacts are evaluated in terms of comparison to Emergency Response Planning Guidelines. Emergency Response Planning Guideline values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The nonradiological (hazardous chemical) impacts of potential facility accidents associated with either option of the No Action Alternative are summarized in Table 4–6.

**Table 4–6 Hazardous Chemical Accident Impacts for the No Action Alternative**

<i>Accident</i>	<i>Frequency (event per year)</i>	<i>Receptor Location</i>	<i>Exposure</i>
Uranium-handling accident	0.01	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker at 100 meters	Cadmium: less than ERPG-1 Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1 Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline

**4.2.5 Environmental Justice**

As discussed in Section 4.2.4, operations conducted under either option of this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed individual over the 35 years’ duration of interim storage operation and removal from the INEEL site (which is assumed to occur by 2035) would be 0.000014 (or 1 chance in 71,400), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.00039 (or 1 chance in 2,560). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small regardless of the racial and ethnic composition of the population and independent of the economic status of individuals comprising the population. Operation of spent nuclear fuel storage facilities at ANL-W and INTEC would have no disproportionately high and adverse effects on minority or low-income populations.

**4.2.6 Waste Management**

Various types of waste would be generated associated with sodium-bonded spent nuclear fuel storage activities at ANL-W and INTEC, including transuranic waste, mixed transuranic waste, low-level radioactive waste, mixed waste, hazardous, and nonhazardous wastes. In addition, during the first two years of operation under either option of this alternative, ANL-W would continue to generate high-level radioactive waste as the Electrometallurgical Treatment Demonstration Project cladding hull waste and electrorefiner salt are stabilized to metallic and ceramic high-level radioactive waste forms for ultimate disposal. **Table 4–7** shows the

anticipated categorization of these waste types and their expected interim storage and final disposal locations. The quantities of ceramic and metal waste forms generated, along with other generated wastes, are presented in **Table 4–8**.

**Table 4–7 Summary of Process Waste Material Categories for the No Action Alternative**

<i>Waste Stream</i>	<i>Category</i>	<i>Interim Storage Location</i>	<i>Final Disposal Location</i>
<b>Process Wastes</b>			
Fuel hardware	Low-level radioactive waste	None	Radioactive Waste Management Complex
Metal waste form	High-level radioactive waste	Radioactive Scrap and Waste Facility	Offsite (proposed geologic) repository
Ceramic waste form	High-level radioactive waste	Radioactive Scrap and Waste Facility	Offsite (proposed geologic) repository
<b>Other Associated Process Wastes</b>			
less than 100 nanocuries per gram transuranic waste <sup>a</sup>	Low-level radioactive waste	None	Radioactive Waste Management Complex
greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Radioactive Waste Management Complex	Waste Isolation Pilot Plant
Cadmium-contaminated	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant or Radioactive Waste Management Complex after treatment
Nonradioactive	Sanitary waste	None	INEEL landfill
<b>Deactivation Wastes</b>			
Electrorefiner cadmium	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant
Equipment less than 100 nanocuries per gram transuranic waste <sup>a</sup>	Low-level radioactive waste	None	Radioactive Waste Management Complex
Equipment greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Radioactive Waste Management Complex	Waste Isolation Pilot Plant
Cadmium-contaminated	Mixed waste	Radioactive Scrap and Waste Facility	Waste Isolation Pilot Plant or Radioactive Waste Management Complex

<sup>a</sup> “As noted in Section 3.2.11.3, the Radioactive Waste Management Complex cannot be used for the disposal of the alpha low-level radioactive waste (between 10 and 100 nanocuries per gram). Wastes in this category may be treated by the Advanced Mixed Waste Treatment Project and then disposed of at the Waste Isolation Pilot Plant.”

*Direct Process Wastes*

Under either option of the No Action Alternative, small amounts of metal and ceramic high-level radioactive waste would be produced at ANL-W as a result of the completion of the Electrometallurgical Treatment Demonstration Project. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metal waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. Both the ceramic and metal waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–8 are for the standardized canisters required for disposal of these materials.

**Table 4–8 Amount of Wastes Generated for the No Action Alternative<sup>a</sup>**

Waste Stream	Total Waste Generated	
	Volume (Cubic Meters)	Mass (Kilograms)
<b>Direct Process Wastes</b>		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	9.4	14,000
High-level radioactive metal waste	0.6	460
Spent nuclear fuel	92/142 <sup>b</sup>	72,000
<b>Other Associated Process Wastes</b>		
Low-level radioactive wastes	700	142,000
Transuranic wastes	8.4	3,000
Mixed wastes	35	19,000
Sanitary wastes	2,500	867,000
<b>Deactivation Wastes</b>		
Low-level radioactive wastes	112	38,000
Transuranic wastes	1.6	853
Mixed wastes	3	2,100

<sup>a</sup> These waste generation estimates are through the year 2035. This is the date by which materials of this type are required to be out of the State of Idaho.

<sup>b</sup> Volumes for interim storage/direct disposal.

Source: ANL 1999.

The metal and ceramic high-level radioactive waste generated as a result of the electrometallurgical treatment demonstration at ANL-W would be stored temporarily for 10 to 15 years at the Radioactive Scrap and Waste Facility at ANL-W in a manner that allows retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste, and shielding will be provided by a combination of steel storage liners in which the waste would be stored, and by the soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste cans containing the metal and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

#### *Other Associated Process Low-Level Radioactive Wastes*

Other associated process low-level radioactive wastes would be generated as a result of the deactivation and conversion of demonstration high-level radioactive waste into suitable forms for the repository, as well as from other ongoing activities, including keeping a hot cell facility operational to handle unforeseen problems while storing the fuel in the Radioactive Scrap and Waste Facility. These wastes are the result of activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from either option of the No Action activities at ANL-W that would require disposal (after volume reduction) would be a maximum of about 50 cubic meters (1,766 cubic feet) per year, and most years would result in approximately 17 cubic meters (600 cubic feet). This maximum volume represents a small fraction (approximately 1 percent) of the total annual volume of low-level radioactive waste currently being disposed of at the Radioactive Waste Management Complex, and the total of 700 cubic meters (24,700 cubic feet) represents approximately 0.6 percent of the total Radioactive Waste Management Complex disposal inventory.

### *Other Associated Process Transuranic Wastes*

Other associated process transuranic wastes would be generated at ANL-W under either option of the No Action Alternative from decontamination activities, repair and maintenance of items, and miscellaneous work associated with demonstration fuel processing or other activities. Transuranic wastes would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

For the No Action Alternative, the volume of transuranic waste generated at ANL-W would amount to a maximum of approximately 1 cubic meter per year (35 cubic feet per year), and most years would result in approximately 0.2 cubic meters (7 cubic feet). This maximum volume is approximately 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex. The total volume of incidental transuranic waste generated under the No Action Alternative is approximately 8.4 cubic meters (300 cubic feet), which is 0.005 percent of the estimated total volume of transuranic waste to be emplaced at the Waste Isolation Pilot Plant.

### *Other Associated Process Sanitary Wastes*

These sanitary wastes that are nonradioactive and nonhazardous would continue to be generated under either option of the No Action Alternative. These solid wastes would be typical of industrial operations and would be disposed of at the INEEL landfill. Based on an estimated eventual INEEL landfill volume of  $3 \times 10^6$  cubic meters (106 million cubic feet), the total volume of solid sanitary waste generated and disposed of under this alternative is approximately 0.1 percent of the INEEL landfill volume.

### *Other Associated Process Mixed Wastes*

These mixed waste would be generated primarily from the disposal of any cadmium-contaminated equipment or clean-up material and the analysis of cadmium samples. At ANL-W, mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated at ANL-W have been identified in the INEEL Site Treatment Plan (DOE 1996b).

### *Deactivation Wastes*

A variety of wastes would be generated as part of deactivation activities at ANL-W. These would include process equipment and process material such as electrorefiner cadmium. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation wastes under either option of the No Action Alternative would be low-level radioactive waste, generated as a result of equipment dismantling and disposal. Components that would require disposal include the existing electrorefiner and hot isostatic press, as well as other processing components. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. Under the No Action Alternative, it is anticipated that the deactivation waste volumes would be generated over a period of one year. The total deactivation wastes represent approximately 17 percent over the total incidental waste (excluding sanitary wastes) requiring disposal.

Once the residual high-level radioactive wastes from the Electrometallurgical Treatment Demonstration Project have been stabilized, which is estimated to require about two years, the types and quantities of wastes generated under the No Action Alternative would be consistent with current activities at INEEL, as presented in Section 3.2.11.1.

### **4.3 ALTERNATIVE 1: ELECTROMETALLURGICAL TREATMENT OF BLANKET AND DRIVER FUEL AT ANL-W**

Under this alternative, the sodium-bonded spent nuclear fuel would be treated at ANL-W using the electrometallurgical process, described in Appendix C. The various process steps in this technology are performed at the Fuel Conditioning Facility and the Hot Fuel Examination Facility hot (air or argon) cells. The processes at the Fuel Conditioning Facility include: fuel chopping, electrorefining, cathode processing, and metal casting (see Appendix C for details on each processing step). These processes would separate the uranium from the fission products. Separated uranium is not considered a waste. The separated uranium would be made into a low-enriched uranium ingot, and the metallic sodium would be oxidized in the electrorefiner lithium-potassium salt and removed along with the fission products as high-level radioactive waste. The salts from the electrorefiner then would be solidified and sent to the Hot Fuel Examination Facility for further processing. The processes at the Hot Fuel Examination Facility include waste treatment, metal melting, and high-level radioactive waste production. These processes would produce two waste forms—a ceramic waste form consisting of fission products and transuranic elements including plutonium elements, and a metal waste form consisting of noble metal fission products and cladding hulls from the spent nuclear fuel. The low-enriched uranium metal ingot would be stored at the Zero Power Physics Reactor Material Storage Building. The ceramic and metal waste forms would be temporarily stored at the Radioactive Scrap and Waste Facility pending packaging for disposition in a geologic repository.

The electrometallurgical process at ANL-W facilities would treat about 5 metric tons of heavy metal of sodium-bonded spent nuclear fuel per year. Appendix E, Section E.4.1, provides details on the process duration and the amount of blanket and driver spent nuclear fuel treated annually. The treatment of blanket and driver spent nuclear fuel under this alternative could start as early as 2000 and could be completed by 2012.

#### **4.3.1 Air Quality**

##### *Nonradiological Gaseous Emissions*

It is expected that this alternative at ANL-W will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

##### *Radiological Gaseous Emissions*

Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that are released to the argon cell at the Fuel Conditioning Facility during fuel element chopping and electrorefining processes. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries ( $10^9$  curies) and are released to the environment through the facility stack, along with krypton and elemental tritium. The maximum release of radioactive gaseous emissions occurs during the first five years of the electrometallurgical treatment process, where a combination of EBR-II blanket and driver spent nuclear fuel elements are processed. During these five years, about 600 kilograms of heavy metal driver spent nuclear fuel and about 4,400 kilograms of heavy metal blanket spent nuclear fuel would be processed annually. The

combined process would release about 11,600 curies of krypton-85 and 770 curies of elemental tritium annually; see Appendix E.4.1 for details. The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.3.4.

#### **4.3.2 Water Resources**

##### *Surface Water*

No surface water is used at ANL-W. Flood waters from the Big Lost River are not expected to reach the facilities at ANL-W, as shown in Figure 3-3.

##### *Nonradiological Liquid Effluent*

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with the electrometallurgical treatment processes. Current operating and monitoring practices would continue for NPDES stormwater and liquid effluent discharges associated with facilities at ANL-W.

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996a).

##### *Radiological Liquid Effluent*

No radiological liquid effluent or waste generated by the electrometallurgical treatment process would be discharged to surface water.

##### *Groundwater*

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. The current water use at ANL-W is 188 million liters (49.6 million gallons) per year.

##### *Nonradiological Liquid Effluent*

No nonradiological liquid effluent generated by the electrometallurgical treatment process would be discharged to groundwater.

##### *Radiological Liquid Effluent*

No radiological liquid effluent generated by the electrometallurgical treatment process would be discharged to groundwater.

### 4.3.3 Socioeconomics

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

### 4.3.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in **Tables 4–9** and **4–10** for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in **Tables 4–11** and **4–12**. The impacts from hazardous chemical releases during accident conditions are presented in **Table 4–13**. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

#### 4.3.4.1 Normal Operations

##### *Radiological Impacts*

Under this alternative, radioactive releases would occur during fuel chopping and from the operation of electrorefiners. Both of these activities are performed in the Fuel Conditioning Facility argon cell. Appendix E, Sections E.3 and E.4.1, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result from treating 0.6 metric tons of heavy metal of EBR-II driver spent nuclear fuel and 4.4 metric tons of heavy metal of EBR-II blanket spent nuclear fuel. This combination of fuel treatment would continue for six years, after which only Fermi-1 blanket spent nuclear fuel would be treated. Overall, it would require 13 years to treat all the sodium-bonded fuel (see Appendix E, Section E.4.1 for details).

Calculated maximum annual radiological impacts to the public are given in Table 4–9. The impacts are calculated for two receptor groups: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public are releases of tritium gases (about 1 percent of which conservatively was assumed to be in oxidized form) and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table.

Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with the electrometallurgical treatment of sodium-bonded spent nuclear fuel. The estimated annual total worker population dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 13 years of electrometallurgical treatment activities, the cumulative worker population dose would be 286 person-rem, leading to a risk of 0.11 latent cancer fatalities (see Table 4–10).

**Table 4–9 Annual Radiological Impacts to the Public From Operational Activities Associated With Alternative 1**

<i>Receptor</i>	<i>Alternative 1</i>		
	<i>Electrometallurgical Treatment of Driver Spent Nuclear Fuel</i>	<i>Electrometallurgical Treatment of Blanket Spent Nuclear Fuel</i>	<i>Total</i>
<b>Population Dose Within 80 Kilometers (50 Miles) in the Year 2010</b>			
Dose (person-rem)	0.0028	0.000084	0.0029
Latent cancer fatalities	$1.4 \times 10^{-6}$	$4.2 \times 10^{-8}$	$1.5 \times 10^{-6}$
<b>Annual Dose to the Maximally Exposed Offsite Individual</b>			
Dose (millirem)	0.00033	0.00001	0.00034
Latent cancer fatality risk	$1.6 \times 10^{-10}$	$5.0 \times 10^{-12}$	$1.7 \times 10^{-10}$
Percent of natural background <sup>a</sup>	0.000092	$2.7 \times 10^{-6}$	0.000094
<b>Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)<sup>b</sup></b>			
Dose (millirem)	0.000012	$3.5 \times 10^{-7}$	0.000012
Latent cancer fatality risk	$6.0 \times 10^{-12}$	$1.8 \times 10^{-13}$	$6.0 \times 10^{-12}$

<sup>a</sup> The annual natural background radiation level at INEEL is about 360 millirem for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.

<sup>b</sup> Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of the ANL-W in the year 2010 (240,338).

**Table 4–10 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 1**

<i>Impact</i>	<i>Alternative 1</i>
<b>Worker<sup>a</sup></b>	
Total dose (person-rem per year)	22
13-year fatal cancer risk	0.11
Average worker dose (millirem per year)	60
13-year fatal cancer risk	0.00031

<sup>a</sup> The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to an involved worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year as established for all DOE activities in DOE Order N441.1.

Source: ANL 1999.

As shown in Tables 4–9 and 4–10:

- The annual dose to the maximally exposed offsite individual would be 0.00034 millirem per year, with an associated  $1.7 \times 10^{-10}$  risk per year of developing a fatal cancer (or one in 5.9 billion years).
- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0029 person-rem per year, with an associated  $1.5 \times 10^{-6}$  latent cancer fatalities per year (or one in 667,000 years).
- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one in 113 years).

*Hazardous Chemical Impacts*

It is expected that hazardous chemical impacts associated with this alternative will be negligible, as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described in Section 3.2.10.2.

**4.3.4.2 Facility Accidents**

*Radiological Impacts*

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during electrometallurgical treatment operational activities are summarized and presented in this section. Since electrometallurgical treatment processes are performed in both the Fuel Conditioning Facility and the Hot Fuel Examination Facility, accidents at both facilities are considered. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios including fire, spills, criticality, earthquake, and aircraft crash. Aircraft crash and criticality accidents were determined to have an accident frequency of less than  $10^{-7}$  per year, and were not analyzed further. Table 4–11 presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for 95<sup>th</sup> percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50<sup>th</sup> percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents. (See discussions on the involved worker in Section 4.2.4.2.) The accident risks for the same receptors are summarized in Table 4–12.

**Table 4–11 Accident Frequency and Consequences for Alternative 1**

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population within 80 kilometers (50 miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>	Dose (person- rem)	Latent Cancer Fatalities <sup>b</sup>	Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>
<b>Driver Spent Nuclear Fuel</b>							
Salt powder spill	0.01	0.00046	$2.3 \times 10^{-10}$	0.000098	$4.9 \times 10^{-8}$	$4.7 \times 10^{-7}$	$1.9 \times 10^{-13}$
Salt transfer drop	$1.0 \times 10^{-7}$	0.19	$9.5 \times 10^{-8}$	0.022	0.000011	0.073	$2.9 \times 10^{-8}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Cask drop	0.01	0.030	$1.5 \times 10^{-8}$	0.0035	$1.8 \times 10^{-6}$	0.00084	$3.4 \times 10^{-10}$
Earthquake (design-basis earthquake)	0.008	12	$6.0 \times 10^{-6}$	1.4	0.0007	4.7	$1.9 \times 10^{-6}$
Earthquake (beyond-design-basis earthquake)	0.00001	22,000	0.022	2,500	1.3	370	0.00015
<b>Blanket Spent Nuclear Fuel</b>							
Salt powder spill	0.01	0.00012	$6.2 \times 10^{-11}$	0.000027	$1.3 \times 10^{-8}$	$1.1 \times 10^{-6}$	$4.4 \times 10^{-13}$
Salt transfer drop	$1 \times 10^{-7}$	0.052	$2.6 \times 10^{-8}$	0.0062	$3.1 \times 10^{-6}$	$0.17 \times 10^{-3}$	$6.8 \times 10^{-8}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Cask drop	0.01	0.0024	$1.2 \times 10^{-9}$	0.00028	$1.4 \times 10^{-7}$	0.000049	$2.0 \times 10^{-11}$

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population within 80 kilometers (50 miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>	Dose (person- rem)	Latent Cancer Fatalities <sup>b</sup>	Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>
Earthquake (design-basis earthquake at Hot Fuel Examination Facility)	0.008	3.3	$1.6 \times 10^{-6}$	0.4	0.0002	11	$4.4 \times 10^{-6}$
Earthquake (beyond-design-basis earthquake)	0.00001	0.071	0.00035	83	0.041	38	0.000019

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

**Table 4-12 Annual Cancer Risks Due to Accidents for Alternative 1**

Accident	Maximally Exposed Offsite Individual <sup>a</sup>	Population within 80 kilometers (50 miles) <sup>b</sup>	Noninvolved Worker <sup>a</sup>
<b>Driver Spent Nuclear Fuel</b>			
Salt powder spill	$2.3 \times 10^{-12}$	$4.9 \times 10^{-10}$	$1.9 \times 10^{-15}$
Salt transfer drop	$9.5 \times 10^{-15}$	$1.1 \times 10^{-12}$	$2.9 \times 10^{-15}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.6 \times 10^{-9}$	$8.8 \times 10^{-11}$
Cask drop	$1.5 \times 10^{-10}$	$1.8 \times 10^{-8}$	$3.4 \times 10^{-12}$
Earthquake (design-basis earthquake)	$4.8 \times 10^{-8}$	$5.6 \times 10^{-6}$	$1.5 \times 10^{-8}$
Earthquake (beyond-design-basis earthquake)	$2.2 \times 10^{-7}$	0.000013	$1.5 \times 10^{-9}$
<b>Blanket Spent Nuclear Fuel</b>			
Salt powder spill	$6.2 \times 10^{-13}$	$1.3 \times 10^{-10}$	$4.4 \times 10^{-15}$
Salt transfer drop	$2.6 \times 10^{-15}$	$3.1 \times 10^{-13}$	$6.8 \times 10^{-15}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.5 \times 10^{-9}$	$8.7 \times 10^{-11}$
Cask drop	$1.2 \times 10^{-11}$	$1.4 \times 10^{-9}$	$2.0 \times 10^{-13}$
Earthquake (design-basis earthquake)	$1.3 \times 10^{-8}$	$1.6 \times 10^{-6}$	$3.5 \times 10^{-8}$
Earthquake (beyond-design-basis earthquake)	$3.5 \times 10^{-9}$	$4.1 \times 10^{-7}$	$1.7 \times 10^{-9}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

For the accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be  $2.2 \times 10^{-7}$  per year (or one in 4.5 million years) and  $3.5 \times 10^{-8}$  per year (or one in 28.6 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one in 76,920 years).

### Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to Emergency Response Planning Guidelines. Emergency Response Planning Guideline values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The nonradiological impacts of potential facility accidents associated with the electrometallurgical treatment alternative at ANL-W are summarized in Table 4–13.

**Table 4–13 Hazardous Chemical Accident Impacts for Alternative 1**

<i>Accident</i>	<i>Frequency (event per year)</i>	<i>Receptor Location</i>	<i>Exposure</i>
Uranium-handling accident	0.01	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker at 100 meters	Cadmium: less ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline.

#### 4.3.5 Environmental Justice

As discussed in Section 4.3.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the 13 years of electrometallurgical treatment operation would be  $2.2 \times 10^{-9}$  (or one chance in 454 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000020 (or one chance in 50,000). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small regardless of the racial and ethnic composition of the population, and independent of the economic status of individuals comprising the population. Operation of electrometallurgical processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

#### 4.3.6 Waste Management

Electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W would generate process wastes from treatment operations, other associated process wastes from normal support operations and deactivation wastes following the conclusion of operations. Process wastes would include fuel hardware and metal and ceramic high-level radioactive wastes. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of waste types and their expected interim storage and final disposal locations are given in Table 4–7 (see Section 4.2.6).

Estimates of the total amount of other associated process waste generated as a result of electrometallurgical treatment at ANL-W are provided in **Table 4–14**. These values are based on an evaluation of waste forecasts from ANL-W that account only for the fraction of total ANL-W waste that would be attributable to the processing of sodium-bonded spent nuclear fuel under this alternative. The values in Table 4–14 are for

disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of electrometallurgical treatment could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

**Table 4-14 Amounts of Wastes Generated for Alternative 1<sup>a</sup>**

Waste Stream	Total Waste Generated	
	Volume (Cubic Meters)	Mass (Kilograms)
<b>Direct Process Wastes</b>		
Fuel hardware (low-level radioactive waste)	12	6,600
High-level radioactive ceramic waste	78	120,000
High-level radioactive metal waste	6.3	9,000
<b>Other Associated Process Wastes</b>		
Low-level radioactive wastes <sup>b</sup>	706	143,000
Transuranic wastes	12.5	5,400
Mixed wastes	35.3	19,000
Sanitary wastes	4,960	$1.72 \times 10^6$
<b>Deactivation Wastes</b>		
Low-level radioactive wastes <sup>b</sup>	143	48,000
Transuranic wastes	1.6	853
Mixed wastes	4.2	2,900

<sup>a</sup> These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage are accomplished during this time period.

<sup>b</sup> The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

*Direct Process Wastes*

For electrometallurgical treatment, fuel assembly hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures.

Under Alternative 1, metal and ceramic high-level radioactive waste would be a primary product. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metal waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metal waste form. Both the ceramic and metal waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4-14 are for the standardized canisters required for disposal of these materials.

The metal and ceramic high-level radioactive waste generated would be temporarily stored for 10 to 15 years at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste, and shielding would be provided by a combination of steel storage liners storing the waste, and the shielding provided by soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste

cans containing the metal and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

#### *Other Associated Process Low-Level Radioactive Wastes*

These low-level radioactive wastes would be generated as a result of electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from electrometallurgical treatment at ANL-W that will require disposal (after volume reduction) would be approximately 48 cubic meters (1,695 cubic feet) per year. This represents approximately 0.08 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 706 cubic meters (24,932 cubic feet) represents approximately 0.9 percent of the total Radioactive Waste Management Complex disposal capacity.

#### *Other Associated Process Transuranic Wastes*

These transuranic wastes would be generated by decontamination activities, repair and maintenance of items, and miscellaneous work associated with the electrometallurgical processing. Transuranic wastes would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is approximately 13 cubic meters (459 cubic feet), which is less than 0.008 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

#### *Other Associated Process Mixed Wastes*

Mixed waste of this category would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1996b).

#### *Deactivation Wastes*

A variety of wastes would be generated as part of deactivation activities associated with electrometallurgical treatment processing at ANL-W. These would include process equipment and process material, such as electrorefiner cadmium. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation wastes would be low-level radioactive waste, transuranic waste, and mixed waste generated as a result of equipment dismantling and disposal. Components that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. If the deactivation waste volume is generated in a single year, the wastes would represent an increase of approximately 3.5 times the annual waste generated by electrometallurgical treatment requiring disposal. The total deactivation wastes represent approximately 30 percent over the total incidental waste (excluding sanitary wastes) requiring disposal.

#### **4.4 ALTERNATIVE 2: PACKAGE BLANKET FUEL IN HIGH-INTEGRITY CANS AND TREAT (ELECTROMETALLURGICAL TREATMENT) DRIVER FUEL AT ANL-W**

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be cleaned to remove metallic sodium and placed in high-integrity cans. These cans then would be placed into overpack containers prior to dry storage at the Radioactive Scrap and Waste Facility, pending repackaging and transportation for disposal in a geologic repository. The removed sodium contains radioactive elements, principally cesium. The cesium would be separated from the sodium and stabilized as ceramic waste. The sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuels would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its placement in high-integrity cans could start in 2003 and be completed by 2009.

##### **4.4.1 Air Quality**

###### *Nonradiological Gaseous Emissions*

It is expected that this alternative at ANL will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

###### *Radiological Gaseous Emissions*

The cleaning of the blanket spent nuclear fuel to remove metallic sodium and the electrometallurgical treatment of the driver spent nuclear fuel would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries ( $10^{-9}$  curies) and are released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions occurs when cleaning blanket spent nuclear fuel for placement in high-integrity cans and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. This simultaneous operation was estimated to occur over a three-year period starting in 2003. Based on an annual cleaning throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and an electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere. The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.4.4.

#### **4.4.2 Water Resources**

##### *Surface Water*

No surface water is used at ANL-W. Flood waters from the Big Lost River are not expected to reach the facilities at ANL-W, as shown in Figure 3-3.

##### *Nonradiological Liquid Effluent*

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with high-integrity can operations and electrometallurgical treatment process operations. Current operating and monitoring practices would continue for NPDES stormwater and liquid effluent discharges associated with facilities at ANL-W.

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996a).

##### *Radiological Liquid Effluent*

No radiological liquid effluent or waste generated by high-integrity can and electrometallurgical treatment process operations would be discharged to surface water.

##### *Groundwater*

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. The current water usage at ANL-W is 188 million liters (49.6 million gallons) per year.

##### *Nonradiological Liquid Effluent*

No nonradiological liquid effluent generated by high-integrity can and electrometallurgical treatment process operations would be discharged to groundwater.

##### *Radiological Liquid Effluent*

No radiological liquid effluent or waste generated by high-integrity can and electrometallurgical treatment process operations would be discharged to groundwater.

#### **4.4.3 Socioeconomics**

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

#### 4.4.4 Public and Occupational Health and Safety

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in **Tables 4–15** and **4–16** for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in **Tables 4–17** and **4–18**. The impacts from hazardous chemical releases during accident conditions are presented in **Table 4–19**. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

##### 4.4.4.1 Normal Operations

###### *Radiological Impacts*

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel cleaning and driver spent nuclear fuel chopping. All of these activities are performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when cleaning of blanket spent nuclear fuels and treatment of driver spent nuclear fuels are performed simultaneously under this alternative. Appendix E, Section E.4.2, provides details on the treatment process duration and throughputs for each fuel type. The duration of the treatment process is estimated to be nine years.

Calculated maximum annual radiological impacts to the public are given in Table 4–15. The impacts are calculated for two receptor groups: the general public living within 80 kilometers (50 miles) of ANL-W in 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public are releases of tritium gases (about 1 percent of which was conservatively assumed to be in oxidized form) and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table.

