APPENDIX F
EVALUATION OF HUMAN HEALTH EFFECTS FROM FACILITY ACCIDENTS

F.1 INTRODUCTION

This appendix presents the methodology and assumptions used for estimating potential impacts and risks associated with both radiological and toxic chemical releases, due to postulated accidents, at the facilities being considered for the processing of sodium-bonded spent nuclear fuel. Analysis of radiological impacts is presented in section F.2. This is followed by a summary of the risk results for the various alternatives analyzed in detail. Chemical risk methodologies and results are presented in section F.3. Information regarding the impacts of normal operations, along with background information on the health impacts from exposure to ionizing radiation is provided in Appendix E.

F.2 IMPACT OF RADIOLOGICAL ACCIDENTS ON HUMAN HEALTH

This section of Appendix F addresses the radiological impacts associated with accidents at management facilities. Potential accident scenarios have been identified for both the Argonne National Laboratories-West (ANL-W) and Savannah River Site (SRS) facilities proposed for the treatment and management of sodium-bonded spent nuclear fuel.

F.2.1 Overview of Methodology and Basic Assumptions

For the radiological evaluation, the GENII computer program (PNL 1988) has been used to calculate radiation doses to the general population and to selected individuals. Appendix E provides the detailed description of this code, and therefore only the GENII data specific to the accident analysis is presented in this appendix.

The impact of radiation exposure was evaluated for the following segments of the population for each accident scenario:

- **Noninvolved Worker—**An individual (a noninvolved worker) located 100 meters (330 feet) from the radioactive material release point.[1] The dose to the noninvolved worker is calculated for the 50th percentile meteorology only, as specified in DOE-STD-1027-92 (DOE 1992). Noninvolved workers are exposed unprotected to the plume for a limited time (a maximum of 5 minutes), receiving exposure via inhalation, air immersion, and ground surface pathways only.

- **Maximally Exposed Offsite Individual—**A hypothetical individual living at the management site boundary and receiving the maximum exposure. The hypothetical member of the public is located directly downwind of the accident and is exposed to radioactivity via inhalation, ingestion, air immersion, and ground surface pathways. The individual would be exposed to the plume for the entire release duration.

- **Population—**The general public living within an 80-kilometer (50-mile) radius of the facility, residing directly downwind of the accident, and receiving the maximum exposure via inhalation, ingestion, air immersion, and ground surface pathways.

[1] *For elevated release, the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 meters (330 feet) for an elevated release.*

The doses to the maximally exposed offsite individual and the general public are calculated for the 50th and 95th percentile meteorological conditions. Meteorology specific to ANL-W and SRS were used in the evaluation. The site-specific meteorological data was obtained in the form of a joint frequency distribution in terms of percentage of time that the wind blows in specific directions for the given midpoint (or average) wind speed and atmospheric stability. Accident consequences were calculated for both 50th and 95th percentile meteorological conditions. The 50th percentile condition represents the median meteorological condition, and is defined as that for which more severe conditions occur 50 percent of the time. The 95th percentile condition represents relatively low probability meteorological conditions that produce higher calculated exposures, and is defined as that condition that is not exceeded more than 5 percent of the time. GENII determines 50th and 95th percentile meteorological conditions using site-specific joint frequency distribution weather data.

The following conditions were used in the calculations:

❒ **Meteorological Data**

- Site-specific joint frequency distribution weather data are used to define 50th and 95th percentile meteorological conditions for each processing technology at management sites.

- If a release occurs through a stack, the release is assumed to occur at an elevated level consistent with the site's effluent emission stack height. The effects of plume rise were not credited in the analysis.

- Mixing layer height is 1,000 meters (3,280 feet). Airborne materials freely diffuse in the atmosphere near the ground level in what is known as the mixing depth. A stable layer exists above the mixing depth and restricts vertical diffusion above 1,000 meters.

- Wet deposition is zero (it is assumed that no rains occur to accelerate deposition and reduce the size of the area affected by the release).

- Dry deposition of the cloud is modeled. During movement of the radioactive plume, a fraction of the radioactive material in the plume is deposited on the ground due to gravitational forces. The quantity of deposited radioactive material is proportional to the particle size and deposition velocities (in meters per second). The deposited material contributes to the exposure from ground surface radiation and ingestion.