**Table 4–15 Annual Radiological Impacts to the Public From Operational Activities Associated With Alternative 2**

<i>Receptor</i>	<i>Alternative 2</i>		
	<i>Electrometallurgical Treatment of Driver Spent Nuclear Fuel</i>	<i>Clean and Place Blanket Spent Nuclear Fuel in High-Integrity Cans</i>	<i>Total</i>
<b>Population Dose Within 80 Kilometers (50 Miles) in the Year 2010</b>			
Dose (person-rem)	0.0028	0.00028	0.0031
Latent cancer fatalities	$1.4 \times 10^{-6}$	$1.4 \times 10^{-7}$	$1.6 \times 10^{-6}$
<b>Annual Dose to the Maximally Exposed Offsite Individual</b>			
Dose (millirem)	0.00033	0.000048	0.00038
Latent cancer fatality risk	$1.6 \times 10^{-10}$	$2.4 \times 10^{-11}$	$1.9 \times 10^{-10}$
Percent of natural background <sup>a</sup>	0.000092	0.000013	0.00011
<b>Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)<sup>b</sup></b>			
Dose (millirem)	0.000012	$1.2 \times 10^{-6}$	0.000013
Latent cancer fatality risk	$6.0 \times 10^{-12}$	$6.0 \times 10^{-13}$	$6.6 \times 10^{-12}$

<sup>a</sup> The annual natural background radiation level at INEEL is 360 millirem for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.

<sup>b</sup> Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of the ANL-W in the year 2010 (240,338).

Table 4–16 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with sodium-bonded spent nuclear fuel high-integrity can and electrometallurgical treatment processes. It was concluded that the average worker dose would not be different from that currently being experienced. The estimated annual total worker population dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the nine years of treatment activities, the cumulative worker population dose would be 198 person-rem, leading to a risk of 0.079 latent cancer fatalities.

**Table 4–16 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 2**

<i>Impact</i>	<i>Alternative 2</i>
Worker <sup>a</sup>	
Total dose (person-rem per year)	22
Nine-year fatal cancer risk	0.079
Average worker dose (millirem per year)	60
Nine-year fatal cancer risk	0.00022

<sup>a</sup> The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to an involved worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year as established for all DOE activities in DOE Order N441.1.

Source: ANL 1999.

As shown in Tables 4–15 and 4–16:

- The annual dose to the maximally exposed offsite individual would be 0.00038 millirem per year, with an associated  $1.9 \times 10^{-10}$  risk per year of developing fatal cancer (or one in 5.3 billion years).
- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0031 person-rem per year, with an associated  $1.6 \times 10^{-6}$  latent cancer fatalities per year (or one in 625,000 years).
- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one in 113 years).

*Hazardous Chemical Impacts*

It is expected that hazardous chemical impacts associated with this alternative at ANL-W will be small, as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described in Section 3.2.12.2.

**4.4.4.2 Facility Accidents**

*Radiological Impacts*

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during cleaning activities for placement of blanket spent nuclear fuel elements in high-integrity cans and the electrometallurgical treatment operational activities for driver spent nuclear fuels are summarized and presented in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F.

The detailed analysis considered a wide spectrum of potential accident scenarios including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than  $10^{-7}$  per year, and consequence analyses for these two events were not performed. Cleaning of the blanket spent nuclear fuel is performed in the Hot Fuel Examination Facility; treatment of the driver spent nuclear fuel is performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing takes place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event. The cleaning of the blanket spent nuclear fuel is performed only in the Hot Fuel Examination Facility. The multifacility impacts of the beyond-design-basis earthquake are not relevant to the blanket spent nuclear fuel. Therefore, the higher frequency design-basis seismic event was analyzed for blanket spent nuclear fuels only. Table 4–17 presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility.

The dose to the maximally exposed offsite individual was calculated for the 95<sup>th</sup> percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50<sup>th</sup> percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents. (See the discussion on the involved worker in Section 4.2.4.2.) The accident risks for the same receptors are summarized in Table 4–18.

**Table 4–17 Accident Frequency and Consequences at ANL-W for Alternative 2**

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>	Dose (person- rem)	Latent Cancer Fatalities <sup>b</sup>	Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>
<b>Driver Spent Nuclear Fuel</b>							
Salt powder spill	0.01	0.00046	$2.3 \times 10^{-10}$	0.000098	$4.9 \times 10^{-8}$	$4.7 \times 10^{-7}$	$1.9 \times 10^{-13}$
Salt transfer drop	$1.0 \times 10^{-7}$	0.19	$9.5 \times 10^{-8}$	0.022	0.000011	0.073	$2.9 \times 10^{-8}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Cask drop	0.01	0.030	$1.5 \times 10^{-8}$	0.0035	$1.8 \times 10^{-6}$	0.00084	$3.4 \times 10^{-10}$
Earthquake (design-basis earthquake at Hot Fuel Examination Facility)	0.008	12	$6.0 \times 10^{-6}$	1.4	0.0007	4.7	$1.9 \times 10^{-6}$
Earthquake (beyond-design-basis earthquake)	0.00001	22,000	0.022	2,500	1.3	370	0.00015
<b>Blanket Spent Nuclear Fuel</b>							
Cask drop	0.01	0.0024	$1.2 \times 10^{-9}$	0.00028	$1.4 \times 10^{-7}$	0.000049	$2.0 \times 10^{-11}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Sodium fire <sup>c</sup>	0.008	5.9	$3.0 \times 10^{-6}$	0.689	0.00034	0.054	$2.2 \times 10^{-8}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

<sup>c</sup> The frequency for this accident is the frequency for the facility design-basis earthquake initiating cell fire.

**Table 4–18 Annual Cancer Risks Due to Accidents at ANL-W for Alternative 2**

<i>Accident</i>	<i>Maximally Exposed Offsite Individual<sup>a</sup></i>	<i>Population Within 80 Kilometers (50 miles)<sup>b</sup></i>	<i>Noninvolved Worker<sup>a</sup></i>
<b><i>Driver Spent Nuclear Fuel</i></b>			
Salt powder spill	$2.3 \times 10^{-12}$	$4.9 \times 10^{-10}$	$1.9 \times 10^{-15}$
Salt transfer drop	$9.5 \times 10^{-15}$	$1.1 \times 10^{-12}$	$2.9 \times 10^{-15}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.6 \times 10^{-9}$	$8.8 \times 10^{-11}$
Cask drop	$1.5 \times 10^{-10}$	$1.8 \times 10^{-8}$	$3.4 \times 10^{-12}$
Earthquake (design-basis earthquake)	$4.8 \times 10^{-8}$	$5.6 \times 10^{-6}$	$1.5 \times 10^{-8}$
Earthquake (beyond-design-basis earthquake)	$2.2 \times 10^{-7}$	0.000013	$1.5 \times 10^{-9}$
<b><i>Blanket Spent Nuclear Fuel</i></b>			
Cask drop	$1.2 \times 10^{-11}$	$1.4 \times 10^{-9}$	$2.0 \times 10^{-13}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.5 \times 10^{-9}$	$8.7 \times 10^{-11}$
Sodium fire	$2.4 \times 10^{-8}$	$2.7 \times 10^{-6}$	$1.7 \times 10^{-10}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be  $2.2 \times 10^{-7}$  per year (or one in 4.5 million years) and  $1.5 \times 10^{-8}$  per year (or one in 66.7 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one in 76,920 years).

***Hazardous Chemical Impacts***

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The hazardous chemical impacts of potential facility accidents associated with the treatment of driver spent nuclear fuels using the electrometallurgical treatment process are summarized in Table 4–19.

**Table 4–19 Hazardous Chemical Impacts Due to Accidents at ANL-W for Alternative 2**

<i>Accident</i>	<i>Frequency (event per year)</i>	<i>Receptor Location</i>	<i>Exposure</i>
Uranium-handling accident	0.01	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.008	Noninvolved worker at 100 meters	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker at 100 meters	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline

#### **4.4.5 Environmental Justice**

As discussed in Section 4.4.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the nine-year period of cleaning blanket spent nuclear fuel for placement in high-integrity cans and treatment of driver spent nuclear fuels using the electrometallurgical treatment would be  $1.7 \times 10^{-9}$  (or one chance in 588 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000014 (or one chance in 71,400). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small, regardless of the racial and ethnic composition of the population, and independent of the economic status of individuals comprising the population. Operation of high-integrity can and electrometallurgical processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

#### **4.4.6 Waste Management**

This alternative would generate process wastes from treatment operations, incidental wastes from normal support operations, and deactivation wastes following the conclusion of operations. Process wastes would include fuel assembly hardware and metal and ceramic high-level radioactive wastes. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types and their expected interim storage and final disposal locations are given in Table 4-7 (see Section 4.2.6).

Estimates of the total amount of other associated process wastes generated as a result of Alternative 2 are provided in **Table 4-20**. These values are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4-20 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of Alternative 2 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

##### *Direct Process Wastes*

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures.

Under this alternative, metal and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metal waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metal waste form. Both the ceramic and metal waste would be categorized as high-level radioactive waste.

**Table 4–20 Amounts of Wastes Generated for Alternative 2<sup>a</sup>**

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
<b>Direct Process Wastes</b>		
Fuel assembly hardware (low-level radioactive waste)	12.5	6,000
High-level radioactive ceramic waste	16.3	24,400
High-level radioactive metal waste	2.0	2,500
Spent nuclear fuel	25.2	63,000
<b>Other Associated Process Wastes</b>		
High-level radioactive wastes	0.4	220
Low-level radioactive wastes <sup>b</sup>	555	113,000
Transuranic wastes	9.1	3,800
Mixed wastes	27.5	14,800
Sanitary wastes	4,960	1.72 × 10 <sup>6</sup>
<b>Deactivation Wastes</b>		
Low-level radioactive wastes <sup>b</sup>	166.2	56,000
Transuranic wastes	1.6	853
Mixed wastes	4.8	3,200

<sup>a</sup> These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage are accomplished during this time period.

<sup>b</sup> The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

The packaged spent nuclear fuel volume is based on placing the blanket spent nuclear fuel in high-integrity cans which will be placed in standardized canisters. The volumes of waste forms provided in Table 4–20 are for the standardized canisters required for disposal of these materials.

The metal and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel and packaged spent nuclear fuel would be temporarily stored for 10 to 15 years at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste and shielding would be provided by a combination of steel storage liners storing the waste, and shielding provided by the soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste cans containing these materials would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

*Other Associated Process High-Level Radioactive Waste*

These high-level radioactive wastes could be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbant used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single high-level radioactive waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

### *Other Associated Process Low-Level Radioactive Wastes*

These low-level radioactive wastes would be generated as a result of processing at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste at ANL-W that will require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area in the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.5 percent of the total Radioactive Waste Management Complex disposal capacity.

### *Other Associated Process Transuranic Wastes*

These transuranic wastes would be generated by Alternative 2 from decontamination activities, repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic wastes would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated as a result of the treatment of sodium-bonded spent nuclear fuel at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is approximately 9 cubic meters (318 cubic feet), which is 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

### *Other Associated Process Mixed Wastes*

These mixed wastes would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1996b).

### *Deactivation Wastes*

A variety of wastes would be generated as part of deactivation activities associated with electrometallurgical treatment processing at ANL-W. These would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation wastes would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers,

as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. Deactivation waste volume is generated in two years. The total deactivation wastes represent an additional 30 percent over the total incidental waste (excluding sanitary waste) requiring disposal.

#### **4.5 ALTERNATIVE 3: DECLAD AND CLEAN BLANKET FUEL AND TREAT (ELECTROMETALLURGICAL TREATMENT) DRIVER FUEL AT ANL-W; PUREX PROCESS BLANKET FUEL AT SRS**

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be declad and cleaned to remove metallic sodium, packaged in aluminum cans at ANL-W, and shipped to SRS for treatment using the PUREX process at F-Canyon. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The high-level radioactive waste generated from treatment of the blanket spent nuclear fuel at SRS would be in the form of borosilicate glass and would be stored at the SRS Defense Waste Processing Facility, pending repackaging and transportation for disposal in a geologic repository. The process steps for the electrometallurgical treatment of driver spent nuclear fuels would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its shipment to SRS could start in 2003 and be completed 2009. PUREX processing of blanket spent nuclear fuel at SRS could be completed by 2010.

##### **4.5.1 Air Quality**

###### *Nonradiological Gaseous Emissions*

It is expected that this alternative at ANL will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

The concentrations of nonradiological air pollutants attributed to this alternative at SRS are presented in **Table 4–21** along with the total estimated site air pollutant concentrations. The concentrations for the alternative are based on information in the *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement* (SRS Spent Nuclear Fuel Management Draft EIS) (DOE 1998g). The concentrations have been adjusted to account for the increased mass of sodium-bonded spent nuclear fuel. The total concentrations are equal to the concentrations for the alternative, plus the baseline concentrations from Section 3.3.3.1. The concentrations are compared to their corresponding ambient air quality standards. Only those air pollutants that are expected to be emitted under this alternative and that have ambient air quality standards are presented in the table. Note that there are no Prevention of Significant Deterioration increment-consuming sources at SRS; therefore, a Prevention of Significant Deterioration increment analysis was not performed.

**Table 4–21 Nonradiological Air Quality Concentrations Associated With Alternative 3 at SRS for Comparison With Ambient Air Quality Standards**

<i>Pollutant</i>	<i>Averaging Period</i>	<i>Most Stringent Standard or Guideline (micrograms per cubic meter)</i>	<i>Alternative 3 Concentration (micrograms per cubic meter)</i>	<i>Total Site Concentration (micrograms per cubic meter)</i>
<b>Criteria pollutants</b>				
Carbon monoxide	8 hours	10,000	1.22	633.02
	1 hour	40,000	9.06	5023.66
Nitrogen dioxide	Annual	100	3.11	11.91
PM <sub>10</sub>	Annual	50	less than 0.01	4.8
	24 hours (interim)	150	0.11	80.72
	24 hours (99 <sup>th</sup> percentile over 3 years)	150	NA	NA
PM <sub>2.5</sub>	3-year annual	15	NA	NA
	24 hours (98 <sup>th</sup> percentile over 3 years)	65	NA	NA
Sulfur dioxide	Annual	80	less than 0.01	16.3
	24 hours	365	0.12	215.52
	3 hours	1,300	0.91	691.1 1
<b>State regulated pollutants</b>				
Gaseous fluoride	30 days	0.8	0.01	0.12
	7 days	1.6	0.03	0.14
	24 hours	2.9	0.06	0.66
	12 hours	3.7	0.11	2.4
Total suspended particulates	Annual	75	less than 0.01	43.3
<b>Hazardous and other toxic compounds</b>				
1,1,1-trichloroethane	24 hours	9,550	less than 0.01	less than 22.01
Benzene	24 hours	150	0.01	31.01
Ethanolamine	24 hours	200	less than 0.01	less than 0.02
Ethyl benzene	24 hours	4,350	less than 0.01	less than 0.13
Ethylene glycol	24 hours	650	less than 0.01	less than 0.09
Formaldehyde	24 hours	15	less than 0.01	less than 0.02
Glycol ethers	24 hours	No standard	less than 0.01	less than 0.02
Hexachloronaphthalene	24 hours	1	less than 0.01	less than 0.02
Hexane	24 hours	900	0.01	0.08
Manganese	24 hours	25	less than 0.01	less than 0.11
Methyl alcohol	24 hours	1,310	less than 0.01	less than 0.52
Methyl ethyl ketone	24 hours	14,750	less than 0.01	less than 1
Methyl isobutyl ketone	24 hours	2,050	less than 0.01	less than 0.52
Methylene chloride	24 hours	8,750	0.01	1.81
Naphthalene	24 hours	1,250	less than 0.01	less than 0.02
Nitric acid	24 hours	125	0.28	6.98
Phenol	24 hours	190	less than 0.01	less than 0.04
Phosphorous	24 hours	0.5	less than 0.01	less than 0.01
Sodium hydroxide	24 hours	50	less than 0.01	less than 0.02
Toluene	24 hours	2,000	0.01	1.61
Trichloroethane	24 hours	6,750	less than 0.01	less than 1.01

<i>Pollutant</i>	<i>Averaging Period</i>	<i>Most Stringent Standard or Guideline (micrograms per cubic meter)</i>	<i>Alternative 3 Concentration (micrograms per cubic meter)</i>	<i>Total Site Concentration (micrograms per cubic meter)</i>
Vinyl acetate	24 hours	176	less than 0.01	less than 0.03
Xylene	24 hours	4,350	0.02	3.82

Source: Bickford et. al. 1997, plus baseline concentrations from Section 3.3.3.1.  
NA = Not Available.

### *Radiological Gaseous Emissions*

The decladding and cleaning of blanket spent nuclear fuel and the electrometallurgical treatment of driver spent nuclear fuel at ANL-W would release gaseous fission products to the hot (argon) cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The released tritium in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries ( $10^{-9}$  curies) and are released to the atmosphere through the facility stack, along with the krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions occurs when decladding blanket spent nuclear fuel for packaging and shipment to SRS and electrometallurgical treatment of driver spent nuclear fuel are performed simultaneously. This simultaneous operation was estimated to occur over a three-year period starting in 2003. Based on an annual decladding throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and an electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere.

Since declad and clean fuels are packaged and sent to SRS, some gaseous fission products are expected to be present in that fuel. However, it was conservatively assumed that all gaseous fission products in the blanket spent nuclear fuels would be released to the environment during PUREX processing at SRS over a six-month period (see Appendix E, Section E.4.3). The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.5.4.

### **4.5.2 Water Resources**

As stated in Section 4.4.2, decladding and cleaning of blanket spent nuclear fuels and treatment of driver spent nuclear fuels using electrometallurgical treatment would not discharge any radiological chemical material to the surface or groundwater at the INEEL site. These activities also would not impact the current groundwater usage at the site. For a discussion of impacts on water resources at ANL-W, see Section 4.4.2.

The impacts on water resources from processing blanket spent nuclear fuels at F-Canyon are described below.

#### **Surface Water**

No surface water would be used for PUREX processing of blanket spent nuclear fuel at the F-Area. The F-Canyon processing facilities are outside the 100-year floodplain, as shown in Figure 3-7.

*Nonradiological Liquid Effluent*

The major sources of liquid effluents from PUREX processing of blanket spent nuclear fuel at SRS would be process cooling water and steam condensate. There are sufficient capacities in existing wastewater treatment facilities to handle the liquid effluents from this processing. Liquid effluents associated with PUREX processes would use these facilities and the existing permitted outfalls (Section 3.3.4.1). Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility and discharged through an existing NPDES-permitted outfall (G-10). Since employment would not increase as a result of processing these fuels, the treatment rates through the Central Wastewater Treatment Facility would not be affected and the requirements of the SRS NPDES permit would continue to be met (DOE 1998g).

Process cooling water treatment would result in releases to Upper Three Runs Creek from the F-Area, as shown in **Table 4-22**.

**Table 4-22 Chemical Effluent Concentrations From PUREX Cooling Water Treatment**

Parameter	Effluent Concentrations	Existing Stream Water Concentrations		Water Quality Criterion (milligrams per liter) <sup>c</sup>
	F-Area (milligrams per liter)	Upper Three Runs (upstream) <sup>a</sup> (average) (milligrams per liter)	Upper Three Runs (downstream) <sup>b</sup> (average) (milligrams per liter)	
Aluminum	0.2	0.19	0.24	(d)
Ammonia	0.03	0.0001	NR	(d)
Chromium	0.02	ND	ND	0.1
Copper	0.01	0.018	0.015	1
Manganese	0.01	0.039	0.052	0.05
Nickel	0.05	ND	ND	0.1
Nitrate	0.04	0.36	0.27	10
Zinc	0.07	0.06	0.091	3

<sup>a</sup> Stream monitor U3R-1A.

<sup>b</sup> Stream monitor U3R-4.

<sup>c</sup> Federal Drinking Water Standards and Health Advisories (EPA 1996) and South Carolina Water Quality Criteria for Protection of Human Health (SCDHEC 1998).

<sup>d</sup> No drinking water standard.

Key: ND = not detected; NR = not reported.

Sources: Arnett and Mamatey 1998, DOE 1998g.

Although proposed or final Federal drinking water standards do not apply to the discharges, these standards are used for comparison to SRS discharges. The discharge concentration would not exceed the Federal drinking water standard. The discharges would also comply with the South Carolina Water Quality Standards (SCDHEC 1998). The release concentrations would be no greater than the concentrations measured in Upper Three Runs (Arnett and Mamatey 1998), with the exception of zinc and ammonia. Zinc concentrations in the discharge are within the Federal health advisory limits (EPA 1996).

*Radiological Liquid Effluent*

PUREX processing would release measurable radioactive nuclides to the surface water through the cooling water system. The expected radiological effluents from processing declad and cleaned blanket spent nuclear fuels at F-Canyon were estimated based on the measured data from various effluent streams at F-Area as presented in the SRS Environmental Data for 1997 (Arnett and Mamatey 1998). Since the mechanism associated with releases of liquid effluent from PUREX processing at F-Canyon is essentially the same for

almost every fuel type processed, the F-Area 1997 effluent data were used to conservatively represent the potential releases from a 6-month operation of F-Canyon. **Table 4–23** provides a list of potential radiological isotopes that could be released to the surface water during processing of about 57 metric tons of heavy metal of blanket spent nuclear fuel (see Appendix E, Sections E.3, for details).

**Table 4–23 Estimated Radiological Liquid Effluents During PUREX Processing of Blanket Spent Nuclear Fuels**

<i>Isotope</i>	<i>Curies Released</i>
Tritium	1.54
Strontium-89/Strontium-90	0.000031
Ruthenium-103/Ruthenium-106	0.0022
Uranium-234	0.000085
Promethium-147	0.000011
Uranium-238	0.00019
Plutonium-238	0.000016
Plutonium-239	$7.8 \times 10^{-6}$

Source: Arnett and Mamatey 1998.

### **Groundwater**

All process water would come from groundwater, as would sanitary water. At most, less than 65 million liters (17 million gallons) per year would be required for cooling water. SRS annually withdraws more than 5 billion liters (1.3 billion gallons) per year of groundwater (DOE 1998g).

#### *Nonradiological Liquid Effluent*

No nonradiological chemicals would be discharged to groundwater from PUREX processing of blanket spent nuclear fuels at F-Canyon and the FB-Line in F-Area.

#### *Radiological Liquid Effluent*

No radiological liquid effluent or waste would be discharged to groundwater from PUREX processing of blanket spent nuclear fuels at F-Canyon and the FB-Line in F-Area.

### **4.5.3 Socioeconomics**

Under this alternative, the existing facilities at ANL-W and SRS would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the regions around INEEL and SRS.

### **4.5.4 Public and Occupational Health and Safety**

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in **Tables 4–24** and **4–25** for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in **Tables 4–28** and **4–29**. The impacts from hazardous chemical releases during accident conditions are presented in **Table 4–32**. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

#### **4.5.4.1 Normal Operations**

##### *Radiological Impacts*

Under this alternative, radioactive releases would occur during PUREX processing at F-Canyon. Appendix E, Sections E.3 and E.4.3, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. Doses to the public would result from treating about 57 metric tons of heavy metal of blanket spent nuclear fuel. The blanket spent nuclear fuels being processed at SRS are already declad and cleaned at ANL-W; therefore, the gaseous fission products are assumed to have already been released. However, for the analytical purposes of this EIS, it was conservatively assumed that the gaseous fission products are still within the matrix of the fuel and would be released during PUREX processing at SRS. The processing was assumed to continue for six months (see Appendix E.4.3).

Calculated incremental maximum annual radiological impacts to the public are given in Table 4–24. The impacts are calculated for two receptor groups: the general public living within 80 kilometers (50 miles) of F-Canyon in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the SRS site boundary and receiving the maximum dose). Since PUREX processing would produce radiological air emissions as well as radiological liquid effluent, doses to the public were calculated considering both the air emissions and liquid effluent. Primary contributors to public doses are from tritium gases (assumed to be tritium oxide) and krypton-85, which together contribute over 95 percent of the total calculated doses. The doses resulting from liquid effluent were estimated from data provided in support of the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g) (see Appendix E, Section E.4.3, for details). The doses and duration from decladding and cleaning blanket spent nuclear fuels and treatment of driver spent nuclear fuels at ANL-W would be similar to those presented for Alternative 2 in Section 4.4.4.1. To put the impacts into perspective, comparisons to natural background radiation levels are included in Table 4–25.

Table 4–25 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by SRS workers involved with the PUREX process. The estimated annual total worker population dose would be 75 person-rem, with an average individual dose of 500 millirem per year for each of the 150 involved workers. If these estimates were projected for six months of PUREX activities, the cumulative worker population dose would be 38 person-rem, leading to a risk of 0.015 latent cancer fatalities. The estimated annual total worker population dose to treat driver spent nuclear fuels at ANL-W is 22 person-rem, as indicated in Section 4.4.4.1.

As shown in Tables 4–24 and 4–25:

- The annual dose to the maximally exposed offsite individual would be 0.00038 millirem per year, with an associated  $1.9 \times 10^{-10}$  risk per year of developing fatal cancer (or one in 5.3 billion years).
- The collective dose to the population within 80 kilometers (50 miles) of the ANL-W facilities would be 0.0031 person-rem per year, with an associated  $1.6 \times 10^{-6}$  latent cancer fatalities per year (or one in 625,000 years).
- The collective dose to ANL-W facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one in 113 years).
- The dose to the maximally exposed offsite individual from six-month PUREX processing would be 0.00051 millirem, with an associated  $2.6 \times 10^{-10}$  risk of developing fatal cancer (or one chance in 3.8 billion).

**Table 4–24 Annual Radiological Impacts to the Public From Operational Activities Associated With Alternative 3**

<i>Receptor</i>	<i>Alternative 3</i>	
	<i>PUREX Process of Declad and Cleaned Blanket Spent Nuclear Fuel at SRS<sup>a</sup></i>	<i>Declad and Clean Blanket Fuel and Treat Driver Spent Nuclear Fuel at ANL-W</i>
<b>Population Dose Within 80 Kilometers (50 Miles) in the Year 2010</b>		
Dose (person-rem)	0.02 <sup>b</sup>	0.0031
Latent cancer fatalities	0.000010	$1.6 \times 10^{-6}$
<b>Annual Dose to the Maximally Exposed Offsite Individual</b>		
Dose (millirem)	0.00051 <sup>b</sup>	0.00038
Latent cancer fatality risk	$2.6 \times 10^{-10}$	$1.9 \times 10^{-10}$
Percent of natural background <sup>c</sup>	0.00017	0.00011
<b>Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)</b>		
Dose (millirem)	0.000024 <sup>d</sup>	0.000013 <sup>e</sup>
Latent cancer fatality risk	$1.2 \times 10^{-11}$	$6.6 \times 10^{-12}$

<sup>a</sup> Includes airborne and liquid dose components over the six-month processing duration.

<sup>b</sup> Liquid dose contributions to the population and the maximally exposed individual dose are 0.00068 person-rem and 0.00012 millirem, respectively.

<sup>c</sup> The annual natural background radiation level at INEEL and at SRS is 360 and 300 millirem, respectively, for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem at INEEL (254,000 at SRS).

<sup>d</sup> Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of the SRS F-Canyon in the year 2010 (848,000).

<sup>e</sup> Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

**Table 4–25 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 3**

<i>Impact</i>	<i>Alternative 3</i>	
	<i>Operations at SRS<sup>b</sup></i>	<i>Operations at ANL-W</i>
<b>Worker<sup>a</sup></b>		
Total dose (person-rem per year)	38 <sup>c</sup>	22
Fatal cancers	0.015 <sup>c</sup>	0.079 <sup>d</sup>
Average worker dose (millirem per year)	250 <sup>c</sup>	60
Fatal cancer risk	0.00010 <sup>c</sup>	0.00022 <sup>d</sup>

<sup>a</sup> The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to an involved worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year as established for all DOE activities in DOE Order N441.1.

<sup>b</sup> Estimates from DOE 1998g.

<sup>c</sup> Operations at SRS to treat blanket spent nuclear fuel at F-Canyon are performed over six months.

<sup>d</sup> Operations at ANL-W to declad and clean blanket spent nuclear fuels and treat driver spent nuclear fuels are performed over nine years.

- The collective dose to the population within 80 kilometers (50 miles) of the F-Canyon would be 0.02 person-rem, with an associated 0.000010 latent cancer fatalities (or one chance in 100,000).
- The collective dose to F-Canyon facility workers would be 38 person-rem, with an associated 0.015 latent cancer fatalities (or one chance in 67).

### *Hazardous Chemical Impacts*

It is expected that hazardous chemical impacts associated with this alternative at ANL-W would be small, as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described in Section 3.2.10.2.

For SRS, both carcinogenic and noncarcinogenic health effects to the public were assessed from exposure to hazardous chemicals. It was assumed that under normal operating conditions, the primary exposure pathway for members of the public would be via air emissions.

The 24-hour concentrations provided in Section 4.5.1 were converted to annual concentrations by using the appropriate regulatory scaling factor of 0.125 based on South Carolina's Air Quality Modeling Guidelines (SCDHEC 1993). This annual concentration for each noncarcinogenic chemical was divided by the corresponding inhalation reference concentration to estimate the hazard quotient for each chemical. The hazard quotients were summed to give the hazard index from all noncarcinogenic chemicals for this alternative. A hazard index less than 1 indicates that adverse health effects from noncancer-causing agents are not expected. For carcinogens, the annual concentration was multiplied by the unit cancer risk to estimate the increased cancer risk from that chemical. Hazardous chemical health effects are summarized in **Tables 4-26** and **4-27**.

#### **4.5.4.2 Facility Accidents**

### *Radiological Impacts*

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities associated with decladding, cleaning, and PUREX processing of blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel are summarized and presented in this section. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than  $10^{-7}$  per year, and consequence analyses for these two events were not performed. Decladding and cleaning of blanket spent nuclear fuel is performed in the Hot Fuel Examination Facility; treatment of driver spent nuclear fuel is performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing takes place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event. Decladding and cleaning blanket spent nuclear fuel is performed only in the Hot Fuel Examination Facility. The multifacility impacts of the beyond-design-basis earthquake are not relevant to the blanket spent nuclear fuel. Therefore, only the higher frequency design-basis seismic event was analyzed for the blanket spent nuclear fuel. Table 4-28 presents the frequencies and consequences of the postulated set of accidents at ANL-W to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility.

**Table 4–26 Hazardous Chemical Impacts to the Public From Operational Activities at SRS for Alternative 3**

<i>Chemical</i>	<i>Annual Concentration (micrograms per cubic meter)</i>	<i>Reference Concentration Inhalation (micrograms per cubic meter)</i>	<i>Unit Cancer Risk (risk per microgram per cubic meter)</i>	<i>Hazard Quotient</i>	<i>Cancer Risk</i>
Benzene	$1.4 \times 10^{-6}$	None	0.0078	None	$1.1 \times 10^{-8}$
Ethyl benzene	$1.3 \times 10^{-6}$	1	None	$1.3 \times 10^{-6}$	None
Formaldehyde	$1.3 \times 10^{-6}$	None	0.013	None	$1.6 \times 10^{-8}$
Hexane	$1.4 \times 10^{-6}$	0.2	None	$7.1 \times 10^{-6}$	None
Manganese	$1.3 \times 10^{-6}$	0.000050	None	0.025	None
Methyl ethyl ketone	$2.5 \times 10^{-6}$	1	None	$2.5 \times 10^{-6}$	None
Methylene chloride	$7.1 \times 10^{-7}$	None	0.00047	None	$3.3 \times 10^{-10}$
Naphthalene	$1.3 \times 10^{-6}$	0.003	None	0.00042	None
Toluene	$1.4 \times 10^{-6}$	0.4	None	$3.5 \times 10^{-6}$	None
Vinyl acetate	$1.3 \times 10^{-6}$	0.2	None	$6.3 \times 10^{-6}$	None
Hazard Index				0.025	None

Source: EPA 1999.

**Table 4–27 Hazardous Chemical Impacts to the Noninvolved Worker From Operational Activities at SRS for Alternative 3**

<i>Chemical</i>	<i>Annual Concentration (micrograms per cubic meter)</i>	<i>Reference Concentration-inhalation (micrograms per cubic meter)</i>	<i>Unit Cancer Risk (risk per microgram per cubic meter)</i>	<i>Hazard Quotient</i>	<i>Cancer Risk</i>
Benzene	0.0071	None	0.0078	None	0.000055
Ethyl benzene	0.0035	1	None	0.0035	None
Formaldehyde	0.0035	None	0.013	None	0.000046
Hexane	0.0071	0.2	None	0.035	None
Manganese	$3.7 \times 10^{-11}$	0.00005	None	$7.4 \times 10^{-7}$	None
Methyl ethyl ketone	0.0035	1	None	0.0035	None
Methylene chloride	0.0071	None	0.00047	None	$3.3 \times 10^{-6}$
Naphthalene	$1.5 \times 10^{-11}$	0.003	None	$5.0 \times 10^{-10}$	None
Toluene	0.0071	0.4	None	0.018	None
Vinyl acetate	0.0035	0.2	None	0.018	None
Hazard Index				0.078	NA

Sources: DOE 1998g, EPA 1999.

The dose to the maximally exposed offsite individual was calculated for the 95<sup>th</sup> percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50<sup>th</sup> percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents. (See the discussion on the involved worker in Section 4.2.4.2.) The accident risks for the same receptors are summarized in Table 4–29.

**Table 4–28 Accident Frequency and Consequences at ANL-W for Alternative 3**

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population within 80 kilometers (50 miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>	Dose (person- rem)	Latent Cancer Fatalities <sup>b</sup>	Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>
<b>Driver Spent Nuclear Fuel</b>							
Salt powder spill	0.01	0.00046	$2.3 \times 10^{-10}$	0.000098	$4.9 \times 10^{-8}$	$4.7 \times 10^{-7}$	$1.9 \times 10^{-13}$
Salt transfer drop	$1.0 \times 10^{-7}$	0.19	$9.5 \times 10^{-8}$	0.022	0.000011	0.073	$2.9 \times 10^{-8}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Cask drop	0.01	0.030	$1.5 \times 10^{-8}$	0.0035	$1.8 \times 10^{-6}$	0.00084	$3.4 \times 10^{-10}$
Earthquake (design-basis earthquake at Hot Fuel Examination Facility)	0.008	12	$6.0 \times 10^{-6}$	1.4	0.0007	4.7	$1.9 \times 10^{-6}$
Earthquake (beyond-design-basis earthquake)	0.00001	22,000	0.022	2,500	1.3	370	0.00015
<b>Blanket Spent Nuclear Fuel</b>							
Cask drop	0.01	$2.4 \times 10^{-3}$	$1.2 \times 10^{-9}$	$2.8 \times 10^{-4}$	$1.4 \times 10^{-7}$	$4.9 \times 10^{-5}$	$2.0 \times 10^{-11}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Sodium fire <sup>c</sup>	0.008	5.9	$3.0 \times 10^{-6}$	0.689	$3.4 \times 10^{-4}$	0.054	$2.2 \times 10^{-8}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

<sup>c</sup> The frequency for this accident is the frequency for the facility design-basis earthquake initiating cell fire.

**Table 4–29 Annual Cancer Risks Due to Accidents at ANL-W for Alternative 3**

Accident	Maximally Exposed Offsite Individual <sup>a</sup>	Population within 80 kilometers (50 miles) <sup>b</sup>	Noninvolved Worker <sup>a</sup>
<b>Driver Spent Nuclear Fuel</b>			
Salt powder spill	$2.3 \times 10^{-12}$	$4.9 \times 10^{-10}$	$1.9 \times 10^{-15}$
Salt transfer drop	$9.5 \times 10^{-15}$	$1.1 \times 10^{-12}$	$2.9 \times 10^{-15}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.6 \times 10^{-9}$	$8.8 \times 10^{-11}$
Cask drop	$1.5 \times 10^{-10}$	$1.8 \times 10^{-8}$	$3.4 \times 10^{-12}$
Earthquake (design-basis earthquake)	$4.8 \times 10^{-8}$	$5.6 \times 10^{-6}$	$1.5 \times 10^{-8}$
Earthquake (beyond-design-basis earthquake)	$2.2 \times 10^{-7}$	0.000013	$1.5 \times 10^{-9}$
<b>Blanket Spent Nuclear Fuel</b>			
Cask drop	$1.2 \times 10^{-11}$	$1.4 \times 10^{-9}$	$2.0 \times 10^{-13}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.5 \times 10^{-9}$	$8.7 \times 10^{-11}$
Sodium fire	$2.3 \times 10^{-8}$	$2.7 \times 10^{-6}$	$1.7 \times 10^{-10}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be  $2.2 \times 10^{-7}$  per year (or one in 4.5 million years) and  $1.5 \times 10^{-8}$  per year (or one in 66.7 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one in 76,920 years).

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during PUREX operational activities at SRS are summarized below. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. **Table 4–30** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 350 meters (1150 feet) from the facility. The 350-meter (1150-foot) distance leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95<sup>th</sup> percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50<sup>th</sup> percentile meteorological conditions. DOE did not quantitatively estimate the facility worker population dose due to accidents. The accident risks for the same receptors are summarized in **Table 4–31**.

**Table 4–30 Accident Frequency and Consequences at SRS for Alternative 3**

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population within 80 kilometers (50 miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>	Dose (person-rem)	Latent Cancer Fatalities <sup>b</sup>	Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>
Fire (F-Canyon)	0.000061	610	0.00031	5500	2.8	2300	0.00092
Explosion (FB-Line)	0.00010	6.5	$3.3 \times 10^{-6}$	53	0.027	19	$7.6 \times 10^{-6}$
Earthquake (F-Canyon)	0.00013	1100	0.00055	2100	1.1	12000	0.0048
Earthquake (FB-Line)	0.00013	58	0.000029	120	0.06	900	0.00036
Criticality	0.00010	11	$5.5 \times 10^{-6}$	59	0.030	37	0.000015
Aircraft crash	NA	NA	NA	NA	NA	NA	NA

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

NA = Not analyzed, because the frequency is estimated to be less than  $10^{-7}$  per year (see Appendix F for details).

**Table 4–31 Annual Cancer Risks Due to Accidents at SRS for Alternative 3**

Accident	Maximally Exposed Offsite Individual <sup>a</sup>	Population within 80 kilometers (50 miles) <sup>b</sup>	Noninvolved Worker <sup>a</sup>
Fire (F-Canyon)	$1.9 \times 10^{-8}$	0.00017	$5.6 \times 10^{-8}$
Explosion (FB-Line)	$3.3 \times 10^{-10}$	$2.7 \times 10^{-6}$	$7.6 \times 10^{-10}$
Earthquake (F-Canyon)	$7.2 \times 10^{-8}$	0.00014	$4.8 \times 10^{-7}$
Earthquake (FB-Line)	$3.8 \times 10^{-9}$	$7.8 \times 10^{-6}$	$4.7 \times 10^{-8}$
Criticality	$5.5 \times 10^{-10}$	$3.0 \times 10^{-6}$	$1.5 \times 10^{-9}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

For accidents at SRS, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be  $7.2 \times 10^{-8}$  per year (or one in 13.9 million years) and  $4.8 \times 10^{-7}$  per year (or one in 2.1 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00017 per year (or one in 5,880 years).

*Hazardous Chemical Impacts*

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The hazardous chemical impacts of potential facility accidents at ANL-W associated with the treatment of driver spent nuclear fuel using electrometallurgical treatment are summarized in Table 4–32.

**Table 4–32 Hazardous Chemical Impacts Due to Accidents at ANL-W for Alternative 3**

<i>Accident</i>	<i>Frequency (event per year)</i>	<i>Receptor Location</i>	<i>Exposure</i>
Uranium-handling accident	0.01	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.008	Noninvolved worker at 100 meters	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker at 100 meters	Cadmium: less than ERPG-1 Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1 Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline

The SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g) analyzed the consequences of three chemical spills involving hazardous chemicals in the F-Area: 1) the loss of 50 percent sodium hydroxide containment from a skid-mounted 1,000-gallon dumpster; 2) the loss of 50 percent nitric acid containment from a skid-mounted 1,000-gallon dumpster; and 3) the loss of 30 percent sodium nitrite containment from a skid-mounted 1,000-gallon dumpster and an adjacent 1,600-gallon holdup tank. These analyses are summarized in the **Table 4–33**, and are considered representative of wet storage accidents at SRS.

**Table 4–33 Hazardous Chemical Impacts Due to Accidents at SRS for Alternative 3**

<i>Accident</i>	<i>Frequency (event per year)</i>	<i>Receptor Location</i>	<i>Exposure</i>
Wet storage, container rupture	0.005	Noninvolved worker	sodium hydroxide: less than Permissible Exposure Limit-Time Weighted Average
Wet storage, container rupture	0.005	Noninvolved worker at 640 meters	nitric acid: less than Permissible Exposure Limit-Time Weighted Average
		Maximally exposed offsite individual	nitric acid: less than Permissible Exposure Limit-Time Weighted Average
Wet storage, container rupture	0.006	Noninvolved worker	sodium nitrite: less than Permissible Exposure Limit-Time Weighted Average

Permissible Exposure Limit-Time Weighted Average is used for chemicals having no ERPG values. It is considered to be less than ERPG-1.

ERPG = Emergency Response Planning Guideline.

Source: DOE 1998g.

### 4.5.5 Environmental Justice

As discussed in Section 4.5.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the six months of PUREX processing of blanket spent nuclear fuel at SRS and nine years of electrometallurgical treatment of driver spent nuclear fuel at ANL-W would be  $1.7 \times 10^{-9}$  (or one chance in 588 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000014 (or one chance in 71,400). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small regardless of the racial and ethnic composition of the population and independent of the economic status of individuals comprising the population. Operation of PUREX at SRS, and electrometallurgical treatment and decladding and cleaning processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

### 4.5.6 Waste Management

#### ANL-W

This alternative would generate process wastes from treatment operations, other associated process wastes from normal support operations, and deactivation wastes following the conclusion of operations. Process wastes would include fuel assembly hardware and metal and ceramic high-level radioactive wastes. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types generated at ANL-W and their expected interim storage and final disposal locations are given in Table 4–7 (see Section 4.2.6).

Estimates of the total amount of other associated process wastes generated as a result of Alternative 3 are provided in **Table 4–34**. These values are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4–34 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of Alternative 3 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL, prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

#### *Direct Process Wastes*

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures. In addition, the blanket spent nuclear fuel cladding is included in the fuel hardware stream.

Under this alternative, metal and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metal waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metal waste form. Both the ceramic and metal waste would be categorized as

high-level radioactive waste. The volumes of waste forms provided in Table 4–34 are for the standardized canisters required for disposal of these materials.

**Table 4–34 Amounts of Wastes Generated at ANL-W for Alternative 3<sup>a</sup>**

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
<b>Direct Process Wastes</b>		
Fuel assembly hardware (low-level radioactive waste)	37.5	13,100
High-level radioactive ceramic waste	16.3	24,400
High-level radioactive metal waste	2.0	2,500
Spent nuclear fuel	0	0
<b>Other Associated Process Wastes</b>		
High-level radioactive wastes	0.4	220
Low-level radioactive wastes <sup>b</sup>	555	113,000
Transuranic wastes	9.1	3,800
Mixed wastes	27.5	14,800
Sanitary wastes	4,960	$1.72 \times 10^6$
<b>Deactivation Wastes</b>		
Low-level radioactive wastes <sup>b</sup>	178	60,000
Transuranic wastes	1.6	853
Mixed wastes	5.1	3,400

<sup>a</sup> These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage are accomplished during this time period.

<sup>b</sup> The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

The metal and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel at ANL-W would be temporarily stored for 10 to 15 years at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste, and shielding would be provided by a combination of steel storage liners storing the waste and by the soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste cans containing the metal and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

*Other Associated Process High-Level Radioactive Waste*

These high-level radioactive wastes could be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbant used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single high-level radioactive waste canister. Conservatively the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

#### *Other Associated Process Low-Level Radioactive Wastes*

These low-level radioactive wastes would be generated as a result of decladding and cleaning blanket spent nuclear fuel and treatment of driver spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from decladding and cleaning blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel at ANL-W that will require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 555 cubic meters (19,600 cubic feet) represents approximately 0.5 percent of the total Radioactive Waste Management Complex disposal inventory.

#### *Other Associated Process Transuranic Wastes*

These transuranic wastes would be generated by Alternative 3 from decontamination activities, repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic wastes would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant facility. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is approximately 9 cubic meters (318 cubic feet), which is less than 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

#### *Other Associated Process Mixed Wastes*

These mixed wastes would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1996b).

#### *Deactivation Wastes*

A variety of wastes would be generated as part of deactivation activities associated with decladding and cleaning blanket spent nuclear fuel and the treatment of driver spent nuclear fuel at ANL-W. These would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation wastes would be low-level radioactive waste generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot isostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The deactivation waste volume is generated over two years. The total deactivation wastes represent an additional 30 percent over the total incidental waste (excluding sanitary wastes) requiring disposal.

**SRS**

The PUREX process at SRS would generate process wastes from treatment operations and other associated process wastes from support operations. Process wastes would include high-level radioactive waste. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. The incidental wastes include low-level radioactive wastes, transuranic wastes, and mixed wastes. All of the waste streams would be categorized according to existing DOE Orders and SRS waste management procedures. The anticipated categorization of the waste types and their expected interim storage and final disposal locations are given in **Table 4-35**.

**Table 4-35 Summary of Waste Material Categories at SRS for Alternative 3**

<i>Waste Stream</i>	<i>Category</i>	<i>Interim Storage</i>	<i>Final Disposal</i>
<b>Process Wastes</b>			
Liquid waste form	High-level radioactive waste	Initial storage in the high-level radioactive waste Tank Farm followed by post-process storage at the Defense Waste Processing Facility.	Offsite (proposed geologic) repository
<b>Other Associated Process Wastes</b>			
Less than 100 nanocuries per gram transuranic waste	Low-level radioactive waste	None	Low-activity waste vaults
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Transuranic waste storage pads	Waste Isolation Pilot Plant
Contaminated	Mixed waste	Mixed waste storage buildings	Off site

Estimates of the amounts of wastes generated as a result of the PUREX alternative at SRS are provided in **Table 4-36**. These values are based on an evaluation of waste forecasts that account only for the fraction of total waste that would be attributable to processing the blanket spent nuclear fuel pins under the PUREX alternative.

As indicated in the following waste type discussions, the amounts of wastes associated with this processing alternative are relatively small compared to onsite and offsite management capacities.

**Table 4-36 Amounts of Wastes Generated at SRS for Alternative 3**

<i>Waste Stream</i>	<i>Total Waste Generated (cubic meters)<sup>a</sup></i>
<b>Direct Process Wastes</b>	
Liquid high-level radioactive waste	510
Equivalent Defense Waste Processing Facility Canisters <sup>b</sup>	5.6 (9 canisters)
Saltstone <sup>b</sup>	1,290
<b>Other Associated Process Wastes</b>	
Low-level radioactive waste	3,600 <sup>c</sup>
Transuranic waste	90
Mixed waste	6.9

<sup>a</sup> These values are estimated based on heavy metal mass ratio of similar materials processed at SRS (20 metric tons of heavy metal) and provided in DOE 1998g.

<sup>b</sup> These wastes result from processing the liquid high-level radioactive waste.

<sup>c</sup> Assuming a volume reduction factor of 4, the estimated disposal volume would be about 900 cubic meters (31,780 cubic feet).

#### *Direct Process Wastes*

During the PUREX process, liquid high-level radioactive waste would be produced (along with plutonium metal and uranium solution). The liquid waste would be processed in the Defense Waste Processing Facility to yield vitrified high-level radioactive waste (borosilicate glass) and saltstone. This high-level radioactive waste would be temporarily stored at the Defense Waste Processing Facility pending ultimate disposal in an offsite (proposed geologic) repository. The saltstone is a cement form low-level radioactive waste that is generated or a by-product of SRS tank farm operations. The saltstone would be disposed of on site in the Z-Area Saltstone Vaults. The volume of this saltstone would be about 0.12 percent of the 1.11 million cubic meters (39.2 million cubic feet) storage capacity of the vaults.

#### *Other Associated Process Low-Level Radioactive Wastes*

These low-level radioactive waste would be generated during the PUREX process. The volume of low-level radioactive waste resulting from this alternative (after volume reduction) would be about 3 percent of the total 30,500-cubic meter (1.08 million-cubic foot) disposal capacity of the low-activity waste vaults.

#### *Other Associated Process Transuranic Wastes*

The volume of transuranic waste generated during the PUREX process would be only about 0.05 percent of the current 168,500-cubic meter (5.95 million-cubic foot) limit for the Waste Isolation Pilot Plant (DOE 1997).

#### *Other Associated Process Mixed Wastes*

These mixed wastes generated during the PUREX process would be temporarily stored on site in the Mixed Waste Storage Buildings prior to eventual offsite disposal. The volume of this waste would be about 0.36 percent of the 1,900-cubic meter (67,100-cubic foot) storage capacity of these storage buildings.

### **4.6 ALTERNATIVE 4: MELT AND DILUTE BLANKET FUEL AND TREAT (ELECTROMETALLURGICAL TREATMENT) DRIVER FUEL AT ANL-W**

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be cleaned to remove metallic sodium and then treated using the melt and dilute process at ANL-W. The melt and dilute product from treatment of this fuel would be stored at the Radioactive Scrap and Waste Facility pending repackaging and transportation for disposal in a geologic repository. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated

at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuels would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel could start in 2003 and subsequent treatment by melt and dilute at ANL-W could start in 2005 and be completed by 2012.

#### **4.6.1 Air Quality**

##### *Nonradiological Gaseous Emissions*

It is expected that this alternative at ANL will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

##### *Radiological Gaseous Emissions*

Cleaning blanket spent nuclear fuel and electrometallurgical treatment of driver spent nuclear fuel would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The tritium released into the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries ( $10^9$  curies) and are released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum annual release of radioactive gaseous emissions occurs when electrometallurgical treatment processing of driver spent nuclear fuels performed simultaneously with cutting blanket spent nuclear fuels for sodium removal prior to the melt and dilute process. This simultaneous operation would occur over a 3-year period during the estimated 10 years of operation, starting in 2003. Based on an annual blanket spent nuclear fuel processing (e.g., chopping and cleaning) throughput of 10 metric tons of heavy metal and electrometallurgical treatment processing of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 could be released annually to the atmosphere. The radiological exposures to the public and workers from radioactive emissions are presented in Section 4.6.4.

#### **4.6.2 Water Resources**

##### *Surface Water*

No surface water is used at ANL-W. Flood waters from Big Lost River are not expected to reach the facilities at ANL-W, as shown in Figure 3–3.

##### *Nonradiological Liquid Effluent*

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and to the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with electrometallurgical and melt and dilute treatment processes. Current operating and monitoring practices would continue for NPDES stormwater and liquid effluent discharges associated with facilities at ANL-W.

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-

contained pipes, leak detection, and secondary containment of tanks are some of the features used to prevent hazardous material releases to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996a).

#### *Radiological Liquid Effluent*

No radiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to surface water.

#### **Groundwater**

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. Water consumption for the electrometallurgical and melt and dilute treatment process operations would not impact the current water usage at ANL-W. The current water usage at ANL-W is 188 million liters (49.6 million gallons) per year.

#### *Nonradiological Liquid Effluent*

No nonradiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to groundwater.

#### *Radiological Liquid Effluent*

No radiological liquid effluent generated by electrometallurgical and melt and dilute treatment process operations would be discharged to groundwater.

### **4.6.3 Socioeconomics**

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around ANL-W and INEEL.

### **4.6.4 Public and Occupational Health and Safety**

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in **Tables 4-37** and **4-38** for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in **Tables 4-39** and **4-40**. The impacts from hazardous chemical releases during accident conditions are presented in **Table 4-41**. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

#### **4.6.4.1 Normal Operations**

##### *Radiological Impacts*

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel cleaning and driver spent nuclear fuel chopping and electrorefining. All of these activities are performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when both blanket and driver spent nuclear fuels are treated

simultaneously under this alternative. Appendix E, Section E.4.2, provides details on the treatment process duration and throughputs for each fuel type.

Calculated maximum annual radiological impacts to the public are given in Table 4–37. The impacts are calculated for two receptor groups: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public are releases of tritium gases (about 1 percent of which was conservatively assumed to be in oxidized form) and krypton-85, which together contribute over 99.9 percent of the total calculated doses. To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table.

**Table 4–37 Annual Radiological Impacts to the Public From Operational Activities Associated With Alternative 4**

<i>Receptor</i>	<i>Alternative 4</i>		
	<i>Electrometallurgical Treatment of Driver Spent Nuclear Fuel</i>	<i>Clean and Melt and Dilute Blanket Spent Nuclear Fuel at ANL-W</i>	<i>Total</i>
<b>Population Dose Within 80 Kilometers (50 Miles) in the Year 2010</b>			
Dose (person-rem)	0.0028	0.00028	0.0031
Latent cancer fatalities	$1.4 \times 10^{-6}$	$1.4 \times 10^{-7}$	$1.6 \times 10^{-6}$
<b>Annual Dose to the Maximally Exposed Offsite Individual</b>			
Dose (millirem)	0.00033	0.000048	0.00038
Latent cancer fatality risk	$1.6 \times 10^{-10}$	$2.4 \times 10^{-11}$	$1.9 \times 10^{-10}$
Percent of natural background <sup>a</sup>	0.000092	0.000013	0.00011
<b>Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)<sup>b</sup></b>			
Dose (millirem)	0.000012	$1.2 \times 10^{-6}$	0.000013
Latent cancer fatality risk	$6.0 \times 10^{-12}$	$6.0 \times 10^{-13}$	$6.6 \times 10^{-12}$

<sup>a</sup> The annual natural background radiation level at INEEL is 360 millirem for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.

<sup>b</sup> Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

Table 4–38 summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by ANL-W workers involved with sodium-bonded spent nuclear fuel electrometallurgical and melt and dilute treatment processes. It was concluded that the average worker dose would not be different from what currently is being experienced. The estimated annual total worker population dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended over the 13 years of treatment activities, the cumulative worker population dose would be 286 person-rem, leading to a risk of 0.11 latent cancer fatalities.

**Table 4–38 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 4**

<i>Impact</i>	<i>Alternative 4</i>
<b>Worker<sup>a</sup></b>	
Total dose (person-rem per year)	22
13-year fatal cancer risk	0.11
Average worker dose (millirem per year)	60
13-year fatal cancer risk	0.00031

<sup>a</sup> The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to an involved worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year as established for all DOE activities in DOE Order N441.1.

Source: ANL 1999.

As shown in Tables 4–37 and 4–38:

- The annual dose to the maximally exposed offsite individual would be 0.00038 millirem per year, with an associated  $1.9 \times 10^{-10}$  risk per year of developing fatal cancer (or one in 5.3 billion years).
- The collective dose to the population within the 80 kilometers (50 miles) of the ANL-W facilities would be 0.0031 person-rem per year, with an associated  $1.6 \times 10^{-6}$  latent cancer fatalities per year (or one in 625,000 years).
- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one in 113 years).

*Hazardous Chemical Impacts*

It is expected that hazardous chemical impacts associated with this alternative at ANL-W will be small as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described in Section 3.2.10.2.

**4.6.4.2 Facility Accidents**

*Radiological Impacts*

Potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities associated with cleaning (sodium removal) blanket spent nuclear fuel for melt and dilute processing and treating driver spent nuclear fuel using electrometallurgical treatment are summarized and presented in this section. The detailed analysis of facility accidents, with their associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than  $10^{-7}$  per year and consequence analyses for these two events were not performed. Processing of blanket spent nuclear fuel is performed in the Hot Fuel Examination Facility; treatment of driver spent nuclear fuel is performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because driver spent nuclear fuel processing takes place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multifacility impacts of this event, and releases from both the Hot Fuel Examination Facility and the Fuel Conditioning Facility from the single seismic event. The melt and dilute processing of blanket spent nuclear fuel is performed only in the Hot Fuel Examination Facility. Melt and dilute processing of the fuel results in a greater number of accidents to be considered (waste processing-related events) in the assessment of accidents

involving blanket spent nuclear fuel at ANL-W than declad and clean operations. The multifacility impacts of the beyond-design-basis earthquake are not relevant to the blanket spent nuclear fuel melt and dilute processing which occurs in only the one facility. Therefore, only the higher frequency design-basis seismic event was analyzed. Table 4–39 presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95<sup>th</sup> percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50<sup>th</sup> percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents. (See the discussion on the involved worker in Section 4.2.4.2.) The accident risks for the same receptors are summarized in Table 4–40.