❒ **Inhalation Data**

- Breathing rate is 330 cubic centimeters per second (0.7 cubic feet per minute) for the worker and the general public at the site boundary and beyond (maximally exposed individual and population) during the passage of the plume; it is 270 cm^3/sec (0.57 $feet^3$/min) for the general public during the other times.

- Exposure during passage of the entire plume is assessed for the maximally exposed offsite individual and the population. Exposures to the noninvolved worker are to a portion of the plume (i.e., the noninvolved worker is exposed to the plume for a limited time) because the worker is assumed to take emergency action.

- Inhalation exposure factors are based on the International Commission on Radiological Protection, Publication 30 (ICRP 1982).

Exposure time assumptions for maximally exposed individuals, workers, and the general public are provided in **Table F–1** below. Since all accident releases are to the air (either gaseous or suspended particulates), drinking water, aquatic food ingestion, and any other pathways that may involve liquid exposure are not examined. Additional information, common to the analysis of the impacts of normal operations and accidents has previously been presented in Appendix E.

Table F–1 GENII Exposure Parameters to Plumes and Soil Contamination (Postulated Accidents)

Maximally Exposed Offsite Individual			*General Population*		
Inhalation and External Exposure			*Inhalation and External Exposure*		
Exposure Time (hours)	*Breathing Rate (cm^3/sec)*	*Soil Contamination (hours)*	*Exposure Time (hours)*	*Breathing Rate (cm^3/sec)*	*Soil Contamination (hours)*
100 percent of Release Time	330	6,136	100 percent of Release Time	330	6,136

cm^3/sec = cubic centimeter per second
Source: PNL 1988.

Radiological impacts to noninvolved workers from postulated accident scenarios were evaluated at onsite locations where a given incident would cause the highest dose. The noninvolved worker was assumed to have an inhalation exposure time of 5 minutes and an external exposure time to soil contamination of 20 minutes. For a ground-level release accident, a noninvolved worker was assumed to be 100 meters from a given release point; for an elevated release, the worker was situated between 200 and 500 meters, depending on the given site's atmospheric dispersion characteristics. All doses to noninvolved workers include a component associated with the intake of radioactivity into the body, and another component resulting from external exposure to direct radiation.

The radiation dose to individuals and the public resulting from exposure to radioactive releases was calculated using the following potential pathways:

- *Air Immersion*—External direct exposure from immersion in the airborne radioactive material
- *Ground Surface*—External direct exposure from radioactive material deposited on the ground
- *Inhalation*—Internal exposure from inhalation of radioactive aerosols and suspended particles
- *Ingestion*—Internal exposure from ingestion of contaminated terrestrial food and animal products.

The radiation dose is estimated by the GENII computer program in a manner recommended by the International Commission on Radiological Protection in Publications 26 and 30 (ICRP 1977, ICRP 1982). Committed dose equivalents[2] are calculated individually for organs such as the gonads, breast, red bone marrow, lungs, thyroid, and bone surface; calculations are combined for the liver, upper large intestine, lower large intestine, small intestine, and stomach. Weighting factors are used for various body organs to calculate weighted or committed effective dose equivalents from radiation inside the body due to inhalation or ingestion. The committed effective dose equivalent value is the summation of the committed dose equivalent to a specific organ weighted, by the relative risk to that organ compared to an equivalent whole-body exposure. Deep-dose equivalent for the external exposure pathways (immersion in the radioactive material and exposure to the ground contamination) and 50-year committed effective dose equivalent for the internal exposure pathways are calculated. The sum of the deep-dose equivalent for external pathways and committed effective dose equivalent for internal pathways is called the total dose in this EIS.

The exposure from ingestion of contaminated terrestrial food and animal products is calculated on a yearly basis. It is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose should exceed that of the protective action guidelines in a radiological accident event (EPA 1991). No reduction of exposure because of protective actions or evacuation of the public was accounted for in this analysis, however. This conservative approach may result in overestimating health effects within an exposed population, but allows for consistent comparisons between alternatives.