**Table 4–39 Accident Frequency and Consequences for Alternative 4**

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population within 80 kilometers (50 miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>	Dose (person- rem)	Latent Cancer Fatalities <sup>b</sup>	Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>
<b>Driver Spent Nuclear Fuel</b>							
Salt powder spill	0.01	0.00046	$2.3 \times 10^{-10}$	0.000098	$4.9 \times 10^{-8}$	$4.7 \times 10^{-7}$	$1.9 \times 10^{-13}$
Salt transfer drop	$1.0 \times 10^{-7}$	0.19	$9.5 \times 10^{-8}$	0.022	0.000011	0.073	$2.9 \times 10^{-8}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Cask drop	0.01	0.030	$1.5 \times 10^{-8}$	0.0035	$1.8 \times 10^{-6}$	0.00084	$3.4 \times 10^{-10}$
Earthquake (design-basis earthquake at Hot Fuel Examination Facility)	0.008	12	$6.0 \times 10^{-6}$	1.4	0.0007	4.7	1.9
Earthquake (beyond-design-basis earthquake)	0.00001	22,000	0.022	2,500	1.3	370	0.00015
<b>Blanket Spent Nuclear Fuel</b>							
Cask drop	0.01	0.0024	$1.2 \times 10^{-9}$	0.00028	$1.4 \times 10^{-7}$	0.000049	$2.0 \times 10^{-11}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Sodium Fire <sup>c</sup>	0.008	5.9	$3.0 \times 10^{-6}$	0.689	0.00034	0.054	$2.2 \times 10^{-8}$
Earthquake (design-basis event)	0.008	471	0.00024	56.1	0.028	15.2	$6.1 \times 10^{-6}$
Waste-handling spill	0.024	15	$7.5 \times 10^{-6}$	1.77	0.00089	0.49	$2.0 \times 10^{-7}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

<sup>c</sup> The frequency of this accident is the frequency of the facility design-basis earthquake initiating a cell fire.

**Table 4–40 Annual Cancer Risks Due to Accidents at ANL-W for Alternative 4**

<i>Accident</i>	<i>Maximally Exposed Offsite Individual<sup>a</sup></i>	<i>Population within 80 kilometers (50 miles)<sup>b</sup></i>	<i>Noninvolved Worker<sup>a</sup></i>
<b><i>Driver Spent Nuclear Fuel</i></b>			
Salt powder spill	$2.3 \times 10^{-12}$	$4.9 \times 10^{-10}$	$1.9 \times 10^{-15}$
Salt transfer drop	$9.5 \times 10^{-15}$	$1.1 \times 10^{-12}$	$2.9 \times 10^{-15}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.6 \times 10^{-9}$	$8.8 \times 10^{-11}$
Cask drop	$1.5 \times 10^{-10}$	$1.8 \times 10^{-8}$	$3.4 \times 10^{-12}$
Earthquake (design-basis earthquake)	$4.8 \times 10^{-8}$	$5.6 \times 10^{-6}$	$1.5 \times 10^{-8}$
Earthquake (beyond-design-basis earthquake)	$2.2 \times 10^{-7}$	0.000013	$1.5 \times 10^{-9}$
<b><i>Blanket Spent Nuclear Fuel</i></b>			
Cask drop	$1.2 \times 10^{-11}$	$1.4 \times 10^{-9}$	$2.0 \times 10^{-13}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.5 \times 10^{-9}$	$8.7 \times 10^{-11}$
Sodium Fire	$2.3 \times 10^{-8}$	$2.7 \times 10^{-6}$	$1.7 \times 10^{-10}$
Earthquake (design-basis event)	$1.9 \times 10^{-6}$	0.00022	$4.9 \times 10^{-8}$
Waste-handling spill	$1.8 \times 10^{-7}$	0.000021	$4.7 \times 10^{-9}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be  $1.9 \times 10^{-6}$  per year (or one in 526,300 years) and  $4.9 \times 10^{-8}$  per year (or one in 20.4 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.00022 per year (or one in 4,545 years).

#### *Hazardous Chemical Impacts*

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendix F, Section F.3.1.2, for details).

The nonradiological impacts of potential facility accidents (hazardous chemical) associated with the treatment of driver spent nuclear fuel using the electrometallurgical treatment process are summarized in Table 4–41.

#### **4.6.5 Environmental Justice**

As discussed in Section 4.6.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the 13 years of melt and dilute processing of blanket spent nuclear fuel and electrometallurgical or melt and dilute treatment of driver spent nuclear fuels would be  $2.5 \times 10^{-9}$  (or one chance in 400 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000021 (or one chance in 47,600). Radiological and nonradiological risks posed by implementation of this alternative would therefore be small regardless of the racial and ethnic composition of the population, and independent of the economic status of individuals comprising the population. Operation of electrometallurgical and melt and dilute treatment processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

**Table 4-41 Nonradiological Impacts of Accidents for Alternative 4**

<i>Accident</i>	<i>Frequency event/year</i>	<i>Receptor Location</i>	<i>Exposure</i>
Uranium-handling accident	0.01	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Design-basis earthquake	0.0002	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Sodium fire	0.00001	Noninvolved worker at 100 meters	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1
Uranium fire	0.00001	Noninvolved worker at 100 meters	Uranium: less than ERPG-1
		Maximally exposed offsite individual	Uranium: less than ERPG-1
Beyond-design-basis earthquake	0.00001	Noninvolved worker at 100 meters	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1
		Maximally exposed offsite individual	Cadmium: less than ERPG-1
			Uranium: less than ERPG-1

ERPG = Emergency Response Planning Guideline

#### 4.6.6 Waste Management

This alternative would generate process wastes from treatment operations, other associated process wastes from normal support operations, and wastes following the conclusion of operations. Process wastes would include metal and ceramic high-level radioactive wastes. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes would include the disposal of process equipment and other materials. The fuel hardware in this alternative is used as additional steel in the melt and dilute process. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of the waste types generated and their expected interim storage and final disposal locations are given in Table 4-7 (see Section 4.2.6).

Estimates of the total amount of other associated process wastes generated as a result of Alternative 4 are provided in **Table 4-42**. These values are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4-42 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of Alternative 4 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

##### *Direct Process Wastes*

For this alternative, fuel assembly hardware would be used as part of the required stainless steel to form the material ingot for disposal of the blanket spent nuclear fuel by melting. Its mass is included as part of the spent nuclear fuel disposal.

Under this alternative, metal and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metal waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products.

The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metal waste form. Both the ceramic and metal waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4–42 are for the standardized canisters required for disposal of these materials.

**Table 4–42 Amounts of Wastes Produced at ANL-W for Alternative 4<sup>a</sup>**

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
<b>Direct Process Wastes</b>		
Fuel assembly hardware (low-level radioactive waste)	0	0
High-level radioactive ceramic waste	16.3	24,400
High-level radioactive metal waste	2.0	2,500
Melt and dilute product	45.6	114,000
<b>Other Associated Process Wastes</b>		
High-level radioactive wastes	0.4	220
Low-level radioactive wastes <sup>b</sup>	650	132,000
Transuranic wastes	11.2	4,730
Mixed wastes	32.1	17,300
Sanitary wastes	4,960	$1.72 \times 10^6$
<b>Deactivation Wastes</b>		
Low-level radioactive wastes <sup>b</sup>	178	66,000
Transuranic wastes	1.6	853
Mixed wastes	5.1	3,600

<sup>a</sup> These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage are accomplished during this time period.

<sup>b</sup> The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

The metal and ceramic high-level radioactive waste and the melted blanket spent nuclear fuel generated at ANL-W would be temporarily stored for 10 to 15 years at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste, and shielding would be provided by a combination of steel storage liners storing the waste and the shielding provided by the soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste cans containing these materials would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

#### *Other Associated Process High-Level Radioactive Waste*

These high-level radioactive wastes could be generated as a result of blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this waste stream would consist of the absorbant used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single high-level radioactive waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

### *Other Associated Process Low-Level Radioactive Wastes*

These low-level radioactive wastes would be generated as a result of processing sodium-bonded spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from electrometallurgical and melt and dilute treatment processing of driver spent nuclear fuel at ANL-W that will require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total 650 cubic meters (22,955 cubic feet) represent approximately 0.6 percent of the total Radioactive Waste Management Complex disposal capacity.

### *Other Associated Process Transuranic Wastes*

These transuranic wastes would be generated by Alternative 4 from decontamination activities, repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic wastes would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant Acceptance Criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is approximately 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is approximately 9 cubic meters (318 cubic feet), which is less than 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

### *Other Associated Process Mixed Wastes*

These mixed wastes would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1996b).

### *Deactivation Wastes*

A variety of wastes would be generated as part of deactivation activities associated with processing at ANL-W. These would include process equipment and process material, such as electrorefiner cadmium from electrometallurgical treatment of driver spent nuclear fuel. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation wastes would be low-level radioactive waste, generated as a result of dismantling and disposal (electrometallurgical treatment and melt and dilute equipment). Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot hydrostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Decontamination of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. If the deactivation waste volume is generated in a single year, the wastes would represent an increase of approximately three times the annual waste generated by the treatment operations of Alternative 4. The total deactivation wastes represent an additional 30 percent over the total incidental waste requiring disposal.

#### **4.7 ALTERNATIVE 5: DECLAD AND CLEAN BLANKET FUEL AND TREAT (ELECTROMETALLURGICAL TREATMENT) DRIVER FUEL AT ANL-W; MELT AND DILUTE BLANKET FUEL AT SRS**

Under this alternative, the sodium-bonded blanket spent nuclear fuel would be declad and cleaned to remove metallic sodium at ANL-W, packaged in aluminum cans, and shipped to SRS for treatment using the melt and dilute process at Building 105-L. The melt and dilute product from the treatment process would be stored at SRS pending repackaging and transportation for disposal in a geologic repository. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The sodium-bonded driver spent nuclear fuel would be treated at ANL-W using the electrometallurgical treatment process. The process steps for the electrometallurgical treatment of driver spent nuclear fuels would be similar to those described earlier in Section 4.3 and in Appendix C. The treatment of driver spent nuclear fuel using the electrometallurgical process could start as early as 2000 and be completed by 2006 to 2007. The preparation of blanket spent nuclear fuel and its shipment to SRS could start in 2003 and be completed by 2009. Current planning at SRS has scheduled the melt and dilute process at Building 105-L for other missions (DOE 1998g). Melt and dilute process of blanket spent nuclear fuel at SRS could start around 2020, if capacity becomes available, and be completed by 2023.

##### **4.7.1 Air Quality**

###### *Nonradiological Gaseous Emissions*

It is expected that this alternative at ANL-W will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

At SRS, nonradiological air emissions result from operation of ancillary support facilities for the melt and dilute process at Building 105-L. These include: site electrical power generators, emergency diesel generators, fuel handling activities in the L-area basin, and increased vehicle emissions. The largest contributors to the emissions are the onsite electrical power generators (Bickford 1999).

The concentrations of nonradiological air pollutants attributed to this alternative at SRS are presented in **Table 4-43**, along with the total estimated site air pollutant concentrations. The concentrations for the alternative are based on information in the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g). The concentrations have been adjusted to account for the increased mass of sodium-bonded spent nuclear fuel. The total concentrations are equal to the concentration for the alternative plus the baseline concentrations from Section 3.3.3.1. The concentrations are compared to their corresponding ambient air quality standards. Only those air pollutants that are expected to be emitted under this alternative and have ambient air quality standards are presented in the table. Note that SRS has no Prevention of Significant Deterioration increment-consuming sources on site; therefore, a Prevention of Significant Deterioration increment analysis was not performed. Health effects from hazardous chemicals associated with this alternative are addressed in Section 4.7.4.1.

**Table 4-43 Nonradiological Air Quality Concentrations Associated with Alternative 5 at SRS for Comparison with Ambient Air Quality Standards**

<i>Pollutant</i>	<i>Averaging Period</i>	<i>Most Stringent Standard or Guideline (micrograms per cubic meter)</i>	<i>Alternative 5 Concentration (micrograms per cubic meter)</i>	<i>Total Site Concentration (micrograms per cubic meter)</i>
<b>Criteria Pollutants</b>				
Carbon monoxide	8 hours	10,000	0.08	631.88
	1 hour	40,000	0.51	5015.11
Nitrogen dioxide	Annual	100	less than 0.01	8.8
PM <sub>10</sub>	Annual	50	ND	4.8
	24 hours (interim)	150	ND	80.6
	24 hours (99 <sup>th</sup> percentile over 3 years)	150	ND	NA
PM <sub>2.5</sub>	3-year annual	15	NA	NA
	24 hours (98 <sup>th</sup> percentile over 3 years)	65	NA	NA
Sulfur dioxide	Annual	80	0.01	16.31
	24 hours	365	0.03	215.43
	3 hours	1,300	ND	690.2
<b>State-Regulated Pollutants</b>				
Gaseous fluoride	30 days	0.8	ND	0.11
	7 days	1.6	ND	0.11
	24 hours	2.9	ND	0.60
	12 hours	3.7	ND	
Total suspended particulates	Annual	75	less than 0.01	43.3
<b>Hazardous and Other Toxic Compounds</b>				
1,1,1-trichloroethane	24 hours	9,550	less than 0.01	less than 22.01
Benzene	24 hours	150	ND	31
Ethanolamine	24 hours	200	less than 0.01	less than 0.02
Ethyl benzene	24 hours	4,350	ND	0.12
Ethylene glycol	24 hours	650	less than 0.01	less than 0.09
Formaldehyde	24 hours	15	less than 0.01	less than 0.02
Glycol ethers	24 hours	No Standard	less than 0.01	less than 0.02
Hexachloronaphthalene	24 hours	1	less than 0.01	less than 0.02
Hexane	24 hours	900	less than 0.01	less than 0.08
Manganese	24 hours	25	ND	0.1
Methyl alcohol	24 hours	1,310	less than 0.01	less than 0.52
Methyl ethyl ketone	24 hours	14,750	less than 0.01	less than 1
Methyl isobutyl ketone	24 hours	2,050	ND	less than 0.51
Methylene chloride	24 hours	8,750	ND	1.8
Naphthalene	24 hours	1,250	less than 0.01	less than 0.02
Nitric acid	24 hours	125	ND	6.7
Phenol	24 hours	190	ND	0.03
Phosphorous	24 hours	0.5	ND	less than 0.001
Sodium hydroxide	24 hours	50	ND	0.01
Toluene	24 hours	2,000	less than 0.01	less than 1.61
Trichloroethane	24 hours	6,750	ND	1
Vinyl acetate	24 hours	176	ND	less than 0.02
Xylene	24 hours	4,350	less than 0.01	less than 3.81

NA = Not Available; ND=not detectable.

Source: Bickford et al. 1997, plus baseline concentrations from Section 3.3.3.1.

### *Radiological Gaseous Emissions*

The decladding and cleaning of the blanket spent nuclear fuel and the electrometallurgical treatment of the driver spent nuclear fuel at ANL-W would release gaseous fission products to the argon cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The tritium released in the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries ( $10^{-9}$  curies) and are released to the atmosphere through the facility stack along with krypton-85 and elemental tritium. The maximum release of radioactive gaseous emissions occurs when decladding the blanket spent nuclear fuel for packaging and shipment to SRS and electrometallurgical treatment of driver spent nuclear fuels are performed simultaneously. This simultaneous operation was estimated to occur over a three-year period starting in 2003. Based on an annual decladding throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and an electrometallurgical treatment process of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements, about 809 curies of elemental tritium and 11,860 curies of gaseous krypton-85 would be released annually to the atmosphere.

Since declad and clean fuels would be packaged and sent to SRS, some gaseous fission products would be expected in that fuel. However, it was conservatively assumed that the gaseous fission products in the blanket spent nuclear fuels also would be released to the environment during the melt and dilute process at SRS. The radiological exposures of the public and workers from radioactive emissions are presented in Section 4.7.4.

#### **4.7.2 Water Resources**

As stated in Section 4.4.2, the decladding and cleaning of blanket spent nuclear fuels and electrometallurgical treatment of driver spent nuclear fuels would not discharge any radiological chemical material to the surface or groundwater at the INEEL site. These activities also would not impact the current groundwater usage at the site. For a discussion of impacts on water resources at ANL-W, see Section 4.4.2.

The impacts on water resources from treating blanket spent nuclear fuels at Building 105-L using the melt and dilute process are described below.

##### *Surface Water*

No surface water would be used for the melt and dilute processing of blanket spent nuclear fuel at Building 105-L. Building 105-L is outside the 100-year floodplain, as shown in Figure 3-6.

##### *Nonradiological Liquid Effluent*

No nonradiological liquid effluent would be generated by melting and diluting blanket spent nuclear fuel at Building 105-L. Sanitary waste would be treated at the SRS Central Wastewater Treatment Facility and discharged through an existing NPDES-permitted outfall (G-10). Since employment would not increase as a result of processing these fuels, the treatment rates through the Central Wastewater Treatment Facility would not be affected and the requirements of the SRS NPDES permit would continue to be met (DOE 1998g).

##### *Radiological Liquid Effluent*

There are no anticipated radiological liquid effluents associated with the melt and dilute process at Building 105-L.

## **Groundwater**

Process water would not be required for the melt and dilute process at Building 105-L. Domestic water would come from groundwater. No increase in domestic water use is anticipated since no increase in employment is expected to result from the melt and dilute operation.

### *Nonradiological Liquid Effluent*

No nonradiological chemicals would be discharged to groundwater from the melt and dilute processing at Building 105-L.

### *Radiological Liquid Effluent*

No radiological liquid effluent would be discharged to groundwater from the melt and dilute process at Building 105-L.

## **4.7.3 Socioeconomics**

Under this alternative, the existing facilities at ANL-W and SRS would remain operational. No new employment or in-migration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around INEEL and SRS.

## **4.7.4 Public and Occupational Health and Safety**

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological and chemical impacts from normal operations are presented in Tables 4-44 through 4-46 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4-47 through 4-50. The impacts from hazardous chemical releases during accident conditions are similar to those presented in Section 4.5.4.1. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

### **4.7.4.1 Normal Operations**

#### *Radiological Impacts*

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel decladding and cleaning, driver spent nuclear fuel chopping, and electrorefining. All of these activities are performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public at ANL-W would result when decladding and cleaning of blanket spent nuclear fuels and treatment of driver spent nuclear fuels are performed simultaneously under this alternative. The doses from decladding and cleaning blanket spent nuclear fuels and treating driver spent nuclear fuels at ANL-W would be similar to those presented for Alternative 2 in Section 4.4.4.1.

Calculated maximum annual radiological impacts to the public are given in **Table 4-44**. The impacts are calculated for two receptor groups: the general public living within 80 kilometers (50 miles) of ANL-W and Building 105-L at SRS in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL or SRS site boundary and receiving the maximum dose). Primary contributors to doses to members of the public at ANL-W are releases of tritium gases (about 1 percent of

which was conservatively assumed to be in oxidized form) and krypton-85, which together contribute over 99.9 percent of the total calculated doses.

**Table 4–44 Annual Radiological Impacts to the Public from Operational Activities Associated With Alternative 5**

<i>Receptor</i>	<i>Alternative 5</i>	
	<i>Melt and Dilute Blanket Spent Nuclear Fuel at SRS</i>	<i>Clean Blanket Spent Nuclear Fuel and Electrometallurgical Treatment of Driver Spent Nuclear Fuel at ANL-W</i>
<b>Population Dose Within 80 Kilometers (50 Miles) in the Year 2010</b>		
Dose (person-rem)	0.0076	0.0031
Latent cancer fatalities	$3.8 \times 10^{-6}$	$1.5 \times 10^{-6}$
<b>Annual Dose to the Maximally Exposed Offsite Individual</b>		
Dose (millirem)	0.00010	0.00038
Latent cancer fatality risk	$5.0 \times 10^{-11}$	$1.9 \times 10^{-10}$
Percent of natural background <sup>a</sup>	0.000033	0.000011
<b>Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)</b>		
Dose (millirem)	0.000011 <sup>b</sup>	0.000013 <sup>c</sup>
Latent cancer fatality risk	$5.5 \times 10^{-12}$	$6.6 \times 10^{-12}$

<sup>a</sup> The annual natural background radiation level at INEEL and at SRS is 360 and 300 millirem, respectively, for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 and 215,000 person-rem, respectively.

<sup>b</sup> Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of SRS Building 105-L (L-Reactor Area) in the year 2010 (715,000).

<sup>c</sup> Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of ANL-W in the year 2010 (240,338).

The blanket spent nuclear fuel has been decontaminated and cleaned at ANL-W, where it is expected that the gaseous fission products would have been released. However, for the melt and dilute process, it is conservatively assumed that these gaseous fission products are released at SRS. The melt and dilute process is assumed to continue for three years. [Appendix E, Section E.4.4, provides the details on the treatment process duration.] To put the operational impacts into perspective, comparisons with impacts from natural background radiation also are included in the table.

**Table 4–45** summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by SRS workers involved with the melt and dilute process. The estimated annual total worker population dose would be 50 person-rem, with an average individual dose of 500 millirem per year for each of the 100 involved workers. If these estimates were projected for maximum process activities over three years, the cumulative worker population dose would be 150 person-rem, leading to a risk of 0.06 latent cancer fatalities. The estimated annual total worker population dose to decontaminate and clean blanket spent nuclear fuels and treat driver spent nuclear fuels at ANL-W is 22 person rem, as indicated in Section 4.4.4.1.

**Table 4–45 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 5**

Impact	Alternative 5	
	Operations at SRS	Operations at ANL-W
<b>Worker<sup>a</sup></b>		
Total dose (person-rem per year)	50	22
Fatal cancer risk	0.06 <sup>b</sup>	0.079 <sup>c</sup>
Average worker dose (millirem per year)	500	60
Fatal cancer risk	0.00060 <sup>b</sup>	0.00022 <sup>c</sup>

<sup>a</sup> The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to an involved worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N441.1.

<sup>b</sup> Operations at SRS to treat blanket spent nuclear fuel using melt and dilute at Building 105-L are performed over three years.

<sup>c</sup> Operations at ANL-W to declad and clean blanket spent nuclear fuels and treat driver spent nuclear fuels are performed over nine years.

Sources: ANL 1999, DOE 1998g.

As shown in Tables 4–44 and 4–45:

- The annual dose to the maximally exposed offsite individual would be 0.00038 millirem per year, with an associated  $1.9 \times 10^{-10}$  risk per year of developing fatal cancer (or one in 5.3 billion years).
- The collective dose to the population within the 80 kilometers (50 miles) of the ANL-W facilities would be 0.0031 person-rem per year, with an associated  $1.6 \times 10^{-6}$  latent cancer fatalities per year (or one in 625,000 years).
- The collective dose to ANL-W facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities per year (or one in 113 years).
- The annual dose to the maximally exposed offsite individual from melt and dilute processing at Building 105-L would be 0.00010 millirem per year, with an associated  $5.0 \times 10^{-11}$  risk per year of developing fatal cancer (or one in 20 billion years).
- The annual collective dose to the population within 80 kilometers (50 miles) of Building 105-L would be 0.0076 person-rem per year, with an associated  $3.8 \times 10^{-6}$  latent cancer fatalities per year (or one in 263,100 years).
- The collective dose to Building 105-L facility workers would be 50 person-rem per year, with an associated 0.020 latent cancer fatalities (or one in 50 years).

#### *Hazardous Chemical Impacts*

It is expected that the hazardous chemical impacts associated with this alternative at ANL-W will be small, as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described in Section 3.2.10.2.

For SRS, both carcinogenic and noncarcinogenic health effects to the public were assessed from exposure to hazardous chemicals. It was assumed that, under normal operating conditions, the primary exposure pathway for members of the public would be via air emissions.

The 24-hour concentrations provided in Section 4.7.1 were converted to annual concentrations by using the appropriate regulatory scaling factor of 0.125 based on South Carolina’s Air Quality Modeling Guidelines (SCDHEC 1993). The annual concentration for each noncarcinogenic chemical was divided by the corresponding inhalation reference concentration to estimate the hazard quotient for each chemical. The hazard quotients were summed to give the hazard index from all noncarcinogenic chemicals for this alternative. A hazard index less than one indicates that adverse health effects from noncancer-causing agents are not expected. For carcinogens, the annual concentration was multiplied by the unit cancer risk to estimate the increased cancer risk from that chemical. Hazardous chemical health effects to the public and noninvolved worker are summarized in **Table 4–46** and **4–47**, respectively.

**Table 4–46 Hazardous Chemical Impacts to the Public from Operational Activities at SRS for Alternative 5**

<i>Chemical</i>	<i>Annual Concentration (milligrams per cubic meter)</i>	<i>Reference Concentration Inhalation (milligrams per cubic meter)</i>	<i>Unit Cancer Risk (risk per milligram per cubic meter)</i>	<i>Hazard Quotient</i>	<i>Cancer Risk</i>
Benzene	ND	None	0.0078	None	ND
Ethyl benzene	ND	1	None	ND	None
Formaldehyde	$1.3 \times 10^{-6}$	None	0.013	None	$1.6 \times 10^{-8}$
Hexane	$1.3 \times 10^{-6}$	0.2	None	$6.3 \times 10^{-6}$	None
Manganese	ND	0.00005	None	ND	None
Methyl ethyl ketone	$1.3 \times 10^{-6}$	1	None	$1.3 \times 10^{-6}$	None
Methylene chloride	ND	None	0.00047	None	ND
Naphthalene	$1.3 \times 10^{-6}$	0.003	None	0.00042	None
Toluene	$1.3 \times 10^{-6}$	0.4	None	$3.1 \times 10^{-6}$	None
Vinyl acetate	ND	0.2	None	ND	None
Hazard Index				0.00043	None

Source: EPA 1999.  
Key: ND = Not detectable.

**Table 4–47 Hazardous Chemical Impacts to the Noninvolved Worker from Operational Activities at SRS for Alternative 5**

<i>Chemical</i>	<i>Annual Concentration (milligrams per cubic meter)</i>	<i>Reference Concentration Inhalation (milligram per cubic meter)</i>	<i>Unit Cancer Risk (risk per milligram per cubic meter)</i>	<i>Hazard Quotient</i>	<i>Cancer Risk</i>
Benzene	ND	None	0.0078	None	ND
Ethyl benzene	ND	1	None	ND	None
Formaldehyde	0.0035	None	0.013	None	0.000046
Hexane	0.0035	0.2	None	0.018	None
Manganese	ND	0.000050	None	ND	None
Methyl ethyl ketone	0.0035	1	None	0.0035	None
Methylene chloride	ND	None	0.00047	None	ND
Naphthalene	$1.9 \times 10^{-12}$	0.003	None	$6.3 \times 10^{-10}$	None
Toluene	0.0035	0.4	None	0.0088	None
Vinyl acetate	ND	0.2	None	ND	None
Hazard Index				0.00043	None

Source: EPA 1999.  
Key: ND = Not detectable above background levels.

#### 4.7.4.2 Facility Accidents

##### *Radiological Impacts*

The potential radiological impacts to the public and a noninvolved onsite worker resulting from accidents during decladding and cleaning and melting and diluting the blanket spent nuclear fuel elements, and from electrometallurgical treatment of driver spent nuclear fuel operational activities at ANL-W and SRS, are summarized and presented in this section. The detailed analysis of facility accidents, with associated assumptions, is presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash and criticality events were determined to have an occurrence frequency of less than  $10^{-7}$  per year, and consequence analyses for these two events were not performed. Processing of the blanket spent nuclear fuel is performed in the Hot Fuel Examination Facility; treatment of the driver spent nuclear fuel is performed in both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Because the processing of the driver spent nuclear fuel takes place in both of these facilities, the beyond-design-basis earthquake event is assessed for the driver spent nuclear fuel, taking into account the multi-facility impacts of this event. The decladding and cleaning of the blanket spent nuclear fuel is performed only in the Hot Fuel Examination Facility. The multi-facility impacts of the beyond-design-basis earthquake are not relevant to this blanket spent nuclear fuel processing. Therefore, only the higher frequency design-basis seismic event was analyzed. **Table 4-48** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 foot) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95<sup>th</sup> percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50<sup>th</sup> percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents (see the discussion on the involved worker in Section 4.2.4.2). The accident risks for the same receptors are summarized in **Table 4-49**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be  $2.2 \times 10^{-7}$  per year (or one in 4.5 million years) and  $1.5 \times 10^{-8}$  per year (or one in 66.7 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.000013 per year (or one in 76,920 years).

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during melt and dilute operational activities at SRS are summarized below. The detailed analysis of facility accidents, with the associated assumptions, is presented in Appendix F. **Table 4-50** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual, the offsite population residing within 80 kilometers (50 miles) of the facility, and a noninvolved worker located 100 meters (330 feet) to 300 meters (980 feet) from the facility. The 300-meter (980-foot) distance leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

**Table 4–48 Accident Frequency and Consequences at ANL-W for Alternative 5**

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population within 80 kilometers (50 miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>	Dose (person- rem)	Latent Cancer Fatalities <sup>b</sup>	Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>
<b>Driver Spent Nuclear Fuel</b>							
Salt powder spill	0.01	0.00046	$2.3 \times 10^{-10}$	0.000098	$4.9 \times 10^{-8}$	$4.7 \times 10^{-7}$	$1.9 \times 10^{-13}$
Salt transfer drop	$1.0 \times 10^{-7}$	0.19	$9.5 \times 10^{-8}$	0.022	0.000011	0.073	$2.9 \times 10^{-8}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Cask drop	0.01	0.030	$1.5 \times 10^{-8}$	0.0035	$1.8 \times 10^{-6}$	0.00084	$3.4 \times 10^{-10}$
Earthquake (design-basis earthquake at Hot Fuel Examination Facility)	0.008	12	$6.0 \times 10^{-6}$	1.4	0.0007	4.7	$1.9 \times 10^{-6}$
Earthquake (beyond-design-basis earthquake)	0.00001	22,000	0.022	2,500	1.3	370	0.00015
<b>Blanket Spent Nuclear Fuel</b>							
Cask drop	0.01	0.0024	$1.2 \times 10^{-9}$	0.00028	$1.4 \times 10^{-7}$	0.000049	$2.0 \times 10^{-11}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Sodium fire <sup>c</sup>	0.008	5.9	$3.0 \times 10^{-6}$	0.689	0.00034	0.054	$2.2 \times 10^{-8}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

<sup>c</sup> The frequency for this accident is the frequency for the facility design-basis earthquake-initiating cell fire.

**Table 4–49 Annual Cancer Risks Due to Accidents at ANL-W for Alternative 5**

Accident	Maximally Exposed Offsite Individual <sup>a</sup>	Population within 80 kilometers (50 miles) <sup>b</sup>	Noninvolved Worker <sup>a</sup>
<b>Driver Spent Nuclear Fuel</b>			
Salt powder spill	$2.3 \times 10^{-12}$	$4.9 \times 10^{-10}$	$1.9 \times 10^{-15}$
Salt transfer drop	$9.5 \times 10^{-15}$	$1.1 \times 10^{-12}$	$2.9 \times 10^{-15}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.6 \times 10^{-9}$	$8.8 \times 10^{-11}$
Cask drop	$1.5 \times 10^{-10}$	$1.8 \times 10^{-8}$	$3.4 \times 10^{-12}$
Earthquake (design-basis earthquake)	$4.8 \times 10^{-8}$	$5.6 \times 10^{-6}$	$1.5 \times 10^{-8}$
Earthquake (beyond-design-basis earthquake)	$2.2 \times 10^{-7}$	0.000013	$1.5 \times 10^{-9}$
<b>Blanket Spent Nuclear Fuel</b>			
Cask drop	$1.2 \times 10^{-11}$	$1.4 \times 10^{-9}$	$2.0 \times 10^{-13}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.5 \times 10^{-9}$	$8.7 \times 10^{-11}$
Sodium fire	$2.3 \times 10^{-8}$	$2.7 \times 10^{-6}$	$1.7 \times 10^{-10}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

The dose to the maximally exposed offsite individual was calculated for the 95<sup>th</sup> percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50<sup>th</sup> percentile meteorological conditions. DOE did not quantitatively estimate the facility worker population dose due to accidents. The accident risks for the same receptors are summarized in **Table 4–51**.

**Table 4–50 Accident Frequency and Consequences at SRS for Alternative 5**

Accident	Frequency (event per year)	Maximally Exposed Offsite Individual		Population within 80 kilometers (50 miles)		Noninvolved Worker	
		Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>	Dose (person-rem)	Latent Cancer Fatalities <sup>b</sup>	Dose (millirem)	Latent Cancer Fatality Risk <sup>a</sup>
Waste-handling spill	0.024	2.1	$1.1 \times 10^{-6}$	3.6	0.0018	0.17	$6.8 \times 10^{-8}$
Loss of power	0.006	2100	0.0011	3500	1.8	140	0.000056
Loss of cooling water	0.05	120	0.000060	500	0.25	1.3	$5.2 \times 10^{-7}$
Fire	0.075	86	0.000043	140	0.07	6.3	$2.5 \times 10^{-6}$
Criticality	NA	NA	NA	NA	NA	NA	NA
Aircraft crash	NA	NA	NA	NA	NA	NA	NA

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

Key: NA = Not analyzed, because the frequency is estimated to be less than  $10^{-7}$  per year (see Appendix F for details).

**Table 4–51 Annual Cancer Risks of Accidents at SRS for Alternative 5**

Accident	Maximally Exposed Offsite Individual <sup>a</sup>	Population within 80 kilometers (50 miles) <sup>b</sup>	Noninvolved Worker <sup>a</sup>
Waste-handling spill	$2.6 \times 10^{-8}$	0.000043	$1.6 \times 10^{-9}$
Loss of power	$6.6 \times 10^{-6}$	0.011	$3.4 \times 10^{-7}$
Loss of cooling water	$3.0 \times 10^{-6}$	0.013	$2.6 \times 10^{-8}$
Fire	$3.2 \times 10^{-6}$	0.0053	$1.9 \times 10^{-7}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

For accidents at SRS, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and to a noninvolved worker would be  $6.6 \times 10^{-6}$  per year (or one in 151,500 years) and  $3.4 \times 10^{-7}$  per year (or one in 2.9 million years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.013 per year (or one in 77 years).

#### Hazardous Chemical Impacts

The impacts of accidents involving hazardous chemicals for this alternative are the same as those described in Section 4.5.4.2 for Alternative 3: Electrometallurgical treatment of driver spent nuclear fuel at ANL-W and PUREX processing of blanket spent nuclear fuel at SRS.

#### 4.7.5 Environmental Justice

As discussed in Section 4.7.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the three years of melt and dilute treatment of blanket spent nuclear fuel at SRS and the nine years of combined operations to clean blanket spent nuclear fuels and electrometallurgically treat driver spent nuclear fuels at ANL-W would be  $1.7 \times 10^{-9}$  (or one chance in 588 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000014 (or one chance in 74,100). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small, regardless of the racial and ethnic composition of the population and independent of the economic status of individuals comprising the

population. Operation of melt and dilute treatment at SRS and electrometallurgical processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

#### 4.7.6 Waste Management

##### ANL-W

This alternative would generate process wastes from treatment operations, other associated process wastes from normal support operations, and deactivation wastes following the conclusion of operations. Process wastes would include fuel hardware and metal and ceramic high-level radioactive wastes. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types generated at ANL-W and their expected interim storage and final disposal locations are given in Table 4–7 (see Section 4.2.6).

Estimates of the total amount of other associated process wastes generated as a result of Alternative 5 are provided in **Table 4–52**. These values are based on an evaluation of waste forecasts from ANL-W that accounted only for the fraction of total ANL-W waste that would be attributable to the treatment of sodium-bonded spent nuclear fuel. The values in Table 4-52 are for disposal and account for volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of Alternative 5 could be volume-reduced at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

**Table 4–52 Amounts of Wastes Produced at ANL-W for Alternative 5<sup>a</sup>**

Waste Stream	Waste Quantities	
	Volume (cubic meters)	Mass (kilograms)
<b>Direct Process Wastes</b>		
Fuel assembly hardware (low-level radioactive waste)	37.5	13,100
High-level radioactive ceramic waste	16.3	24,400
High-level radioactive metal waste	2.0	2,500
Spent nuclear fuel	0	0
<b>Other Associated Process Wastes</b>		
High-level radioactive wastes	0.4	220
Low-level radioactive wastes <sup>b</sup>	555	113,000
Transuranic wastes	5.1	3,800
Mixed wastes	27.5	14,800
Sanitary wastes	4,960	1.7 × 10 <sup>6</sup>
<b>Deactivation Wastes</b>		
Low-level radioactive wastes <sup>b</sup>	178	60,000
Transuranic wastes	1.6	853
Mixed wastes	9.1	3,400

<sup>a</sup> These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage are accomplished during this time period.

<sup>b</sup> The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

### *Direct Process Wastes*

For this alternative, fuel hardware would be removed from the fuel elements in the Fuel Conditioning Facility air cell and disposed of as low-level radioactive waste. These components are primarily stainless steel materials that contain short-lived radionuclides. This waste stream has been produced at ANL-W for many years and would be handled, as in the past, according to DOE Orders and ANL-W waste management procedures. In addition, the blanket spent nuclear fuel cladding is included in the fuel hardware stream.

Under this alternative, metal and ceramic high-level radioactive waste would be a primary product of the electrometallurgical treatment of driver spent nuclear fuel. The salt removed from the electrorefiners would contain the majority of fission products and transuranics from the spent nuclear fuel. This removed salt would be packaged and transferred to the Hot Fuel Examination Facility for processing into ceramic waste. The metal waste form would consist primarily of stainless steel cladding hulls containing the noble metal fission products. The hulls would be removed from the electrorefiner and packaged for shipment to the Hot Fuel Examination Facility for processing into the metal waste form. Both the ceramic and metal waste would be categorized as high-level radioactive waste. The volumes of waste forms provided in Table 4-52 are for the standardized canisters required for disposal of these materials.

The metal and ceramic high-level radioactive waste generated as a result of electrometallurgical treatment of driver spent nuclear fuel at ANL-W would be temporarily stored for 10 to 15 years at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste, and shielding would be provided by a combination of steel storage liners storing the waste and the shielding provided by the soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste cans containing the metal and ceramic high-level radioactive waste would be removed from storage, shipped to the INEEL Dry Transfer Facility, and prepared for shipment to the repository.

### *Other Associated Process High-Level Radioactive Wastes*

These high-level radioactive wastes could be generated as a result of blanket spent nuclear fuel processing at ANL-W and SRS. This would result from activities in the Hot Fuel Examination Facility (at ANL-W) and Building 105-L (at SRS). Material in this waste stream would consist of the absorbant used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single high-level radioactive waste canister. Conservatively the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

### *Other Associated Process Low-Level Radioactive Wastes*

These low-level radioactive wastes would be generated as a result of decladding and cleaning blanket spent nuclear fuel and electrometallurgically treating driver spent nuclear fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste resulting from decladding and cleaning of blanket spent nuclear fuel and electrometallurgical treating driver spent nuclear fuel at ANL-W that will require disposal (after volume reduction) would be approximately 50 cubic meters per year (1,766 cubic feet per year). This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the

total of 555 cubic meters (19,600 cubic feet) represents approximately 0.7 percent of the total Radioactive Waste Management Complex disposal capacity.

#### *Other Associated Process Transuranic Wastes*

These transuranic wastes would be generated by Alternative 5 from decontamination activities, repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic wastes would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated at ANL-W would be packaged and certified in accordance with the Waste Isolation Pilot Plant acceptance criteria prior to transport to the Waste Isolation Pilot Plant facility. The transuranic waste generated would amount to approximately 1 cubic meter per year (35 cubic feet per year), which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is approximately 9 cubic meters (318 cubic feet), which is approximately 0.005 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

#### *Other Associated Process Mixed Wastes*

These mixed wastes would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment have been identified in the INEEL Site Treatment Plan (DOE 1996b).

#### *Deactivation Wastes*

A variety of wastes would be generated as part of deactivation activities at ANL-W. These would include process equipment and process material such as electrorefiner salt and cadmium from electrometallurgical treatment of driver spent nuclear fuel. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation wastes would be low-level radioactive wastes generated as a result of equipment dismantling and disposal. Components of electrometallurgical treatment of the driver spent nuclear fuel that would require disposition include two electrorefiners, two hot hydrostatic presses, and two V-mixers, as well as other components such as the grinder/crusher. Deactivation of these components would generate additional mixed, transuranic, and low-level radioactive waste that would require management.

The deactivation waste volume is generated over a period of two years. The total deactivation wastes represent an additional 30 percent over the total incidental waste requiring disposal.

**SRS**

The melt and dilute process at SRS would generate process wastes from treatment operations and other associated process wastes from support operations. Process wastes would include metallic high-level radioactive waste. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. The incidental wastes include low-level radioactive wastes, transuranic wastes, and mixed wastes. All of the waste streams would be categorized according to existing DOE Orders and SRS waste management procedures. The anticipated categorization of the waste types and their expected interim storage and final disposal locations are given in **Table 4-53**.

**Table 4-53 Summary of Waste Material Categories at SRS for Alternative 5**

<i>Waste Stream</i>	<i>Category</i>	<i>Interim Storage</i>	<i>Final Disposal</i>
<b>Process Wastes</b>			
Metallic waste form Off-gas filters	Conditioned spent nuclear fuel High-level radioactive waste	L-Area L-Area	Offsite (proposed geologic) repository Offsite (proposed geologic) repository
<b>Other Associated Process Wastes</b>			
Less than 100 nanocuries per gram transuranic waste	Low-level radioactive waste	None	Low-activity waste vaults Waste Isolation Pilot Plant
Greater than 100 nanocuries per gram transuranic waste	Transuranic waste	Transuranic waste storage pads	Offsite
Contaminated	Mixed waste	Mixed waste storage buildings	Offsite

Estimates of the amounts of wastes generated as a result of the melt and dilute alternative at SRS are provided in **Table 4-54**. These values are based on an evaluation of waste forecasts that accounts only for the fraction of total waste that would be attributable to processing the blanket spent nuclear fuel pins under the melt and dilute alternative.

**Table 4-54 Amounts of Wastes Generated at SRS for Alternative 5**

<i>Waste Stream</i>	<i>Total Waste Generated (cubic meters)<sup>a</sup></i>
<b>Direct Process Wastes</b>	
Canisters of melt and dilute products	76 (189 canisters)
Liquid high-level radioactive wastes <sup>b</sup>	30 <sup>c</sup>
Saltstone <sup>b</sup>	78
<b>Other Associated Process Wastes</b>	
Low-level radioactive waste	1,320 <sup>d</sup>
Transuranic waste	16.5
Mixed waste	3

<sup>a</sup> Except for the number of canisters of melt and dilute products, the values given are estimated based on the heavy metal mass ratio of similar material processed at SRS (20 metric tons of heavy metal), and provided in DOE 1998g.

<sup>b</sup> These are secondary process wastes, high-level radioactive wastes.

<sup>c</sup> This is a liquid high-level radioactive waste volume which results in about one Defense Waste Processing facility borosilicate glass high-level radioactive waste canister or a solid high-level radioactive waste volume of 0.62 cubic meters.

<sup>d</sup> Assuming a volume reduction factor of 4, the estimated disposal volume would be about 330 cubic meters (11,650 cubic feet).

As indicated in the following waste-type discussions, the amounts of wastes associated with this processing alternative are relatively small compared to onsite and offsite management capacities.

#### *Direct Process Wastes*

During the melt and dilute process, high-level radioactive waste ingots would be the primary product. This waste would be temporarily stored in L-Area prior to ultimate disposition in an offsite (proposed geologic) repository. In addition, some high-level radioactive wastes are generated from cleaning the off-gas filter system, which contains cesium, tellurium, and other isotopes volatilized during the melt and dilute process. The high-level radioactive waste would be processed in the Defense Waste Processing Facility to yield vitrified high-level radioactive waste and saltstone. The vitrified high-level radioactive waste would be temporarily stored at the Defense Waste Processing Facility pending ultimate disposal in an offsite (proposed geologic) repository. The saltstone is a cement form of low-level radioactive waste that is generated as a by-product of SRS high-level radioactive waste tank form operations. The saltstone would be disposed of on site in the Z-Area Saltstone Vaults. The volume of this saltstone would be about 0.0070 percent of the 1.11 million-cubic meter (39.2 million-cubic foot) disposal capacity of the low-activity waste vaults.

#### *Other Associated Process Low-Level Radioactive Wastes*

These low-level radioactive wastes would be generated during the melt and dilute process. The volume of low-level radioactive waste resulting from this alternative (after volume reduction) would be about 1.1 percent of the total 30,500-cubic meter (1.08 million-cubic foot) disposal capacity of the low-activity waste vaults.

#### *Other Associated Process Transuranic Wastes*

The volume of transuranic waste generated during the melt and dilute process would be about 0.01 percent of the current 168,500-cubic meter (5.95 million-cubic foot) limit for the Waste Isolation Pilot Plant (DOE 1997).

#### *Other Associated Process Mixed Wastes*

These mixed wastes generated during the melt and dilute process would be temporarily stored on site in the Mixed Waste Storage Buildings prior to eventual offsite disposal. The volume of this waste would be about 0.16 percent of the 1,900-cubic meter (67,100-cubic foot) storage capacity of these storage buildings.

### **4.8 ALTERNATIVE 6: MELT AND DILUTE BLANKET AND DRIVER FUEL AT ANL-W**

Under this alternative, sodium-bonded blanket and driver spent nuclear fuel would be treated using the melt and dilute process at ANL-W. The melt and dilute products produced from this treatment process would be stored at the Radioactive Scrap and Waste Facility pending repackaging and transportation for disposal in a geologic repository. Both blanket and driver spent nuclear fuels would be cleaned to remove metallic sodium to the extent possible. The removed sodium would be stabilized using an oxidation/carbonation process (ANL 1999). The treatment of driver and blanket spent nuclear fuel by melt and dilute at ANL-W could start in 2005 and could be completed by 2015.

#### **4.8.1 Air Quality**

##### *Nonradiological Gaseous Emissions*

It is expected that this alternative at ANL will have a small impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Baseline air quality concentrations are presented in Section 3.2.3.1.

##### *Radiological Gaseous Emissions*

The cleaning of the blanket and driver spent nuclear fuels and the melt and dilute treatment of these fuels would release gaseous fission products to the hot argon cell environment. Krypton-85 and elemental tritium are the most prevalent radioactive gaseous fission products that would be released to the environment. The tritium released into the cell would not be oxidized due to a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries ( $10^{-9}$  curies) and are released to the atmosphere through the facility stack, along with krypton-85 and elemental tritium. The maximum release of radioactive gases occurs when chopping of blanket and driver spent nuclear fuels to remove metallic sodium is performed simultaneously. This simultaneous operation could occur over a two-year period during the estimated 10 years of operation, starting in 2003. Based on an annual processing throughput of 10 metric tons of heavy metal of blanket spent nuclear fuel elements and about 1.7 metric tons of heavy metal of driver spent nuclear fuel elements, about 2,162 curies of elemental tritium and 32,650 curies of gaseous krypton-85 would be released annually to the atmosphere.

#### **4.8.2 Water Resources**

##### *Surface Water*

No surface water is used at ANL-W. The facilities at ANL-W are not expected to be reached by flood waters from Big Lost River, as shown in Figure 3–3.

##### *Nonradiological Liquid Effluent*

There are no discharges to the surface waters at ANL-W, except for discharges of nonhazardous liquid waste to the sewage pond and the industrial waste pond. Big Lost River, Little Lost River, and Birch Creek would not be impacted by activities associated with the melt and dilute treatment process. Current operating and monitoring practices would continue for NPDES stormwater and liquid effluent discharges associated with facilities at ANL-W.

During fuel treatment and associated activities, some hazardous materials may be used inside buildings. To prevent potential releases to surface or subsurface waters resulting from spills of hazardous materials used in buildings, these facilities are designed, constructed, and maintained to contain these materials. Double-contained pipes, leak detection, and secondary containment of tanks are some of the features to prevent hazardous materials from release to the environment. Following existing written procedures, spill containment and cleanup equipment is present in areas where hazardous materials are stored or used (DOE 1996a).

##### *Radiological Liquid Effluent*

No radiological liquid effluent generated by melt and dilute treatment process operations would be discharged to surface water at ANL-W.

## **Groundwater**

Under this alternative at ANL-W, there would be little change in groundwater consumption for domestic use since there is little change expected in the number of workers. Water consumption for the melt and dilute treatment process operations would not impact the current water usage at ANL-W. The current water usage at ANL-W is 188 million liters per year (49.6 million gallons per year).

### *Nonradiological Liquid Effluent*

No nonradiological liquid effluent generated by melt and dilute treatment process operations would be discharged to groundwater.

### *Radiological Liquid Effluent*

No radiological liquid effluent generated by melt and dilute treatment process operations would be discharged to groundwater.

## **4.8.3 Socioeconomics**

Under this alternative, the existing facilities at ANL-W would remain operational. No new employment or immigration of workers would be required. Thus, there would be no additional impacts on the socioeconomic conditions in the region around INEEL.

## **4.8.4 Public and Occupational Health and Safety**

The assessments of potential incremental radiological and chemical impacts associated with this alternative are presented in this section. Summaries of radiological impacts from normal operations are presented in Tables 4-55 and 4-56 for the public and workers, respectively. The radiological impacts from a spectrum of hypothetical accident scenarios are provided in Tables 4-57 and 4-58. The impacts from hazardous chemical releases during accident conditions are presented in Table 4-59. Background information on the effects of radiation on human health and safety is presented in Appendix E, Section E.2.

### **4.8.4.1 Normal Operations**

#### *Radiological Impacts*

Under this alternative, radioactive releases would occur during sodium-bonded blanket spent nuclear fuel and driver spent nuclear fuel cleaning and melt and dilute processes. All of these activities are performed in the argon cell. Appendix E, Sections E.3, E.4.1, and E.4.2, details the method and assumptions used for calculating the impacts of normal operational radiological releases on the public health and safety. The maximum annual dose to the public would result when both blanket and driver spent nuclear fuels are treated simultaneously under this alternative. Appendix E, Section E.4.5, provides the details on treatment process duration and throughputs for each fuel type.

Calculated maximum annual radiological impacts to the public are given in **Table 4-55**. The impacts are calculated for two receptor groups: the general public living within 80 kilometers (50 miles) of ANL-W in the year 2010, and a maximally exposed offsite individual (a member of the public assumed to be residing at the INEEL site boundary and receiving the maximum dose). Primary contributors to doses to members of the public are releases of tritium gases (about 1 percent of which were conservatively assumed to be in oxidized form) and krypton-85; together they contribute over 99.9 percent of the total calculated doses. To put the

operational impacts into perspective, comparisons with impacts from natural background radiation are also included in the table.

**Table 4–55 Annual Radiological Impacts to the Public from Operational Activities Associated With Alternative 6**

Receptor	Alternative 6		
	Melt and Dilute Blanket Spent Nuclear Fuel at ANL-W	Melt and Dilute Driver Spent Nuclear Fuel at ANL-W	Total
<b>Population Dose Within 80 Kilometers (50 Miles) in the Year 2010</b>			
Dose (person-rem)	0.00028	0.012	0.012
Latent cancer fatalities	$1.4 \times 10^{-7}$	$6.0 \times 10^{-6}$	$6.0 \times 10^{-6}$
<b>Annual Dose to the Maximally Exposed Offsite Individual</b>			
Dose (millirem)	0.000048	0.0019	0.002
Latent cancer fatality risk	$2.4 \times 10^{-11}$	$9.5 \times 10^{-10}$	$1.0 \times 10^{-7}$
Percent of natural background <sup>a</sup>	0.000013	0.00053	0.00054
<b>Annual Dose to the Average Individual Within 80 Kilometers (50 Miles)<sup>b</sup></b>			
Dose (millirem)	$1.2 \times 10^{-6}$	0.00005	0.000051
Latent cancer fatality risk	$6.0 \times 10^{-13}$	$2.5 \times 10^{-11}$	$2.6 \times 10^{-11}$

<sup>a</sup> The annual natural background radiation level at INEEL is 360 millirem for the average individual; the population within 80 kilometers (50 miles) in the year 2010 would receive 86,500 person-rem.

<sup>b</sup> Obtained by dividing the population dose by the number of people projected to live within 80 kilometers (50 miles) of the ANL-W in the year 2010 (240,338).

**Table 4–56** summarizes worker population doses. Occupational doses were estimated by examining the type and duration of various operations performed by the ANL-W workers involved with the melt and dilute treatment processes. It was concluded that the average worker dose would not be different from that currently being experienced. The estimated annual total worker population dose would be 22 person-rem, with an average individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates were extended out over the 10 years of treatment activities, the cumulative worker population dose would be 220 person-rem, leading to a risk of 0.088 latent cancer fatalities.

**Table 4–56 Annual and Total Radiological Impacts to Workers From Operational Activities Associated With Alternative 6**

Impact	Alternative 6
<b>Worker<sup>a</sup></b>	
Total dose (person-rem per year)	22
10-year fatal cancer risk	0.088
Average worker dose (millirem per year)	60
10-year fatal cancer risk	0.00024

<sup>a</sup> The regulatory dose limit for an individual worker is 5,000 millirem per year (10 CFR 835). However, the maximum annual dose to an involved worker would be kept below the DOE Administrative Control Level of 2,000 millirem per year, as established for all DOE activities in DOE Order N441.1.

Source: ANL 1999.

As shown in Tables 4-55 and 4-56:

- The annual dose to the maximally exposed offsite individual would be 0.00074 millirem per year, with an associated  $3.7 \times 10^{-10}$  risk per year of developing fatal cancer (or one in 2.7 billion years).

- The collective dose to the population within 80 kilometers of the ANL-W facilities would be 0.0044 person-rem per year, with an associated  $2.2 \times 10^{-6}$  latent cancer fatalities per year (or one in 454,000 years).
- The collective dose to facility workers would be 22 person-rem per year, with an associated 0.0088 latent cancer fatalities (or one in 113 years).

#### *Hazardous Chemical Impacts*

It is expected that hazardous chemical impacts associated with this alternative at ANL-W will be small, as any emissions of hazardous chemicals would be very low (ANL 1999). The existing chemical environment is described in Section 3.2.10.2.

#### **4.8.4.2 Facility Accidents**

##### *Radiological Impacts*

The potential radiological impacts to the public and a noninvolved onsite worker due to accidents during operational activities related to melt and dilute processing of fuel elements are summarized and presented in this section. The detailed analysis of facility accidents and the associated assumptions are presented in Appendix F. The detailed analysis considered a wide spectrum of potential accident scenarios, including fire, spills, criticality, earthquake, and aircraft crash. The aircraft crash event was determined to have an occurrence frequency of less than  $10^{-7}$  per year, and consequence analyses for this event were not performed. Double-batching of the driver spent nuclear fuel was determined potentially to result in a criticality event (see Appendix F), and this event was analyzed for the driver spent nuclear fuel only. Processing of the blanket and driver spent nuclear fuels is performed in the Hot Fuel Examination Facility. The multi-facility impacts of the beyond-design-basis earthquake are not relevant to processing of the fuel under this option. Therefore, only the higher frequency design-basis seismic event was analyzed. **Table 4-57** presents the frequencies and consequences of the postulated set of accidents to the maximally exposed offsite individual; the offsite population residing within 80 kilometers (50 miles) of the facility; and a noninvolved worker located 100 meters (330 feet) to 230 meters (755 feet) from the facility. The 230-meter (755-foot) distance is the ANL-W bus staging area, which leads to a higher dose to the noninvolved worker for the scenarios with elevated releases.

The dose to the maximally exposed offsite individual was calculated for the 95<sup>th</sup> percentile meteorological conditions. The doses to the population and the noninvolved worker were calculated using 50<sup>th</sup> percentile meteorological conditions. DOE did not quantitatively estimate the involved worker dose due to accidents. [See the discussion on the involved worker in Section 4.2.4.2.] The accident risks for the same receptors are summarized in **Table 4-58**.

For accidents at ANL-W, the highest risk of a latent cancer fatality to the maximally exposed offsite individual and a noninvolved worker would be 0.000076 per year (or one in 13,160 years) and  $2.7 \times 10^{-6}$  per year (or one in 370,400 years), respectively. The increased number of latent cancer fatalities in the surrounding population would be 0.0088 per year (or one in 113 years).

**Table 4–57 Accident Frequency and Consequences for Alternative 6**

Accident	Frequency (event/yr)	Maximally Exposed Offsite Individual		Population within 80 km (50 mi)		Noninvolved Worker	
		Dose (mrem)	Latent Cancer Fatality Risk <sup>a</sup>	Dose (person-rem)	Latent Cancer Fatalities <sup>b</sup>	Dose (mrem)	Latent Cancer Fatality Risk <sup>a</sup>
<b>Driver Spent Nuclear Fuel</b>							
Waste-handling spill	0.024	597	0.00030	70.8	0.035	26.7	0.000011
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Cask Drop	0.01	0.030	$1.5 \times 10^{-8}$	0.0035	$1.8 \times 10^{-6}$	0.00084	$3.4 \times 10^{-10}$
Earthquake (DBE)	0.008	19000	0.0095	2200	1.1	840	0.00034
Sodium Fire <sup>c</sup>	0.008	282	0.00014	33	0.016	2.6	$1.0 \times 10^{-6}$
Criticality	0.003	0.52	$2.6 \times 10^{-7}$	0.085	0.000043	0.47	$1.9 \times 10^{-7}$
<b>Blanket Spent Nuclear Fuel</b>							
Waste-handling spill	0.024	14.9	$7.5 \times 10^{-6}$	1.77	0.00089	0.49	$2.0 \times 10^{-7}$
Transuranic waste fire	0.001	0.059	$3.0 \times 10^{-8}$	0.0071	$3.6 \times 10^{-6}$	0.22	$8.8 \times 10^{-8}$
Cask Drop	0.01	0.0024	$1.2 \times 10^{-9}$	0.00028	$1.4 \times 10^{-7}$	0.000049	$2.0 \times 10^{-11}$
Earthquake (DBE)	0.008	472	0.00024	56.1	0.028	15.3	$6.1 \times 10^{-6}$
Sodium Fire <sup>c</sup>	0.008	5.9	$3.0 \times 10^{-6}$	0.689	0.00034	0.054	$2.2 \times 10^{-8}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

<sup>c</sup> The frequency for this event is the frequency for the facility design-basis earthquake-initiating cell fire.