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[2]*The definitions of committed dose equivalents, committed effective dose equivalents, and total effective dose equivalents are consistent with those given in 10 Code of Federal Regulations (CFR) Part 835, "Occupational Radiation Protection; Final Rule."*

F.2.2 Selection of Facility Accidents for Detailed Evaluations

The alternatives for the treatment of sodium-bonded spent nuclear fuel assume the use of facilities currently in operation, although modifications to SRS Building 105-L will be necessary before it can be used for the melt and dilute alternative. The selection of accident scenarios is based on those evaluated in the facility safety analysis reports.

Postulated facility accident scenarios were developed based on the review of the analyzed accidents in previous safety analysis, risk assessment, and environmental assessment documents at ANL-W and SRS where the sodium-bonded fuel may be handled or processed. After reviewing a wide range of documents, postulated accident scenarios were developed based on information contained in the following:

- *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement* (DOE 1995a)

- *Electrometallurgical Treatment Research and Demonstration Project in the Fuel Conditioning Facility at ANL-West* (DOE 1996a)

- *Fuel Cycle Facility Final Safety Analysis Report, Revision 4* (ANL 1998a)

- *Safety Analysis Report for the Hot Fuel Examination Facility, Rev 00 DRAFT* (ANL 1998b)

- *Accident Assessments for Idaho National Engineering Laboratory Facilities, DOE/ID* (Slaughterbeek et. al. 1995)

- *Safety Analysis-200 Area, Savannah River Site F-Canyon Operation, F-Canyon SAR Addendum* (WSRC 1994)

- *Savannah River Site Spent Nuclear Fuel Management Draft Environmental Impact Statement,* (DOE 1998)

Based on this review of analyzed accident scenarios at ANL-W and SRS facilities that deal with sodium-bonded fuel, a spectrum of potential accidents was identified. This process started with systematically identifying initiating events, subsequent accident progressions, and onsite or offsite releases. Then, based on accident initiators, selected accidents were grouped into the following three categories:

- Natural phenomena (e.g., earthquake, tornado),
- External events (e.g., aircraft crash), and
- Process-related events (e.g., explosion, nuclear criticality, fire, spills).

The potential process-related events were further subdivided based on the impact the accident has on the accident release factors. High energy events would be expected to damage some of the confinement barriers provided in the facility design and will result in release factors that approach unity. Medium energy events may reduce the effectiveness of the barriers, but are not expected to defeat them; while low energy events would have almost no impact on the ability of the confinement barriers to perform their function.

A review of the accident scenarios indicated that only severe accident conditions (e.g., accidents involving confinement failure) could result in a significant release of radioactive material to the environment or an increase in radiation levels. These severe accident conditions are associated with beyond-design-basis events, combinations of events for which the facility was not specifically designed. While these events may have consequences larger than those associated with design-basis events, their frequency is expected to be much lower than the design-basis event frequency. Natural phenomena (e.g., earthquake) and fire accidents creating a direct path for releases to the environment, represented the situation with the most consequences to the public. Some types of accidents, such as procedure violations, spills of small materials containing radioactive particles, and most other types of common human error, occur more frequently than the more severe accidents analyzed. However, these accidents do not involve enough radioactive material or radiation to result in significant release to the environment, although the impact to operational personnel may be as significant as that resulting from beyond-design-basis events. The airborne particles from a process-related accident would normally pass through at least one bank and possibly two to four banks of high-efficiency particulate air filters before entering the environment. Spent nuclear fuel handling operations are performed inside such confinement barriers as hot cells or canyon walls. The hot cells are equipped with safety significant features, such as inert gas atmosphere, pressure control, and heat detection. These features are credited when their operability is not compromised by the sequence of events associated with the accident progression.

While severe accidents (also referred to as beyond-design-basis events) are expected to have the most significant impacts on the population, that is the highest consequences, these accidents may not have as significant a risk impact on all receptors as higher frequency, lower consequence accidents. For this reason, higher frequency accident scenarios were included in the accident analysis. Three categories of accidents were identified, and for each category at least one accident scenario was selected for analysis. The three categories consist of abnormal events (defined as events with a frequency of greater than 1×10^{-3} per year), design-basis events (with frequencies between 1×10^{-3} and 1×10^{-6} per year) and the beyond-design-basis events (frequency less than 1×10^{-6} but limited to those greater than 1×10^{-7} per year).