Key: DBE = design-basis earthquake; km = kilometers; mi = miles; mrem = millirem; yr = year.

**Table 4–58 Annual Cancer Risks Due to Accidents at ANL-W for Alternative 6**

Accident	Maximally Exposed Offsite Individual <sup>a</sup>	Population within 80 km (50 mi) <sup>b</sup>	Noninvolved Worker <sup>a</sup>
<b>Driver Spent Nuclear Fuel</b>			
Waste Liquid Spill	$7.2 \times 10^{-6}$	0.00084	$2.6 \times 10^{-7}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.6 \times 10^{-9}$	$8.8 \times 10^{-11}$
Cask Drop	$1.5 \times 10^{-10}$	$1.8 \times 10^{-8}$	$3.4 \times 10^{-12}$
Earthquake (DBE)	0.000076	0.0088	$2.7 \times 10^{-6}$
Sodium Fire	$1.1 \times 10^{-6}$	0.00013	$8.0 \times 10^{-9}$
Criticality	$7.8 \times 10^{-10}$	$1.3 \times 10^{-7}$	$5.7 \times 10^{-10}$
<b>Blanket Spent Nuclear Fuel</b>			
Waste Liquid Spill	$1.8 \times 10^{-7}$	0.000021	$4.8 \times 10^{-9}$
Cask drop	$1.2 \times 10^{-11}$	$1.4 \times 10^{-9}$	$2.0 \times 10^{-13}$
Transuranic waste fire	$3.0 \times 10^{-11}$	$3.5 \times 10^{-9}$	$8.7 \times 10^{-11}$
Earthquake (DBE)	$1.9 \times 10^{-6}$	0.00022	$4.9 \times 10^{-8}$
Sodium Fire	$2.3 \times 10^{-8}$	$2.7 \times 10^{-6}$	$1.7 \times 10^{-10}$

<sup>a</sup> Increased likelihood of a latent cancer fatality.

<sup>b</sup> Increased number of latent cancer fatalities.

Key: DBE = design-basis earthquake; km = kilometers; mi = miles.

### Hazardous Chemical Impacts

Nonradiological impacts are evaluated in terms of comparison to ERPG. ERPG values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects (see Appendixes F, Section F.3.1.2 for details).

The hazardous chemical impacts of potential facility accidents associated with the treatment of the driver spent nuclear fuels using the electrometallurgical process are summarized in **Table 4–59**.

**Table 4–59 Hazardous Chemical Impacts Due to Accidents at ANL-W for Alternative 6**

<i>Accident</i>	<i>Frequency (event per year)</i>	<i>Receptor Location</i>	<i>Exposure</i>
Sodium fire	0.008	Noninvolved Worker at 100 meters	Sodium: less than ERPG-1
		Maximally exposed offsite individual	Sodium: less than ERPG-1

#### 4.8.5 Environmental Justice

As discussed in Section 4.8.4, operations conducted under this alternative would pose no significant health or other environmental risks to the public. The maximum likelihood of a latent cancer fatality for the maximally exposed offsite individual over the 10 years of melt and dilute treatment of sodium-bonded spent nuclear fuel at ANL-W would be  $3.7 \times 10^{-9}$  (or one chance in 270 million), and the expected number of latent cancer fatalities among the general population residing in the potentially affected area would be 0.000022 (or one chance in 45,500). Radiological and nonradiological risks posed by implementation of this alternative therefore would be small, regardless of the racial and ethnic composition of the population and independent of the economic status of individuals comprising the population. Operation of melt and dilute treatment processing facilities at ANL-W would have no disproportionately high and adverse effects on minority or low-income populations.

#### 4.8.6 Waste Management

This alternative would generate process wastes from treatment operations, other associated process wastes from normal support operations, and deactivation wastes following the conclusion of operations. Process wastes would include metal and ceramic high-level radioactive wastes from stabilizing the residual wastes from the existing Electrometallurgical Demonstration Project. Other associated process wastes would include operational wastes such as failed equipment, rags, packaging materials, and other miscellaneous items. Deactivation wastes would include the disposal of process equipment and other materials. All of these materials would be categorized according to existing DOE Orders and ANL-W waste management procedures. The anticipated categorization of these waste types and their expected interim storage and final disposal locations are given in Table 4–7 (see Section 4.2.6).

Estimates of the total amount of other associated process waste generated as a result of Alternative 6 are provided in **Table 4–60**. These values are based on an evaluation of waste forecasts from ANL-W, together with an understanding of melt and dilute process activities resulting in the generation of each waste category. The values in Table 4–60 are for disposal and include volume reduction. It is anticipated that a large fraction of the low-level radioactive waste generated as a result of Alternative 6 could be volume-reduced by up to 100 percent at the Waste Experimental Reduction Facility at INEEL prior to disposal at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex.

**Table 4–60 Amounts of Wastes Produced at ANL-W for Alternative 6<sup>a</sup>**

Waste Stream	Waste Quantities	
	Volume (m <sup>3</sup> )	Mass (kg)
<b>Direct Process Wastes</b>		
Fuel assembly hardware (low-level radioactive wastes)	0	0
High-level radioactive ceramic waste	19.4	29,000
High-level radioactive metal waste	0.6	460
Melt and dilute product	65.6	136,400
<b>Other Associated Process Wastes</b>		
High-level radioactive wastes	0.4	220
Low-level radioactive wastes <sup>b</sup>	711	144,000
Transuranic wastes	12.5	5,400
Mixed wastes	35.3	19,000
Sanitary wastes	4,960	1.72 × 10 <sup>6</sup>
<b>Deactivation Wastes</b>		
Low-level radioactive wastes <sup>b</sup>	213	72,000
Transuranic wastes	1.6	853
Mixed wastes	5.9	3,500

<sup>a</sup> These waste generation estimates are through the year 2015. This is the assumed date that these materials might be sent to the repository. Treatment, high-level radioactive waste processing, deactivation, and interim storage are accomplished during this time period.

<sup>b</sup> The volumes listed represent final disposal volumes following volume reduction at the Waste Experimental Reduction Facility at INEEL.

#### *Direct Process Wastes*

For this alternative, fuel hardware would be used as part of the stainless steel to form the metal ingot for disposal of the fuel by melting. Its mass is included as part of the spent nuclear fuel disposal.

Under this alternative, metal and ceramic high-level radioactive waste would be produced from existing process material at ANL-W. These wastes would be generated to stabilize materials produced during the demonstration. In addition, the salt removed from the melting furnace used for driver spent nuclear fuel would contain fission products that would be stabilized in ceramic waste. The volumes of waste forms provided in Table 4–60 are for the standardized canisters required for disposal of these materials.

A second metal high-level radioactive waste would be generated as a result of the melt and dilute treatment of fuel at ANL-W. It would be temporarily stored for 10 to 15 years at the Radioactive Scrap and Waste Facility at ANL-W to allow retrieval for future disposal. The Radioactive Scrap and Waste Facility was designed and constructed for temporary storage of this type of waste, and shielding would be provided by a combination of steel storage liners and the shielding provided by the soil surrounding the liners. When an offsite (proposed geologic) repository is available, the waste cans containing the two metals and ceramic high-level radioactive wastes would be removed from storage, shipped to the INEEL dry transfer facility, and prepared for shipment to the repository.

#### *Other Associated Process High-Level Radioactive Waste*

These high-level radioactive wastes could be generated as a result of driver and blanket spent nuclear fuel processing at ANL-W. This would result from activities in the Hot Fuel Examination Facility. Material in this

waste stream would consist of the absorbant used in the off-gas system which has collected the volatile radionuclides released from the spent nuclear fuel when heated.

The volume of high-level radioactive waste generated is expected to be less than the amount needed to fill a single high-level radioactive waste canister. Conservatively, the volume of a single canister, 0.4 cubic meters, has been used for the volume of high-level radioactive waste generated.

#### *Other Associated Process Low-Level Radioactive Wastes*

These low-level radioactive wastes would be generated as a result of the melt and dilute treatment of fuel at ANL-W. This would result from activities in the Fuel Conditioning Facility and the Hot Fuel Examination Facility at ANL-W (e.g., equipment decontamination and repair), as well as in other facilities at ANL-W (e.g., analytical laboratory activities). Material in this waste stream has been generated and routinely handled at ANL-W for many years.

The volume of low-level radioactive waste at ANL-W that will require disposal (after volume reduction) would be approximately 50 cubic meters (1,766 cubic feet) per year. This represents approximately 1 percent of the total annual volume of low-level radioactive waste currently being disposed of at the INEEL Subsurface Disposal Area at the Radioactive Waste Management Complex, and the total of 711 cubic meters (25,100 cubic feet) represents approximately 0.6 percent of the total Radioactive Waste Management Complex disposal capacity.

#### *Other Associated Process Transuranic Wastes*

These transuranic wastes would be generated by Alternative 6 from decontamination activities, repair and maintenance of items, and miscellaneous work associated with processing the sodium-bonded spent nuclear fuel. Transuranic wastes would be generated primarily from activities conducted in gloveboxes and hot cells at ANL-W.

All of the transuranic waste generated at ANL-W would be packaged and certified in accordance with Waste Isolation Pilot Plant acceptance criteria prior to transport to the Waste Isolation Pilot Plant. The transuranic waste generated would amount to approximately 1 cubic meter (35 cubic feet) per year, which is less than 0.002 percent of the volume of transuranic waste in retrievable storage at the Radioactive Waste Management Complex at INEEL. The total volume of transuranic waste is approximately 13 cubic meters (459 cubic feet), which is less than 0.008 percent of the estimated total volume of transuranic waste to be placed at the Waste Isolation Pilot Plant.

#### *Other Associated Process Mixed Wastes*

These mixed wastes would be generated primarily from the disposal of any cadmium-contaminated equipment or cleanup material and the analysis of cadmium samples. Mixed waste would be handled according to ANL-W procedures that require limited accumulation at the point of generation. Interim storage of this waste would be accomplished at the Radioactive Scrap and Waste Facility prior to eventual disposal. The Radioactive Scrap and Waste Facility is a permitted mixed waste storage facility for these materials. The mixed waste streams that contribute to the overall mixed waste generated by electrometallurgical treatment are identified in the INEEL Site Treatment Plan (DOE 1996b).

### *Deactivation Wastes*

A variety of wastes would be generated as part of deactivation activities associated with melt and dilute treatment of sodium-bonded spent nuclear fuel at ANL-W. Waste categories generated would include low-level radioactive waste, transuranic waste, and mixed waste. These wastes would be categorized and disposed of according to DOE Orders and ANL-W radioactive waste management procedures, as described above for each waste category.

The largest volume of deactivation wastes would be low-level radioactive wastes generated as a result of dismantling and disposal of electrometallurgical treatment and melt and dilute processing equipment. Components of the electrometallurgical demonstration project that would require disposition include two electrorefiners; two hot hydrostatic presses; and one V-mixer, as well as other components such as the grinder/crusher. Deactivation of components would generate additional mixed, transuranic, and low-level radioactive waste that would require management. The total deactivation wastes represent approximately 30 percent over the total incidental waste requiring disposal.

## **4.9 TRANSPORTATION IMPACTS**

Transportation impacts may be divided into two parts: the impacts of incident-free or routine transportation, and the impacts of transportation accidents. Incident-free transportation and transportation accident impacts are divided into two parts: nonradiological impacts and radiological impacts. Incident-free transportation impacts include radiological impacts on the public and the crew from the radiation field that surrounds the package. Nonradiological impacts of incident-free transportation are from vehicular emissions. Nonradiological impacts of potential transportation accidents include traffic accident fatalities. Only in the worst conceivable conditions, which are of low probability, could a transportation cask of the type used to transport radioactive material be so damaged that a release of radioactivity to the environment could occur.

The impact of a specific accident is expressed in terms of probabilistic risk, which is the probability of that accident occurring multiplied by its consequence. Hypothetical accidents ranging from a low-speed impact to those involving high-speed impacts with or without fires leading to cask failure are analyzed. The accident frequencies and consequences are binned using the method developed for the NRC, which is known as the "Modal Study" (NRC 1987). The overall risk is obtained by summing the individual risks from all accident bins. The risks for radiological accidents are expressed as additional latent cancer fatalities and as additional immediate fatalities for nonradiological accidents. The risks of incident-free effects are expressed in additional latent cancer fatalities.

The first step in the ground transportation analysis was to determine the incident-free and accident risk factors on a per-shipment basis for transportation of the various materials. Calculation of risk factors was accomplished by using the HIGHWAY (Johnson et al. 1993) computer codes to choose representative routes according to U.S. Department of Transportation regulations. These codes provide population estimates so that RADTRAN 5 (Neuhauser and Kanipe 1998) codes could be used to determine the radiological risk factors. This analysis is discussed in Appendix G. **Table 4-61** lists the fuels that could be shipped as part of the applicable alternatives used to treat sodium-bonded spent nuclear fuel.

**Table 4–61 Transportation Summary for Sodium-Bonded Fuels**

<i>Fuel Type</i>	<i>Applicable Alternatives<sup>a</sup></i>	<i>Metric Tons of Heavy Metal</i>	<i>Origin/State</i>	<i>Destination/State</i>	<i>Cask</i>	<i>Number of Shipments/Type of Transport</i>
EBR-II driver	All	1.1	ANL-W/ ID	ANL-W/ ID	HFEF-5	84/Onsite, intrafacility transfers
EBR-II driver	1,2,3,4,5,6	2.0	INTEC/ID	ANL-W/ ID	TN-FSV NAC-LWT	17/Onsite with roads open 43/Onsite with roads open
EBR-II blanket	All	22.4	ANL-W/ID	ANL-W/ ID	HFEF-5	165/Onsite, intrafacility transfers
Fast Flux Test Facility Driver	All	0.25	Hanford/WA	ANL-W/ID	T-3	10/ Public highways
Fermi-1 blanket	All	34.2	INTEC/ID	ANL-W/ID	PB-1	14/Onsite with road closed
Miscellaneous	All	0.04	Oak Ridge National Laboratory/TN Sandia National Laboratories/ NM	ANL-W/ID	To be determined by DOE	1/ Public highways  1/ Public highways
Declad EBR-II blanket	3,5	22.4	ANL-W	SRS/SC	NAC-LWT	11/Public highways
Declad Fermi-1 blanket	3,5	34.2	ANL-W	SRS/SC	NAC-LWT	18/Public highways

<sup>a</sup> “All” Includes the six alternatives plus the No Action Alternative.

Key: ID = Idaho; NM = New Mexico; SC = South Carolina; TN = Tennessee; WA = Washington.

The transportation of Fast Flux Test Facility driver spent nuclear fuel currently stored at the Hanford site and miscellaneous spent nuclear fuel currently stored at Oak Ridge National Laboratory and at Sandia National Laboratories are shipment campaigns related to sodium-bonded spent nuclear fuel and were analyzed by DOE in other NEPA documents, so they are not treated in detail in this impact analysis. See Appendix G for more details.

All EBR-II blanket and some EBR-II driver spent nuclear fuel are currently stored at ANL-W and would be subject to a building-to-building movement for processing. Since the movement is a short distance on closed DOE-controlled roads, DOE procedures and the NRC regulations do not require the use of a certified Type B cask. No incident-free risk analysis is necessary because the public would receive no measurable exposure. The worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. The probability and consequence of potential accidents during movement are bounded in frequency and consequence by handling accidents.

Fermi blanket spent nuclear fuel would be shipped from INTEC to ANL-W in the Type B cask (PB-1 Cask). Since DOE would close the roads between INTEC and ANL-W using existing traffic gates, and the road is uninhabited, no quantitative analysis is necessary. No incident-free risk analysis is necessary because the public would receive no measurable exposure. The worker dose is included in the process and handling dose estimates because the same personnel would be moving the spent nuclear fuel. Once the cask is closed for movement on the closed roads, the likelihood and consequence of any foreseeable accident are very small and are not further quantified.

EBR-II driver spent nuclear fuel would be shipped from INTEC to ANL-W in a certified Type B cask, either model TN-FSV or model NAC-LWT. Since the cask would be certified, DOE would not close the roads between INTEC and ANL-W. However, since the road is uninhabited, limited quantitative analysis is necessary. No incident-free risk analysis for exposure to the public at stops or in their homes is necessary. The worker dose is analyzed for the transportation crew, and the dose to other vehicles using the road is estimated. No accident analysis is necessary because potential accidents during movement are bounded in frequency and consequence by the handling accidents. Once the cask is closed for movement on the closed roads, the likelihood and consequence of any foreseeable accident are very small.

The following provides a summary of transportation impacts. Appendix G details the methods and assumptions used.

#### **4.9.1 No Action Alternative Impacts**

Under the No Action Alternative, it is assumed that the fuel would remain at current locations at INEEL/ANL-W, so there would be no impacts from transportation. The sodium-bonded Fast Flux Test Facility driver spent nuclear fuels and other miscellaneous fuels are assumed to be at, or brought to, Idaho, consistent with the Record of Decision for the Programmatic Spent Nuclear Fuel EIS, as amended (61 FR 9441).

#### **4.9.2 Onsite Transportation Impacts - Alternatives 1, 2, 4 and 6**

Approximately 17 shipments with the model TN-FSV cask or 43 shipments with the model NAC-LWT cask would be made by DOE under all alternatives. The analysis assumes that 43 shipments are made. The total distance traveled by trucks carrying radioactive materials on public roads located on the INEEL site would be 1,660 kilometers (1,000 miles). The dose rate from the cask is conservatively estimated to equal the maximum regulatory limit of 10 millirem per hour at two meters from the vehicle.

#### **Impacts of Onsite Incident-Free Transportation**

The dose to transportation workers from all transportation activities required by these alternatives has been estimated at  $4.7 \times 10^{-5}$  person-rem; the dose to the public would be  $3.5 \times 10^{-4}$  person-rem. Accordingly, incident-free transportation of radioactive material would result in  $1.9 \times 10^{-8}$  latent cancer fatalities among transportation workers and  $1.7 \times 10^{-7}$  latent cancer fatalities in the total affected population over the duration of the transportation activities. Latent cancer fatalities resulting from radiological exposures were estimated by multiplying the occupational (worker) dose by  $4 \times 10^{-4}$  latent cancer fatalities per person-rem of exposure, and the public accident and accident-free doses by  $5 \times 10^{-4}$  latent cancer fatalities per person-rem of exposure (ICRP 1991).

#### **Impacts of Onsite Accidents During Ground Transportation**

The maximum foreseeable onsite transportation accident under these alternatives (occurrence probability would be more than  $1 \times 10^{-7}$  per year) would not breach the transportation cask. The probability of more severe accidents was estimated to be lower than  $1 \times 10^{-7}$  per year.

Estimates of the total ground transportation accident risks under these alternative are as follows: a radiological dose to the population of less than  $1 \times 10^{-12}$  person-rem, resulting in less than  $1 \times 10^{-15}$  latent cancer fatalities; and traffic accidents resulting in  $8.2 \times 10^{-7}$  traffic fatalities.

### 4.9.3 On- and Offsite Transportation Impacts - Alternatives 3 and 5

In addition to the onsite transportation described above, Alternatives 3 and 5 require shipment of decontaminated and cleaned EBR-II and Fermi-1 blanket spent nuclear fuel from ANL-W to SRS. The impacts for these alternatives include both on- and offsite transportation. The total distance traveled on public roads by trucks carrying radioactive materials (packaged decontaminated and cleaned blanket spent nuclear fuels) would be 110,700 kilometers (69,000 miles). The dose rate from the cask traveling to SRS is calculated from the contents and the shielding capabilities of the cask.

#### Impacts of On- and Offsite Incident-Free Transportation

The dose to transportation workers from all transportation activities required by these alternatives has been estimated at  $2.0 \times 10^3$  person-rem; the dose to the public would be  $1.3 \times 10^2$  person-rem. Accordingly, incident-free transportation of radioactive material would result in  $7.9 \times 10^{-7}$  latent cancer fatalities among transportation workers and  $6.1 \times 10^{-6}$  latent cancer fatalities in the total affected population over the duration of the transportation activities. The estimated number of nonradiological fatalities from vehicular emissions associated with this alternative is  $2.0 \times 10^{-4}$ .

#### Impacts of On- and Offsite Accidents During Ground Transportation

The maximum foreseeable offsite transportation accident under these alternatives (occurrence probability would be more than  $1 \times 10^{-7}$  per year) is shipment of blanket spent nuclear fuels from DOE's facility at ANL-W to SRS, with a severity category V accident<sup>1</sup> in a suburban population zone under neutral (average) weather conditions. The accident could result in a dose of 0.46 person-rem to the public, with an associated  $2.3 \times 10^{-4}$  latent cancer fatalities, and a dose of  $1.9 \times 10^{-3}$  to the hypothetical maximally exposed individual with a latent cancer fatality risk of  $9.9 \times 10^{-7}$ . No fatalities would be expected to occur. The probability of more severe accidents, different weather conditions at the time of accident, or occurrence in a more densely populated area was estimated to have a probability of lower than  $1.0 \times 10^{-7}$  per year.

Estimates of the total ground transportation accident risks under this alternative are as follows: a radiological dose to the population of  $3.0 \times 10^{-6}$  person-rem would result in  $1.5 \times 10^{-9}$  latent cancer fatalities; and traffic accidents would result in 0.002 traffic fatalities.

### 4.10 CUMULATIVE IMPACTS

The Council on Environmental Quality regulations implementing NEPA procedural provisions define cumulative impacts as the impacts on the environment which result from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions, regardless of what agency (Federal or nonfederal) or person undertakes such other actions (40 CFR 1508.7). The cumulative impacts analysis presented in this section is based on the incremental actions associated with the maximum impacts for the treatment and management of sodium-bonded spent nuclear fuel at ANL-W and SRS; other actions associated with onsite activities; and offsite activities with the potential for related environmental impacts. Although it is unlikely that the alternative with the maximum impact would be implemented to treat and manage spent nuclear fuel at ANL-W and SRS, it was used to estimate cumulative impacts to ensure a conservative analysis. In accordance with a handbook recently prepared by the Council on Environmental Quality, DOE identified the resource areas in which the treatment and management of sodium-bonded spent nuclear fuel could add to the impacts of past, present, and reasonably foreseeable actions within the project impact zones, as defined by the Council on Environmental Quality (CEQ 1997).

<sup>1</sup> A category V accident is defined as a hypothetical accident in which the strain on the inner shell of the Type B truck cask, caused by an impact, is between two percent and 30 percent of the maximum, and the centerline of the lead shielding is heated to 343 °C, or less, by a fire.

Based on an examination of the environmental impacts of the proposed action, coupled with DOE and other agency actions, it was determined that cumulative impacts for the following areas need to be presented: (1) air resources, (2) water resources, (3) socioeconomics, (4) public and worker health, and (5) waste generation. Discussion of cumulative impacts for the following resources are omitted because impacts from the proposed treatment and management of sodium-bonded spent nuclear fuel would be so small that their potential contribution to cumulative impacts would be negligible: land resources, site infrastructure, geologic resources, ecological resources, and cultural and paleontological resources.

For determining the impact to air, water, socioeconomic, human health, and waste generation resources from commercial and Federal nuclear facilities, the 80-kilometer (50-mile) radius surrounding ANL-W and SRS was selected as the project impact zone. For liquid releases from SRS, the downstream population that uses the Savannah River as its source of drinking water was included in the SRS project impact zone.

**4.10.1 ANL-W and INEEL**

Significant offsite activities within a 80-kilometer (50-mile) radius of ANL-W and INEEL that potentially would contribute to the cumulative environmental impacts presented in this analysis include the System Integration Corporation quartzite mining operation in Arco Hills and the Food, Machinery, and Chemical Corporation, a phosphate processing operations in Pocatello, Idaho. The Food, Machinery, and Chemical Corporation is a primary source for offsite radiological emissions. These emissions have been evaluated by the EPA. Radiological impacts from the operation of the phosphate processing operations are minimal, and are not included in this assessment (DOE 1999a).

The counties surrounding ANL-W and INEEL have a number of existing and planned industrial and commercial facilities with permitted air emissions and water usage. Because of the distances between ANL-W and INEEL and the private industrial facilities, there is little opportunity for interactions of plant emissions and no major cumulative impact on air or water use. Reasonably foreseeable offsite actions evaluated in this EIS are presented in **Table 4-62**.

**Table 4-62 Offsite Activities Included in the Assessment of Cumulative Impacts at ANL-W and INEEL**

<i>Activity</i>	<i>Description</i>
Housing development, Idaho Falls	300-unit single family housing development planned on approximately 150 acres of vacant land.
Business park, Rexburg	50 acres of vacant land between two light industrial facilities planned for expansion into a light industrial/business park for 30-40 businesses.
Manufacturer, Pocatello	Existing manufactured home factory to expand from approximately 50 to between 140 and 150 employees. Expansion of 22 acres in Pocatello Airport Industrial Park.
Food, Machinery, and Chemical Corp., Pocatello	FMC phosphate manufacturing plant to reduce number of furnaces from 4 to 3 within the next two years; 25-30 jobs could be lost.
Target Department Store, Idaho Falls	Target discount store and associated commercial development near the Teton Mall in Idaho Falls.
System Integration Corporation Arco Hills Quartzite Mine	Quartzite mining operation and ore processing near Arco Hills on 56 acres. Fourteen acres would be disturbed by the quarry operation and a small waste ore dump; 22 acres would be disturbed by the construction of a haul road; 11 acres would be disturbed by the ore crushing facilities; and 9 acres would be disturbed by the loading facilities on the INEEL. The project would employ 40 workers.

Source: DOE 1999a.

DOE also evaluated the impacts from its own proposed future actions by examining impacts to resources and the human environment, as shown in NEPA documentation related to ANL-W and INEEL (see Section 1.6). The NEPA document related to ANL-W and INEEL that is considered in the cumulative impacts section is the *Surplus Plutonium Disposition Draft Environmental Impact Statement* (DOE 1998d). The Surplus Plutonium Disposition EIS analyzes the activities necessary to implement DOE's disposition strategy for surplus plutonium. INEEL is considered for pit disassembly and conversion and mixed oxide fuel fabrication for plutonium disposition. If chosen, these activities would take place at INTEC. Pit disassembly and conversion would be conducted in the Fuel Processing Facility and mixed oxide fuel fabrication in a new facility. Potential impacts from these activities are included in this section.

The cumulative impacts analysis also includes the impacts from actions proposed in this EIS. Risks to members of the public and site workers from radiological and nonradiological releases are based on operational impacts from the alternatives described in Chapter 4 of this EIS.

In addition, the cumulative impacts analysis accounts for other major ANL-W and INEEL operations. These major operations are presented in **Table 4-63**.

**Table 4-63 Ongoing Major Projects at ANL-W and INEEL**

<i>Project Name</i>	<i>Activity</i>
Advanced Mixed Waste Treatment Facility	Construct new facility
Auxiliary Reactor Area-II	Decontamination and decommissioning
Central Facilities Area Hot Laundry	Decontamination and decommissioning
Central Liquid Waste Processing Facility (Building 691)	Decontamination and decommissioning
Dry Fuel Storage Facility; Fuel Receiving Canning, Characterization and Shipping	Construct new facility
Electrometallurgical Process Demonstration (formerly known as Actinide Recycle Project)	Ongoing demonstration project in Fuel Conditioning Facility hot cell
Engineering Test Reactor (and Buildings 642 and 644)	Decontamination and decommissioning
Expended Core Facility Dry Cell Project	Construct new facility
Gravel Pit Expansions	Additional gravel pits as needed
Health Physics Instrument Laboratory	Construct new facility
Industrial/Commercial Landfill Expansion	Expand landfill for nonhazardous wastes
INEEL Site Operations Center	Construct new facility
Materials Test Reactor	Decontamination and decommissioning
Pit 9 Retrieval	Construct new facility
Radioactive Scrap and Waste Facility	Storage of spent nuclear fuels and radioactive scrap waste
Remote Treatment Facility	Construct new facility
Security Training Facility	Decontamination and decommissioning
Tank Farm Heel Removal Project	Construct new facility
Technology Development Center	Construct new facility
Test Area North (Buildings 620 and 656)	Decontamination and decommissioning
Test Reactor Area Filter Pits	Decontamination and decommissioning

Source: DOE 1999a, INEEL 1999.

Temporal limits were defined by examining the period of influence from both the proposed action and other Federal and nonfederal actions that have the potential for cumulative impacts. Actions to support the treatment of sodium-bonded blanket spent nuclear fuel at ANL-W are expected to begin in 2000 in preparation for ultimate offsite disposal, possibly in a geologic repository which probably will not be available until at least

2010. Final offsite shipments of spent nuclear fuel at ANL-W and INEEL for disposal would be completed by 2035.

The period of interest for the cumulative impacts analysis for this EIS includes the proposed construction and operation of facilities identified in the *Advanced Mixed Waste treatment Project Final Environmental Impact Statement* (DOE 1999a), while actions for other nuclear materials and surplus plutonium disposition would be ongoing.

#### **4.10.1.1 Air Resources**

It is expected that the alternatives for the treatment and management of sodium-bonded spent nuclear fuel at ANL-W will have a negligible impact on existing air quality, as any nonradiological emissions would be very low and well below regulatory concern (ANL 1999). Since there ostensibly would be no change from current air quality conditions at ANL-W (see Table 3–2), there would be no cumulative impacts. There also would be no contributory effect on Prevention of Significant Deterioration increment consumption at Craters of the Moon Wilderness (Class I) Area and Class II Areas.

DOE also evaluated the cumulative impacts of airborne radiological releases in terms of dose to a maximally exposed individual at the INEEL boundary and the population within 80 kilometers (50 miles) of ANL-W. The cumulative dose to the maximally exposed offsite individual would be well below the regulatory standard of 10 millirem per year (40 CFR Part 61). Summing the doses to the maximally exposed individual for the proposed action and baseline ANL-W and INEEL operations listed in Table 4–64 is an extremely conservative approach because, to get the calculated dose, the maximally exposed individual would have to occupy different physical locations at the same time, which is impossible.

The doses from current and projected activities at ANL-W and INEEL that are associated with the Advanced Mixed Waste Treatment Facility Program and this EIS would yield a cumulative dose from airborne sources (see Table 4–64).

#### **4.10.1.2 Water Resources**

There would be no liquid effluents released to surface water or groundwater from the operation of ANL-W or INEEL facilities as a result of the proposed action. Therefore, there would be no cumulative impact.

#### **4.10.1.3 Socioeconomic Impacts**

There would be no significant cumulative socioeconomic impacts from the operation of ANL-W or INEEL facilities as a result of the proposed action. Therefore, there would be no cumulative impact.

#### **4.10.1.4 Public and Worker Health**

**Table 4–64** summarizes the cumulative radiological health effects of routine ANL-W and INEEL operations, proposed DOE actions, and nonfederal nuclear facility operations. Impacts resulting from proposed DOE actions are described in the EISs listed in Section 1.6. In addition to estimated radiological doses to the hypothetical maximally exposed offsite individual, the offsite population, and involved workers, Table 4–64 lists the potential number of latent cancer fatalities for the public and workers from exposure to radiation. The radiation dose to the maximally exposed offsite individual would be 0.41 millirem per year, which is well below the applicable DOE regulatory limits (10 millirem per year from the air pathway and 100 millirem per year for all pathways). The total annual population dose of 2.49 person-rem for current and projected activities translates into 0.0012 latent cancer fatalities for each year of exposure for the population living within a 80-kilometer (50-mile) radius of the ANL-W.

The annual collective dose to the involved worker population would be 576.1 person-rem. In addition, doses to individual workers would be kept below the regulatory limit of 5,000 millirem per year (10 CFR 835). Furthermore, as low as reasonably achievable principles would be exercised to maintain individual worker doses below the DOE Administrative Control Level of 2,000 millirem per year.

**Table 4–64 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to Offsite Population and Facility Workers**

Activity	Maximally exposed offsite individual		Population <sup>a</sup>		Workers	
	Dose (millirem)	Latent Cancer Fatality Risk	Collective Dose (person-rem)	Latent Cancer Fatalities	Collective Dose (person-rem)	Latent Cancer Fatalities
ANL-W and INEEL Baseline <sup>b</sup>	0.021	$1.1 \times 10^{-8}$	0.23	0.00012	115	0.046
SBSNF EIS <sup>c</sup>	0.002	$1.10 \times 10^{-9}$	0.012	$6.0 \times 10^{-6}$	22	0.0088
Surplus Plutonium Disposition <sup>d</sup>	0.016	$8 \times 10^{-9}$	2.2	0.0011	345	0.14
Advanced Mixed Waste Treatment Program <sup>e</sup>	0.36	$1.8 \times 10^{-7}$	0.048	0.000024	4.1 <sup>f</sup>	0.0016
Total	0.40	$2.0 \times 10^{-7}$	2.5	0.0013	486.1	0.20

<sup>a</sup> A collective dose to the 80-kilometer (50-mile) population for atmospheric releases. There would be no liquid releases from ANL-W and INEEL facilities as a result of the proposed action.

<sup>b</sup> From Tables 3–7 and 3–8 of this SBSNF EIS.

<sup>c</sup> Alternative 6. Melt and dilute blanket and driver fuel at ANL-W.

<sup>d</sup> DOE 1998d: Tables 4–134 and 4–135.

<sup>e</sup> DOE 1999a: Tables 5.12–1 and E.4–7.

<sup>f</sup> Average number of workers is 50 (DOE 1999a: Table E.4–7)  $\times$  80 millirem = collective dose.

#### 4.10.1.5 Environmental Justice

As discussed in Chapter 4 and Appendix H, implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel at ANL-W and INEEL would have no significant cumulative impacts on public health or the environment. The implementation of any of the alternatives at ANL-W or the No Action Alternative at INEEL would result in no disproportionately high and adverse impacts on minority or low-income populations residing within potentially affected areas.

#### 4.10.1.6 Waste Generation

As stated in the Waste Management discussions for each alternative presented earlier in Chapter 4, high-level radioactive waste, transuranic waste, low-level radioactive waste, and mixed and hazardous waste would be generated from the treatment of sodium-bonded spent nuclear fuel. The largest volume of high-level radioactive waste would be generated at ANL-W under the No Action-Direct Disposal Alternative (152 cubic meters [5,370 cubic feet]). However, as stated in the Waste Management discussions, the projected high-level radioactive waste generation rate would not require additional treatment and storage capacities beyond the current and planned INEEL capacities.

**Table 4–65** lists cumulative the volumes of high-level and low-level radioactive waste, transuranic, and hazardous and mixed wastes that ANL-W and INEEL would generate. The estimated quantity of radioactive/hazardous waste from baseline operations in this forecast during the next 15 years would be 119,550 cubic meters (7.05 million cubic feet). Waste generated by Alternative 6: Melt and Dilute Blanket and Driver Fuel at ANL-W, would add a total of 1,065 cubic meters (37,610 cubic feet). During a 10-year

time period, other reasonably foreseeable activities associated with the disposition of surplus plutonium could add an additional 1,640 cubic meters (57,920 cubic feet). Therefore, the potential cumulative total amount of waste generated from ANL-W and INEEL activities would be 122,255 cubic meters (4.32 million cubic feet). It is important to note that the quantities of waste generated are not equivalent to the amounts that will require disposal. As discussed in Chapter 4 for each of the treatment alternatives, during treatment high-level radioactive material is evaporated and concentrated to a smaller volume for final disposal.

**Table 4–65 Estimated Cumulative Total Waste Generation From ANL-W and INEEL Concurrent Activities (cubic meters)**

<i>Waste Type</i>	<i>ANL-W and INEEL Baseline Operations<sup>a</sup></i>	<i>SBSNF EIS<sup>b</sup></i>	<i>Surplus Plutonium Disposition<sup>c</sup></i>	<i>Total</i>
High-level radioactive	8,250	86	0	8,336
Low-level radioactive	64,500	924	940	66,364
Hazardous/mixed	46,800	41	60	46,901
Transuranic	0	14	640	654
Total	119,550	1,065	1,640	122,255

<sup>a</sup> From SBSNF EIS Table 3–9, 15 years of operation.

<sup>b</sup> Alternative 6. Melt and Dilute Blanket and Driver Fuel at ANL-W, 15 years of operation.

<sup>c</sup> DOE 1998d: Table 4–133, 10 years of operation.

The Central Facilities Area and Bonneville County landfill accepts nonhazardous and nonradioactive solid wastes generated at INEEL. The onsite landfill complex was designed to accommodate combined ANL-W and INEEL solid waste disposal needs for a projected maximum operational life of 30 years. The Cold Waste Handling Facility at INTEC is designed to inspect, recycle, shred, compact, and segregate nonhazardous waste, thereby reducing the amount of material sent to disposal.

The activities supporting the treatment and management of sodium-bonded spent nuclear fuel and other planned ANL-W and INEEL activities would not generate larger volumes of radioactive, hazardous, or solid wastes beyond the current and projected capacities of ANL-W and INEEL waste storage and/or management facilities.

#### 4.10.2 Savannah River Site

Nuclear facilities within a 80-kilometer (50-mile) radius of SRS include Georgia Power’s Vogtle Electric Generating Plant across the river from SRS; Chem-Nuclear Inc., a commercial low-level radioactive waste burial site just east of SRS; and Starmet CMI, Inc. (formerly Carolina Metals), located southeast of SRS, which processes uranium-contaminated metals. Radiological impacts from the operation of the Vogtle Electric Generating Plant, a two-unit commercial nuclear power plant, are minimal, but DOE has factored them into the analysis. As stated in the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g), the South Carolina Department of Health and Environmental Control Annual Report indicates that operation of the Chem-Nuclear Services facility and the Starmet CMI facility do not noticeably impact radiation levels in air or liquid pathways in the vicinity of SRS. Therefore, they are not included in this assessment.

The counties surrounding SRS have numerous existing (e.g., textile mills, paper product mills, and manufacturing facilities) and planned (e.g., Bridgestone Tire and Hankook Polyester) industrial facilities with permitted air emissions and discharges to surface waters. Because of the distances between SRS and the private industrial facilities, there is little opportunity for interactions of plant emissions, and no major cumulative impact on air or water quality. Construction and operation of Bridgestone Tire and Hankook Polyester facilities could affect the regional socioeconomic cumulative impacts.

Additional offsite facilities with the potential to affect the nonradiological environment include South Carolina Electric and Gas Company's Urquhart Station. Urquhart Station is a three-unit, 250-megawatt, coal- and natural gas-fired steam electric plant on Beech Island, South Carolina, located about 32 river kilometers (20 river miles) north of SRS. Because of the distance between SRS and the Urquhart Station and the regional wind direction frequencies, there is little opportunity for any interaction of plant emissions, and no significant cumulative impact on air quality.

DOE also evaluated the impacts from its own proposed future actions by examining impacts to resources and the human environment, as shown in NEPA documentation related to SRS (see Section 1.6). Additional NEPA documents related to SRS that are considered in the cumulative impacts section include the following:

***Environmental Assessment for the Tritium Facility Modernization and Consolidation Project at the Savannah River Site*** (DOE 1998a). This environmental assessment addresses the impacts of consolidating the tritium activities currently performed in Building 232-H into the new Building 233-H and Building 234-H. Tritium extraction functions would be transferred to the Tritium Extraction Facility. The overall impact would be to reduce the tritium facility complex net tritium emissions by up to 50 percent. Another positive effect of this planned action would be to reduce the amount of low-level radioactive job-control waste. Effects on other resources would be negligible. Therefore, impacts from the environmental assessment have not been included in this cumulative impacts analysis.

***Final Environmental Impact Statement on Management of Certain Plutonium Residues and Scrub Alloy Stored at the Rocky Flats Environmental Technology Site*** (DOE 1998e). DOE proposes to process certain plutonium-bearing materials being stored at the Rocky Flats Environmental Technology Site. These materials are plutonium residues and scrub alloy remaining from nuclear weapons manufacturing operations formerly conducted by DOE at Rocky Flats. DOE has decided to remove the plutonium from certain residues that would be shipped from the Rocky Flats Environmental Technology Site to SRS for stabilization. The separated plutonium would be stored at SRS pending disposition decisions. Environmental impacts from using the F-Canyon to chemically separate the plutonium from the remaining materials at SRS are included in this section.

***Draft and Final Environmental Impact Statements for the Construction and Operation of a Tritium Extraction Facility at the Savannah River Site*** (DOE 1998b, DOE 1999c). DOE proposes to construct and operate a Tritium Extraction Facility at SRS to provide the capability to extract tritium from commercial light water reactor targets and targets of similar design. The purpose of the proposed action and alternatives evaluated in the EIS is to provide tritium extraction capability to support reactor tritium production. Environmental impacts from the maximum processing option in this EIS are included in this section.

***Surplus Plutonium Disposition Draft Environmental Impact Statement*** (DOE 1998d). This EIS analyzes the activities necessary to implement DOE's disposition strategy for surplus plutonium. SRS is considered the preferred location for mixed oxide fuel fabrication and plutonium immobilization facilities that would be used for plutonium disposition. SRS also is the preferred site for the pit disassembly and conversion facility. Impacts from this EIS are included in this section.

***Defense Waste Processing Facility Supplemental Environmental Impact Statement*** (DOE 1994b). The selected alternative in the Record of Decision was the completion and operation of the Defense Waste Processing Facility to immobilize high-level radioactive waste at SRS. The facility is currently processing sludge from SRS high-level radioactive waste tanks. However, SRS baseline data are not representative of full Defense Waste Processing Facility operational impacts, including the processing of salt and supernate from these tanks. Therefore, the Defense Waste Processing Facility data is listed separately.

The cumulative impacts analysis also includes the impacts from actions proposed in this SBSNF EIS. Risks to members of the public and site workers from radiological and nonradiological releases are based on operational impacts from the alternatives described in Chapter 4, Sections 4.5 and 4.7, of this EIS.

Temporal limits were defined by examining the period of influence from both the proposed action and other Federal and nonfederal actions that have the potential for cumulative impacts. Actions to support the treatment of sodium-bonded blanket spent nuclear fuel at SRS are expected to begin in 2003 in preparation for ultimate offsite disposal, possibly in a geologic repository which probably will not be available until at least 2010. Final offsite shipments for spent nuclear fuel currently assigned to SRS for disposal would be completed by 2035.

The period of interest for the cumulative impacts analysis for this EIS includes the proposed construction and operation of facilities identified in the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g) and the Tritium Extraction Facility, while actions for nuclear materials, highly enriched uranium, and surplus plutonium disposition would be ongoing.

#### **4.10.2.1 Air Resources**

**Table 4-66** compares the cumulative concentrations of nonradiological air pollutants from SRS to Federal and state regulatory standards. The listed values are the maximum modeled concentrations that could occur at ground level at the site boundary. The data demonstrate that total estimated concentrations of nonradiological air pollutants from SRS in all cases would be below the regulatory standards at the site boundary. The highest percentages of the regulatory standards are for sulfur dioxide concentrations for the shorter time interval (approximately 59 percent of the standard for the 24-hour averaging time); for particulate matter of less than 10 microns, 24-hour interim (approximately 54 percent of the standard); and for sulfur dioxide, 3-hour averaging time (approximately 54 percent of the standard). The remaining pollutant emissions would be below 25 percent of the applicable standards.

DOE also evaluated the cumulative impacts of airborne radiological releases in terms of dose to a maximally exposed individual at the SRS boundary. DOE included the impacts of the Vogtle Plant (NRC 1996) in this cumulative total. The radiological emissions from the operation of the Chem-Nuclear low-level radioactive waste disposal facility just east of SRS are very low (DOE 1998g) and are not included. **Table 4-67** lists the results of this analysis, using 1996 emissions (1992 for the Vogtle Plant) for the SRS baseline. The cumulative dose to the maximally exposed offsite individual would be 0.87 millirem per year, well below the regulatory standard of 10 millirem per year (40 CFR Part 61). Summing the doses to the maximally exposed individual for the proposed action and baseline SRS operations listed in **Table 4-67** is an extremely conservative approach because, to get the calculated dose, the maximally exposed individual would have to occupy different physical locations at the same time, which is impossible.

Adding the population doses from current and projected activities at SRS, the Vogtle Plant, the SRS Spent Nuclear Fuel Management Draft EIS, and this EIS could yield a total annual cumulative dose of 39.77 person-rem from airborne sources. The total annual cumulative dose translates into 0.020 latent cancer fatalities for each year of exposure for the population living within an 80-kilometer (50-mile) radius of SRS.

**Table 4-66 Estimated Maximum Cumulative Ground-Level Concentrations of Nonradiological Pollutants (micrograms per cubic meter) at SRS Boundary**

<i>Pollutant</i>	<i>Averaging Time</i>	<i>Most stringent standard or guideline<sup>a</sup></i>	<i>SRS baseline<sup>b</sup></i>	<i>SBSNF<sup>c</sup></i>	<i>Other foreseeable planned SRS activities<sup>d</sup></i>	<i>Cumulative Concentration</i>	<i>Percent of Standard</i>
Carbon monoxide	8 hours	10,000 <sup>e</sup>	632	1.22	20.61	653.8	7
	1 hour	40,000 <sup>e</sup>	5,010	9.06	89.36	5,108.4	13
Nitrogen dioxide	Annual	100 <sup>e</sup>	8.8	3.11	7.02	18.9	19
PM <sub>10</sub>	Annual	50 <sup>e</sup>	4.8	less than 0.01	0.05	4.9	10
	24 hours (interim)	150 <sup>e</sup>	80.6	0.11	0.29	81	54
	24 hours (99 <sup>th</sup> percentile over 3 years)	150 <sup>f</sup>	(g)	NA	NA	NA	NA
PM <sub>25</sub>	3 year annual	15 <sup>f</sup>	(g)	NA	NA	NA	NA
	24 hours (98 <sup>th</sup> percentile over 3 years)	65 <sup>f</sup>	(g)	NA	NA	NA	NA
Sulfur dioxide	Annual	80 <sup>e</sup>	16.3	less than 0.01	0.14	16.5	21
	24 hours	365 <sup>e</sup>	215	0.12	1.63	216.8	59
	3 hours	1,300 <sup>e</sup>	690	0.91	5.38	696.3	54

N/A = Not available.

<sup>a</sup> The more stringent Federal and state standards are presented if both exist for the averaging period.

<sup>b</sup> Data from Table 3-13 of this EIS.

<sup>c</sup> Alternative 3, PUREX Process Blanket Fuel at SRS F-Canyon.

<sup>d</sup> Data compiled from SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g: Table 5-1).

<sup>e</sup> Federal and state standard.

<sup>f</sup> Federal standard.

<sup>g</sup> No data available with which to assess particulate matter concentrations.

Key: PM<sub>10</sub> = Particulate matter less than or equal to *n* microns.

#### 4.10.2.2 Water Resources

At present, a number of SRS facilities discharge treated wastewater to Upper Three Runs and its tributaries and Fourmile Branch via NPDES-permitted outfalls. These include the F- and H-Area Effluent Treatment Facility and the M-Area Liquid Effluent Treatment Facility. As stated in Sections 4.5.2 and 4.7.2, operations associated with the treatment and management of sodium-bonded spent nuclear fuel are not expected to result in any discharges to groundwater. The only technology that would result in discharges of radiological and nonradiological effluents to surface water would be PUREX processing. The major sources of liquid effluents from facilities associated with PUREX processing would be process cooling water and steam condensate systems that could contain small quantities of radionuclides and chemicals. This process wastewater would be treated at the F-Area Effluent Treatment Facility and then discharged to the Upper Three Runs Creek. Studies of water quality and biota downstream of the Effluent Treatment Facility outfall suggest that discharges have not degraded the water quality of Upper Three Runs (DOE 1998g).

Other potential sources of contaminants into Upper Three Runs during the periods in which sodium-bonded spent nuclear fuel would be treated in F-Area using PUREX, or in L-Area using melt and dilute treatment, include activities described in the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g), the tritium extraction facility, environmental restoration, and decontamination and deactivation activities, as well as modifications to existing SRS facilities. Discharges from activities associated with the SRS Spent Nuclear Fuel Management Draft EIS and tritium extraction facility would not add significant amounts of nonradiological contaminants to Upper Three Runs. The amount of discharge associated with environmental

restoration and decontamination and deactivation activities would vary based on the level of activity. All the potential activities that could result in wastewater discharges would be required to comply with the NPDES permit limits that ensure protection of water quality.

**Table 4–67 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to the Maximally Exposed Offsite Individual and Population in the 80-Kilometer (50-Mile) Radius From Airborne Releases at SRS**

Activity	Offsite Population			
	Maximally Exposed Offsite Individual		Population Within 80 Kilometers (50 Miles)	
	Dose (millirem)	Latent Cancer Fatality Risk	Collective Dose (person-rem)	Latent Cancer Fatalities
SRS Baseline <sup>a</sup>	0.050	$2.5 \times 10^{-8}$	5.5	$2.8 \times 10^{-3}$
This SBSNF EIS <sup>b</sup>	0.00039	$2.0 \times 10^{-10}$	0.019	$9.5 \times 10^{-6}$
Management of Spent Nuclear Fuel <sup>c</sup>	0.015	$7.5 \times 10^{-9}$	0.56	0.00028
Surplus Highly Enriched Uranium Disposition <sup>c</sup>	0.0025	$1.3 \times 10^{-9}$	0.16	0.00008
Interim Management of Nuclear Materials <sup>c</sup>	0.77	$3.9 \times 10^{-7}$	31	0.016
Tritium Extraction Facility <sup>c</sup>	0.02	$1.0 \times 10^{-8}$	0.77	0.00039
Surplus Plutonium Disposition <sup>c</sup>	0.004	$2.0 \times 10^{-9}$	1.6	0.0008
Management of Plutonium Residues/ Scrub Alloy <sup>c</sup>	0.003	$1.5 \times 10^{-9}$	0.038	0.000019
Defense Waste Processing Facility <sup>c</sup>	0.001	$5.0 \times 10^{-10}$	0.071	0.000036
DOE complex miscellaneous components <sup>c</sup>	0.0044	$2.2 \times 10^{-9}$	0.007	$3.3 \times 10^{-6}$
Vogtle Plant <sup>c</sup>	0.00054	$2.7 \times 10^{-10}$	0.042	0.000021
Total	0.87	$4.4 \times 10^{-7}$	39.8	0.020

<sup>a</sup> Data from Table 3–16 of this SBSNF EIS.

<sup>b</sup> Alternative 3. PUREX Process Blanket Fuel at SRS F-Canyon.

<sup>c</sup> Data from SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g: Table 5–2 maximum-impact alternative).

Key: HEU = Highly enriched uranium.

**Table 4–68** summarizes the estimated cumulative radiological doses from waterborne sources to human receptors downstream from SRS. Liquid effluents released to SRS streams that are tributaries of the Savannah River could contain small quantities of radionuclides. The exposure pathways considered in this analysis included drinking water, fish ingestion, shoreline exposure, swimming, and boating. The estimated cumulative dose to the maximally exposed offsite individual from liquid releases would be 0.26 millirem per year, well below the regulatory standard of 4 millirem per year (40 CFR Part 141). Adding the population doses associated with current and projected SRS activities would yield a cumulative annual dose of 3.24 person-rem from liquid sources. This translates into 0.0016 latent cancer fatalities for each year of exposure of the population living within an 80-kilometer (50-mile) radius of SRS.

**Table 4–68 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to Offsite Population in the 80-Kilometer (50-Mile) Radius From Liquid Releases at SRS**

Activity	Offsite Population			
	Maximally Exposed Offsite Individual		Population Within 80 Kilometer (50-miles)	
	Dose (millirem)	Latent Cancer Fatality Risk	Collective Dose (person-rem)	Latent Cancer Fatalities
SRS Baseline <sup>a</sup>	0.13	$6.5 \times 10^{-8}$	2.4	0.0012
SBSNF EIS <sup>b</sup>	0.00012	$6.0 \times 10^{-11}$	0.00068	$3.4 \times 10^{-7}$
Management of Spent Nuclear Fuel <sup>c</sup>	0.057	$2.9 \times 10^{-8}$	0.19	0.000095
Surplus Highly Enriched Uranium Disposition <sup>c</sup>	(d)	(d)	(d)	(d)
Interim Mgmt of Nuclear Materials <sup>c</sup>	0.022	$1.1 \times 10^{-8}$	0.65	0.00033
Tritium Extraction Facility <sup>c</sup>	(d)	(d)	(d)	(d)
Defense Waste Processing Facility <sup>c</sup>	(d)	(d)	(d)	(d)
Surplus Plutonium Disposition <sup>c</sup>	(d)	(d)	(d)	(d)
Management Plutonium Residues/ Scrub Alloy <sup>c</sup>	(d)	(d)	(d)	(d)
DOE complex miscellaneous components <sup>c</sup>	0.000042	$2.1 \times 10^{-11}$	0.00024	$1.2 \times 10^{-7}$
Sodium-Bonded Spent Nuclear Fuel <sup>c</sup>	0.000042	$2.1 \times 10^{-11}$	0.00024	$1.2 \times 10^{-7}$
Plant Vogtle <sup>c</sup>	0.054	$2.7 \times 10^{-8}$	0.0025	$1.3 \times 10^{-6}$
Total	0.26	$1.3 \times 10^{-7}$	3.24	0.0016

<sup>a</sup> Data from Table 3–16 of this SBSNF EIS.

<sup>b</sup> Alternative 3. PUREX Process Blanket Fuel at SRS F-Canyon.

<sup>c</sup> Data from SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g: Table 5-3 maximum-impact alternative).

<sup>d</sup> Less than minimum reportable levels.

Key: HEU = Highly enriched uranium.

#### 4.10.2.3 Socioeconomic Impacts

Cumulative regional economic and population changes from construction and operation of the Transfer and Storage Facility or the Transfer, Storage, and Treatment Facility, both of which are described in the SRS Spent Nuclear Fuel Management Draft EIS, consider the impacts of other coincident economic development projects such as Bridgestone-Firestone and Hankook Synthetics.

Bridgestone-Firestone is building a \$435 million tire manufacturing plant in Aiken County, South Carolina, that will employ 800 workers. Hankook Synthetics announced plans to build an \$850 million polyester fiber plant in Richmond County, Georgia, that will employ 500 workers. Both the Bridgestone-Firestone and Hankook projects are expected to complete construction and be in operation by 2000. Thus, these two projects should not impact the construction workforce for the Transfer and Storage Facility or the Transfer, Storage, and Treatment Facility, which are not scheduled to be constructed until 2000. Competition for construction workers should not overlap (DOE 1998g).

Construction of the Transfer and Storage Facility or the transfer and storage phase of the Transfer, Storage, and Treatment Facility, both of which are described in the SRS Spent Nuclear Fuel Management Draft EIS, would start in 2000, employ 500 workers (375 construction and 125 professional), and require two years to complete. The treatment phase would begin construction at the completion of the transfer and storage phases and also could employ as many as 500 workers and take as long as two years to complete. No additional workers would be required during operations since existing SRS employees would assume those positions. There would be no significant cumulative socioeconomic impacts from construction or operation of the Transfer and Storage Facility or the Transfer, Storage, and Treatment Facility (DOE 1998g).

#### **4.10.2.4 Public and Worker Health**

**Table 4–69** summarizes the cumulative radiological health effects of routine SRS operations, proposed DOE actions, and nonfederal nuclear facility operations (Vogtle Electric Generating Plant). Impacts resulting from proposed DOE actions are described in the EISs listed previously in this chapter. In addition to estimated radiological doses to the hypothetical maximally exposed offsite individual, the offsite population, and the involved workers, Table 4–69 lists the potential number of latent cancer fatalities for the public and workers due to radiation exposure. The radiation dose to the maximally exposed offsite individual from air and liquid pathways would be 1.13 millirem per year, which is well below the applicable DOE regulatory limits (10 millirem per year from the air pathway; four millirem per year from the liquid pathway; and 100 millirem per year for all pathways). The total annual population dose for current and projected activities of 43.07 person-rem translates into 0.02 latent cancer fatalities for each year of exposure for the population living within an 80-kilometer (50-mile) radius of SRS.

The annual radiation dose to the involved worker population would be 1,152 person-rem. In addition, doses to individual workers would be kept below the regulatory limit of 5,000 millirem per year (10 CFR 835). Furthermore, standards and practices to ensure worker doses are as low as reasonably achievable would be exercised to maintain individual worker doses below the DOE Administrative Control Level of 2,000 millirem per year.

#### **4.10.2.5 Environmental Justice**

As discussed in Chapter 4 and Appendix H, implementation of the alternatives for the treatment and management of sodium-bonded spent nuclear fuel at SRS would have no significant cumulative impacts on public health or the environment. The implementation of either of two alternatives at SRS would result in no disproportionately high and adverse impacts on minority or low-income populations residing within potentially affected areas.

#### **4.10.2.6 Waste Generation**

As stated in Sections 4.5.6 and 4.7.6, low-level and high-level radioactive waste, transuranic waste, mixed waste, and hazardous waste would be generated from the treatment of sodium-bonded spent nuclear fuel. The largest volume of high-level radioactive and transuranic waste would be generated with PUREX processing. However, as stated in Sections 4.5.6 and 4.7.6, the projected high-level radioactive and transuranic waste generation rates would not require additional treatment and storage capacities beyond the current and planned SRS capacities.

**Table 4-69 Estimated Average Annual Cumulative Radiological Doses and Resulting Health Effects to Offsite Population and Facility Workers**

Activity	Maximally Exposed Offsite Individual				Population <sup>a</sup>				Workers	
	Dose from Airborne Releases (millirem)	Dose from Liquid Releases (millirem)	Total Dose (millirem)	Latent Cancer Fatalities Risk	Collective Dose from Airborne Releases (person-rem)	Collective Dose from Liquid Releases (person-rem)	Total Collective Dose	Latent Cancer Fatalities	Collective Dose (person-rem)	Excess Latent cancer fatalities
SRS Baseline <sup>b</sup>	0.050	0.13	0.18	$9.5 \times 10^{-8}$	5.5	2.4	7.9	0.0025	165	0.066
SBSNF EIS <sup>c</sup>	0.00039	0.00012	0.00051	$2.6 \times 10^{-10}$	0.019	0.00068	0.020	$1.0 \times 10^{-8}$	38	0.015
Management of Spent Nuclear Fuel <sup>d</sup>	0.015	0.057	0.072	$3.6 \times 10^{-8}$	0.56	0.19	0.75	0.00038	55	0.022
Surplus Highly Enriched Uranium Disposition <sup>d</sup>	0.0025	(e)	0.0025	$1.3 \times 10^{-8}$	0.16	(e)	0.16	0.00008	11	0.00044
Interim Management of Nuclear Materials <sup>d</sup>	0.77	0.022	0.79	$4.0 \times 10^{-7}$	31	0.65	31.7	0.016	130	0.052
Tritium Extraction Facility <sup>d</sup>	0.02	(e)	0.02	$1.0 \times 10^{-8}$	0.77	(e)	0.77	0.00039	4	0.0016
Defense Waste Processing Facility <sup>d</sup>	0.001	(e)	0.001	$5.0 \times 10^{-10}$	0.071	(e)	0.071	0.000036	120	0.048
Surplus Plutonium Disposition <sup>d</sup>	0.004	(e)	0.004	$2.0 \times 10^{-9}$	1.6	(e)	1.6	0.0008	541	0.22
Management Plutonium Residues/ Scrub Alloy <sup>d</sup>	0.003	(e)	0.003	$1.5 \times 10^{-9}$	0.038	(e)	0.038	0.000019	47	0.019
DOE Complex Miscellaneous Components <sup>d</sup>	0.0044	0.000042	0.0044	$2.2 \times 10^{-9}$	0.007	0.00024	0.0072	$3.6 \times 10^{-6}$	2	0.001

<i>Activity</i>	<i>Maximally Exposed Offsite Individual</i>				<i>Population<sup>a</sup></i>				<i>Workers</i>	
	<i>Dose from Airborne Releases (millirem)</i>	<i>Dose from Liquid Releases (millirem)</i>	<i>Total Dose (millirem)</i>	<i>Latent Cancer Fatalities Risk</i>	<i>Collective Dose from Airborne Releases (person-rem)</i>	<i>Collective Dose from Liquid Releases (person-rem)</i>	<i>Total Collective Dose</i>	<i>Latent Cancer Fatalities</i>	<i>Collective Dose (person-rem)</i>	<i>Excess Latent cancer fatalities</i>
Sodium-Bonded Spent Nuclear Fuel <sup>d</sup>	0.00012	0.000042	0.00016	$8.1 \times 10^{-11}$	0.0042	0.00024	0.004	$2.2 \times 10^{-6}$	2	0.001
Vogtle Plant <sup>d</sup>	0.00054	0.054	0.055	$2.7 \times 10^{-8}$	0.042	0.0025	0.045	0.000022	NA	NA
Total	0.87	0.26	1.13	$5.9 \times 10^{-7}$	39.77	3.24	43.07	0.022	1,115	0.45

<sup>a</sup> A collective dose to the 80-kilometer (50-mile) population for atmospheric releases and to the downstream users of the Savannah River for liquid releases.

<sup>b</sup> Data from Tables 3–16 and 3–17 of this EIS.

<sup>c</sup> Alternative 3: PUREX Process Blanket Fuel at SRS F-Canyon.

<sup>d</sup> Data from SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g: Table 5-4, maximum-impact alternative).

<sup>e</sup> Less than minimum reportable levels.

**Table 4-70** lists the cumulative volumes of high- and low-level radioactive, transuranic, hazardous, and mixed wastes that SRS would generate. The table includes data from the SRS 30-year expected waste forecast (DOE 1998g). The 30-year expected waste forecast is based on operations, environmental restoration, and decontamination and deactivation waste forecasts from existing generators and the following assumptions: (1) secondary waste from the Defense Waste Processing Facility, In-Tank Precipitation, and Extended Sludge Processing operations are addressed in the Defense Waste Processing Facility EIS; (2) high-level radioactive waste volumes are based on the selected option for the F-Canyon Plutonium Solutions EIS and the Interim Management of Nuclear Materials at SRS EIS; (3) some investigation-derived wastes are handled as hazardous waste per Resource Conservation and Recovery Act (RCRA) regulations; (4) purge water from well samplings is handled as hazardous waste; and (5) the continued receipt of small amounts of low-level radioactive waste from other DOE facilities and nuclear naval operations (DOE 1998g).

**Table 4-70 Estimated Cumulative Total Waste Generation from SRS Concurrent Activities (Cubic Meters)**

<i>Waste Type</i>	<i>SRS Baseline Operations<sup>a</sup></i>	<i>SBSNF EIS<sup>b</sup></i>	<i>Management<sup>a</sup> of SNF</i>	<i>ER/D&amp;D<sup>a</sup></i>	<i>Other Waste Volume<sup>a</sup></i>	<i>Total</i>
High-level radioactive	150,750	106	11,000	0	69,642	231,498
Low-level radioactive	343,710	1,398	140,000	132,000	194,553	811,661
Hazardous/mixed	90,450	3	270	575,180	5,156	671,059
Transuranic	18,090	17	3,700	4,820	8,760	35,387
Total	603,000	1,524	154,970	712,000	278,111	1,749,605

<sup>a</sup> Data from SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998g) maximum-impact alternative, Table 5-5, based on a total of 30-year expected waste forecast, which includes previously generated waste.

<sup>b</sup> Alternative 5. Melt and Dilute Blanket Fuel at SRS Building 105-L, 15-years of operation.

Key: ER/D&D = environmental restoration/decontamination & deactivation; based on a total 30-year expected waste forecast.

The estimated quantity of radioactive/hazardous waste from SRS operations in this forecast during the next 30 years would be 603,000 cubic meters (21.3 million cubic feet). Waste generated by Alternative 5: Melt and Dilute Blanket Fuel at SRS Building 105-L, would add a total of 1,524 cubic meters (53,820 cubic feet). Waste generated from the conventional (PUREX) processing option described in the SRS Spent Nuclear Fuel Management Draft EIS would add a total of 154,970 cubic meters (5.48 million cubic feet). In addition, radioactive/hazardous waste associated with environmental restoration and decontamination and deactivation activities would have a 30-year expected forecast of 712,000 cubic meters (25.1 million cubic feet) (DOE 1998g). During this same time period, other reasonably foreseeable activities that were not included in the 30-year forecast would add an additional 278,111 cubic meters (9.82 million cubic feet). Therefore, the potential cumulative amount of waste generated from SRS activities during the period of interest would be 1,749,605 cubic meters (61.8 million cubic feet). It is important to note that the quantities of waste generated are not equivalent to the amounts that will require disposal. As discussed in Chapter 4 for each of the treatment alternatives at SRS, during treatment high-level radioactive material is evaporated and concentrated to a smaller volume for final disposal. Combustible low-level radioactive waste is volume-reduced on site in the Consolidated Incineration Facility.

The Three Rivers Solid Waste Authority Regional Waste Management Center at SRS accepts nonhazardous and nonradioactive solid wastes from SRS and eight surrounding South Carolina counties. This municipal solid waste landfill provides state-of-the-art Subtitle D (nonhazardous) facilities for land-filling solid wastes while reducing the environmental consequences associated with construction and operation of multiple county-level facilities (DOE 1998g). It was designed to accommodate combined SRS and county solid waste disposal needs for at least 20 years, with a projected maximum operational life of 45 to 60 years (DOE 1998g). The landfill is designed to handle an average of 1,000 tons per day and a maximum of 2,000 tons per day of municipal solid wastes. The SRS and eight cooperating counties had a combined generation rate of 900 tons

per day in 1995. The Three Rivers Solid Waste Authority Regional Waste Management Center opened in mid-1998.

Activities supporting the treatment and management of sodium-bonded spent nuclear fuel and other planned SRS activities would not generate larger volumes of radioactive, hazardous, or solid wastes beyond the current and projected capacities of SRS waste storage and/or management facilities.

#### **4.11 PROGRAMMATIC CONSIDERATIONS**

The programmatic considerations presented below is a programmatic perspective of the alternatives vis-a-vis the current regulatory environment regarding spent nuclear fuel and high-level radioactive waste and the expected time frame for the disposal of DOE-owned spent nuclear fuel or high-level radioactive waste in a geologic repository.

##### **4.11.1 Regulatory Environment Considerations**

Prior to the acceptance of spent nuclear fuel or high-level radioactive waste at the proposed repository, certain regulatory and DOE Office of Civilian Radioactive Waste Management requirements must be met. Regulatory requirements specific to DOE's sodium-bonded spent nuclear fuel, are identified in the Civilian Radioactive Waste Management Office's March 19, 1999, draft *Waste Acceptance System Requirements Document* (DOE/RW) (DOE 1999b).

One of the key non-DOE (NRC) requirements for acceptance of spent nuclear fuel or high-level radioactive waste is that it cannot contain or generate materials that are explosive, pyrophoric, or chemically reactive (in the repository environment) in a form or amount that could compromise the repository's ability to perform its waste isolation function or to satisfy its performance objective (10 CFR 135(b)(1)). The No Action Alternative may not satisfy this requirement, because the metallic sodium is highly reactive, the metallic uranium is also reactive and potentially pyrophoric, and in some cases the fuel contains highly enriched uranium, which would require criticality control measures. It also is uncertain whether the treatment technology, identified for the blanket spent nuclear fuel under Alternative 2 (cleaning the fuel to remove sodium and packaging in a high-integrity can), would be adequate to meet the repository acceptance criteria. Under all other alternatives, this requirement could be met.

The *Waste Acceptance System Requirements Document* identifies the Civilian Radioactive Waste Management Office's acceptance criteria for spent nuclear fuel and high-level radioactive waste. Under these criteria, the Civilian Radioactive Waste Management Office states that only spent nuclear fuel and high-level radioactive waste that is not subject to regulation under RCRA, Subtitle C, will be accepted for disposal. Untreated sodium-bonded spent nuclear fuel may be regulated under RCRA, Subtitle C, because it exhibits the characteristic of reactivity when exposed to water (40 CFR 261.23 (a)(2), (3)) and is ignitable (40 CFR 261.21 (a)(2)).

Under RCRA 40 CFR 268.9 (c), "... no prohibited waste which exhibits a characteristic under 40 CFR part 261, subpart C, may be land disposed of unless the waste complies with the treatment standards under 40 CFR part 268, subpart D." Deactivation is the waste treatment technology that exhibits the characteristic of reactivity and ignitability (40 CFR 268.40). RCRA land disposal requirements (i.e., 40 CFR 268.40) also require generators of wastes that exhibit the characteristics of reactivity to water or ignitability to identify all underlying hazardous constituents reasonably expected to be present in the waste at the point of generation, and to treat these constituents to the universal treatment standards. If the characteristic waste is treated by the applicable treatment and the waste no longer exhibits the characteristic, then the waste no longer needs to be regulated under RCRA, Subtitle C, and can be managed as a nonhazardous waste (62 FR 62083).

The direct disposal option of the No Action Alternative may not satisfy this requirement, because the sodium-bonded spent nuclear fuel could still be considered to be reactive and ignitable, and therefore, it may not be accepted for disposal at the potential geologic repository. All of the alternatives under the proposed action would be able to deactivate the sodium-bonded fuel and remove the characteristics of reactivity and ignitability. The metallic uranium is described as reactive, and in some cases pyrophoric; however, it would not be a RCRA hazardous characteristic because it is defined under the Atomic Energy Act of 1954, as amended (42 U.S.C. 2001 *et seq.*), as a source, special nuclear, or byproduct material and, therefore, is excluded from RCRA under 40 CFR 261.4 (a)(4).

The *Waste Acceptance System Requirements Document* also identifies the Civilian Radioactive Waste Management Office's specific acceptance criteria for DOE's spent nuclear fuel and high-level radioactive waste. For high-level radioactive waste, the document specifies a standard vitrified high-level radioactive waste form as borosilicate glass. Specific acceptance criteria standards have not been developed for other treated waste forms of high-level radioactive waste (e.g., ceramic forms and metal waste forms). For DOE's spent nuclear fuel, specific acceptance criteria have been developed for canistered DOE spent nuclear fuel, including naval spent nuclear fuel that is intended for disposal in the canister. However, specific acceptance criteria have not been developed for spent nuclear fuel that has been melted into a liquid form and then solidified (e.g., conditioned). The No Action Alternative may be able to meet this requirement for the disposal canisters; however, it may not meet all of the other requirements previously discussed (e.g., reactivity, ignitability, and RCRA regulations); therefore, it may not be acceptable for disposal.

For Alternative 3 (blanket fuel) where the treated waste form would be a vitrified borosilicate glass, the specific acceptance criteria have been developed. However, final approval of this waste form would be contingent upon the requirements in the disposal facilities license.

For Alternative 1 (blanket and driver fuel), Alternative 2 (driver fuel), Alternative 3 (driver fuel), Alternative 4 (driver fuel), and Alternative 5 (driver fuel), performance criteria for the ceramic high-level radioactive waste and the metal high-level radioactive waste form have been developed, but need approval. Again, final approval of this waste form would be contingent upon the requirements in the disposal facilities license.

For Alternative 2 (blanket fuel), the specific acceptance criteria for canistered spent nuclear fuel would apply and most likely could be achieved. However, the long-term durability of the proposed overpack container has not been demonstrated or documented. Without such demonstration of extended containment, the ability of the high-integrity can concept to meet the safety standards proposed by the National Research Council is unknown (National Research Council 1998).

For Alternative 4 (blanket fuel), Alternative 5 (blanket fuel), and Alternative 6 (blanket and driver fuel), the specific acceptance criteria for conditioned spent nuclear fuel would need to be developed and approved.

#### **4.11.2 Schedule Considerations**

The schedule perspective for each of the alternatives is affected by two time frames: the proposed schedule for the construction, operation, and closure of a geologic repository, and 2035, the year by which DOE committed to remove all spent nuclear fuel from Idaho under the 1995 agreement with the State of Idaho.

The proposed schedule for the repository is discussed in *Viability Assessment of a Repository at Yucca Mountain*, Volume 1, December 1998, DOE/RW (DOE 1998h). A site recommendation decision for the geologic repository is scheduled for 2001. If the site were to be subsequently authorized, a license application could be submitted in 2002. The NRC construction authorization decision could occur in 2005 at the earliest. Repository construction would begin upon receipt of this authorization. DOE must update its licensing application and submit it to the NRC before the Commission will issue a license to receive and process nuclear

waste. This update is scheduled for 2008. Assuming repository construction sufficient to begin waste emplacement will take five years, the first waste emplacement at Yucca Mountain could occur in 2010. DOE would design the repository to close as early as approximately 10 years after emplacement of the last waste package, or to be kept open for at least 100 years after initiation of waste emplacement, with a reasonable expectation that the repository actually could be kept open with appropriate maintenance for 300 years after initiation of waste emplacement. The Viability Assessment (DOE 1998h) assumes a reference case in which closure of a monitored geologic repository is initiated in 2110, 100 years after initiation of waste emplacement operations.

Under the No Action Alternative, the untreated sodium-bonded spent nuclear fuel could remain in storage at the current locations until 2035. After that, it would need to be transported outside the State of Idaho and stored or treated at another DOE site. If the waste acceptance criteria are finalized by 2010 and indicates that direct disposal of the sodium-bonded spent nuclear fuel is possible, the fuel could be packaged for direct disposal well before 2035.

The treatment of the driver spent nuclear fuel using the electrometallurgical technology under Alternatives 1 through 5 could start as early as 2000 and could be completed by 2006 to 2007. If the decision to select a technology is delayed until after 2010, when waste acceptance criteria may be finalized, it would require two to three years lead time for the reactivation or installation of new equipment for the electrometallurgical treatment technology and six to seven years for the processing, for a total of approximately 10 years. The high-level radioactive waste would be ready for disposal by 2020.

The treatment of driver spent nuclear fuel only using the melt and dilute process at ANL-W could start as early as 2005 and could be completed by 2007. If installation of the necessary equipment is delayed until after 2010, the conditioned spent nuclear fuel would be ready for disposal in 2017.

The treatment of the blanket spent nuclear fuel using the electrometallurgical technology under Alternative 1 could start as early as 2000 and could be completed by 2012 or 2013. A delayed decision for after 2010 would add 10 to 15 years, depending on the time required to reactivate or install new equipment. The process still could be completed by 2030.

The preparation of the blanket spent nuclear fuel and its placement in high-integrity cans under Alternative 2 could start in 2003. Cleaning and sodium removal activities and packaging would take approximately six years for completion by 2009. Delaying a decision until after 2010 would delay the completion to approximately 2020.

The treatment of blanket spent nuclear fuel using the PUREX process at SRS would not start until 2005 because the F-Canyon is committed to other missions. Once started, however, all blanket spent nuclear fuel could be processed in less than one year. The decladding and sodium removal activities at ANL-W to prepare the blanket spent nuclear fuel for transportation and processing also would not start until 2003, allowing for installation of new equipment. It is estimated that preparation activities at ANL-W for all blanket spent nuclear fuel would last approximately six years. Therefore, the overall process could be completed by approximately 2010. At this time it is not clear whether the decision to process blanket spent nuclear fuel at the F-Canyon could be delayed until after 2010 because DOE has scheduled operations for the F-Canyon until 2005; if there were a programmatic decision to close the F-Canyon after currently scheduled operations are completed, the F-Canyon will not be available.

The preparation of blanket spent nuclear fuel for the melt and dilute process at ANL-W under Alternative 4 could start in 2003, allowing time for the installation of new equipment. The melt and dilute activities could start in 2005 and be completed in seven years, by 2012. The process would require sodium removal activities at ANL-W, which could be done in parallel. The blanket spent nuclear fuel preparation activities would start

in 2003 and would require approximately six years for completion. The overall process could be completed by 2012. A delayed decision until after 2010 would push completion to approximately 2020.

The treatment of blanket spent nuclear fuel using the melt and dilute process at SRS under Alternative 5 could start after 2020 if capacity becomes available. It is estimated that the facility would be operational by 2005, but it is committed to other missions until 2035, as stated in the SRS Spent Nuclear Fuel Management EIS (DOE 1998g). Processing of the blanket spent nuclear fuel at SRS would take approximately three years. The decladding and sodium removal activities at ANL-W that are needed to prepare the fuel could start in 2003 and be completed by 2009, well before processing begins. Delaying a decision until 2010 would push the completion of the decladding activities to 2019, which could be well before processing could begin at SRS.

The treatment of blanket and driver spent nuclear fuel using the melt and dilute process at ANL-W under Alternative 6 could start as early as 2005 and be completed by 2015. Delaying a decision until 2010 would push completion to approximately 2025.

**Table 4–71** provides a summary of the dates for completing the process for each alternative, given that a decision to proceed is made in the year 2000 or the year 2010.

**Table 4–71 Year of Completing Treatment<sup>a</sup>**

	<i>Decision in 2000</i>	<i>Decision in 2010</i>
No Action (Direct Disposal)	Before 2035	Before 2035
<u>Alternative 1</u>		
Driver (only)	2006	2020
Driver and blanket	2012	2030
<u>Alternative 2</u>		
Driver	2006	2020
Blanket	2009	2020
<u>Alternative 3</u>		
Driver	2006	2020
Blanket	2010	F-Canyon may not be available
<u>Alternative 4</u>		
Driver	2006	2020
Blanket	2012	2020
<u>Alternative 5</u>		
Driver	2006	2020
Blanket	2025	2025
<u>Alternative 6</u>		
Driver (only)	2007	2017
Driver and blanket	2015	2025

<sup>a</sup> See Section 2.5 for an explanation of alternatives.

#### 4.12 MITIGATION MEASURES

Following completion of an EIS and its associated Record of Decision, DOE is required to prepare a Mitigation Action Plan to address any mitigation commitments expressed in the Record of Decision (10 CFR 1021.331). The purpose of the Mitigation Action Plan is to explain how measures designed to mitigate adverse environmental impacts will be planned and implemented. The Mitigation Action Plan is prepared prior to DOE taking any action directed by the Record of Decision that is the subject of a mitigation commitment.

Based on analyses of the environmental consequences of the proposed action presented earlier in this chapter, no mitigation measures would be necessary since all potential environmental impacts would be small and well

within applicable requirements. Each DOE site would follow installation and operational practices that would minimize any potential impacts to air and surface water quality, noise, operational and public health and safety, and accident prevention and mitigation. These practices are dictated by Federal and state licensing and permitting requirements, as described in Chapter 5.

#### **4.13 RESOURCE COMMITMENTS**

This section describes the unavoidable adverse environmental impacts that could result from the proposed action; the relationship between short-term uses of the environment and the maintenance and enhancement of long-term productivity; and irreversible and irretrievable commitments of resources. Unavoidable adverse environmental impacts are impacts that would occur after implementation of all feasible mitigation measures. The relationship between short-term uses of the environment and the maintenance and enhancement of long-term productivity addresses issues associated with the condition and maintenance of existing environmental resources used to support the proposed action and the utility of these resources after their use. Resources that would be irreversibly and irretrievably committed are those that cannot be recovered or recycled and those that are consumed or reduced to unrecoverable forms.

##### **4.13.1 Unavoidable Adverse Environmental Impacts**

Implementing any of the alternatives considered in this EIS for the treatment and management of sodium-bonded spent nuclear fuel would result in unavoidable adverse impacts to the human environment. In general, these impacts are expected to be minimal and would come from incremental impacts attributed to the operation of treatment and management facilities at ANL-W and SRS.

Operation of treatment and management facilities at ANL-W and SRS would result in unavoidable increases of radiation exposures to workers and the general public. Workers would be exposed to direct radiation and other chemicals associated with the handling and treatment of the sodium-bonded spent nuclear fuel. The incremental annual dose contribution from the treatment and management of sodium-bonded spent nuclear fuel to the maximally exposed individual, general population, and workers are discussed in Sections 4.3.4, 4.4.4, 4.5.4, 4.6.4, 4.7.4, and 4.8.4.

Also unavoidable would be the generation of additional low-level radioactive waste, which would either be treated and stored on site at ANL-W or SRS, or transported and managed off site at low-level radioactive waste disposal facilities. Any other waste generated during treatment and management activities would be collected at the site, treated and/or stored, and eventually removed for suitable recycling or disposal off site in accordance with applicable EPA regulations.

Operation of treatment and management facilities at ANL-W and SRS would have minimal unavoidable adverse environmental impacts to air and water quality. Air quality could be affected by increases in various chemical or radiological constituents in the routine emissions typical of facility operations at these sites. Impacts to water resources and quality also could be affected by increases in various chemical or radiological constituents in the routine effluent releases. Impacts to the environment associated with the normal operation of facilities at ANL-W and SRS would occur regardless of the treatment and management of spent nuclear fuel. These routine impacts also have been addressed in various other NEPA documentation at these sites.

The alternative treatment processes would generate varying amounts of waste material that could affect storage requirements. This would be an unavoidable impact on the amount of available and anticipated storage space and the requirements of disposal facilities.

#### **4.13.2 Relationship Between Local Short-Term Uses of the Environment and the Maintenance and Enhancement of Long-Term Productivity**

Implementation of the alternatives, including the No Action Alternative, would cause short-term commitments of resources (e.g., air emissions and water discharges) and would permanently commit certain resources (e.g., dilution materials and energy). For each alternative, the short-term use of these resources would result in potential long-term benefits to the environment and the enhancement of long-term productivity by decreasing overall health risks to workers, the public, and the surrounding environment by reducing their exposure to hazardous and radioactive substances. The short-term effect on workers, the public, and the environment from the treatment of sodium-bonded spent nuclear fuel would be offset by the long-term benefits of safe, stable, secure storage of these materials.

Under the No Action Alternative, environmental resources already have been committed to the storage of spent nuclear fuel. This commitment would serve to maintain existing environmental conditions with little or no impacts to the long-term productivity of the environment. The continued storage of sodium-bonded spent nuclear fuel at ANL-W and INEEL until 2035 and the potential for its direct disposal in a geologic repository would result in less exposure to hazardous and radioactive materials for workers, the public, and the environment than would be experienced under the proposed action. Only the direct disposal of the sodium-bonded fuel in a repository would have the potential to enhance the long-term viability of the environment in Idaho.

Under the Proposed Action, the short-term use of environmental resources at ANL-W and SRS would be greater than for the No Action Alternative. The short-term commitment of resources would include the space required for onsite processing, the commitment of processing facilities, transportation, and other disposal resources and materials for the treatment and management of sodium-bonded spent nuclear fuel. Workers, the public, and the environment would be exposed to larger amounts of hazardous and radioactive materials over the short-term from the handling and treatment of the spent nuclear fuel, including process emissions and the handling of wastes. Again, these commitments would be offset by an even greater potential for enhanced long-term viability of the environment than under the No Action Alternative.

Over the life of the proposed action, groundwater would be used at SRS to meet sanitary and process needs. After use and treatment, this water would be discharged into surface water streams. Depending on the site chosen ( F- or L-Area) and the technology implemented over the short-term, the resulting increases in pollutant loadings would take advantage of the natural assimilative capacity of the receiving stream(s). However, these incremental pollutant loadings should not adversely affect either short- or long-term viability of the aquatic ecosystem. These impacts would be assessed during the regulatory permitting process once an alternative has been selected.

Regardless of location, air emissions associated with implementation of any of the technologies would add small amounts of radiological and nonradiological constituents to the air of the regions around ANL-W and SRS. During the project's life, these emissions would result in additional loading and exposure, but would not impact compliance with air quality or radiation exposure standards at either site. There would be no significant residual environmental effects to long-term environmental viability.

The management and disposal of sanitary solid waste and nonrecyclable radiological waste over the project's life would require energy and space at ANL-W and SRS treatment, storage, or disposal facilities. The land required to meet the solid waste needs would require a long-term commitment of terrestrial resources. Upon the facilities' closures, DOE could decontaminate and decommission the facilities and/or equipment and restore them to brown field sites which could be available for future commercial or industrial development.

Regardless of location, continued employment, expenditures, and tax revenues generated during the implementation of any of the alternatives would directly benefit the local, regional, and state economies over the short-term. Long-term economic productivity could be facilitated by local governments investing project-generated tax revenues into infrastructure and other required services.

The use of short-term resources to operate spent nuclear fuel treatment and management facilities at either ANL-W or SRS would not affect the long-term productivity of these sites.

#### **4.13.3 Irreversible and Irretrievable Commitments of Resources**

Irreversible and irretrievable commitments of resources for each alternative potentially would include mineral resources during the life of the project and energy used in treating the waste. The commitment of capital, energy, labor, and material during the implementation of the alternatives generally would be irreversible.

Energy expended would be in the form of fuel for equipment and vehicles, electricity for facility operations, and human labor. Operation of any proposed facility would generate nonrecyclable waste streams, such as radiological and nonradiological solid wastes and some process wastewaters. However, certain materials and equipment used during operation of the proposed facility could be recycled when the facility is decontaminated and decommissioned.

The implementation of the alternatives considered in this EIS, including the No Action Alternative, would require water, electricity, steam, and diesel fuel. Water at SRS and ANL-W would be obtained from onsite groundwater sources and steam from existing onsite sources. Electricity and diesel fuel would be purchased from commercial sources. These commodities are readily available and the amounts required would not have an appreciable impact on available supplies or capacities. From a materials and energy resource commitment perspective, electrometallurgical treatment and PUREX process technologies would recover low-enriched uranium, which is usable as commercial reactor fuel.

The disposal of hazardous and/or radioactive wastes also would cause irreversible and irretrievable commitments of land, mineral, and energy resources. Hazardous waste and low-level radioactive waste disposal would irreversibly and irretrievably commit land for its disposal. For each of the alternatives analyzed in this document, the No Action Alternative would have the least commitment of land, mineral, and energy resources.

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