Based on the review of the existing facility analyses and on guidance provided by the U.S. Department of Energy (DOE) in Section 6.9 of *Recommendations for the Preparation of Environmental Assessments and Environmental Impact Statements* (DOE 1993a), the following types of accidents were selected for each processing technology:

- Explosions
- Nuclear criticality
- Fire
- Earthquake
- Aircraft crash
- Spills/drops

Finally, no specific analyses of the results of terrorist or sabotage acts were considered. This is because the existing security measures in effect at the management sites would essentially preclude any sabotage or terrorist activity. In addition, any acts of terrorism are expected to result in consequences that are bounded by the results of the accident scenarios selected for detailed evaluation.

F.2.2.1 Accident Source Terms and Scenario Description

This section describes the accident scenarios and corresponding source terms developed for ANL-W and SRS. The spectrum of accidents described below were used to determine the incremental consequences (public and worker doses) and risks associated with the treatment of sodium-bonded spent nuclear fuel at each site. These accident scenarios are consistent with those evaluated in either the facility safety analysis report, facility/site environmental reports, or various related DOE safety documents. Secondary accidents were considered when identified in the safety documents. The selected documents were identified and referenced in each of the accident scenarios described. When information was required to further clarify the accident condition, update some of the parameters, and facilitate the evaluation process, additional assumptions were made. Sometimes it was necessary to use different assumptions than those that were used in the referenced report, which are also identified. For example, the material at risk during an earthquake can be different for the treatment in this EIS than those considered in the facility safety analysis report. This change in assumption is necessary because the evaluations in this EIS focus only on the risk resulting from the implementation of alternatives (an incremental risk), and therefore address only the risk associated with the treatment of the sodium-bonded fuel. Cumulative risks can be determined by adding the incremental risks to the existing risks.

F.2.2.1.1 Source Terms

The source term (or building source term) is the amount of respirable radioactive material that is released to the air, in terms of curies or grams, assuming the occurrence of a postulated accident. The airborne source term is typically estimated by the following five-component linear equation:

Source term = MAR × DR × ARF × RF × LPF

where:

MAR	=	Material-at-Risk (grams or curies)
DR	=	Damage Ratio
ARF	=	Airborne Release Fraction (or Airborne Release Rate for continuous release)
RF	=	Respirable Fraction
LPF	=	Leak Path Factor

- **Material at Risk—**The material at risk is the amount of the radionuclides (in curies of activity or grams for each radionuclide) available for release when acted upon by a given physical stress (i.e., an accident). The material at risk is specific to a given process in the facility of interest. It is not necessarily the total quantity of material present, but is that amount of material in the scenario of interest postulated to be available for release.

- **Damage Ratio—**This is the fraction of material exposed to the effects of the energy/force/stress generated by the postulated event. For the accident scenarios discussed in this document, the value of the damage ratio varies from 1×10^{-4} to 1.0.

- **Airborne Release Fraction—**This is the fraction of material that becomes airborne due to the accident. In this analysis, airborne release fraction values from the DOE Handbook on airborne release fraction are used (DOE 1994b).

- **Respirable Fraction—**This is the fraction of the material, with particle size of 10-micrometers (microns) aerodynamic equivalent diameter or less, that could be retained in the respiratory system following inhalation. The respirable fraction values are also taken from the DOE Handbook on airborne release fractions (DOE 1994b).

- **Leak Path Factor—**The leak path factor accounts for the action of removal mechanisms (e.g., containment systems, filtration, deposition) to reduce the amount of airborne radioactivity that is ultimately released to occupied spaces in the facility or the environment. A leak path factor of 1 (i.e., no reduction) is assigned in accident scenarios involving a major failure of confinement barriers.

F.2.2.1.2 Accident Scenarios Description and Source Terms at ANL-W

- **Description of Accident Scenarios for Electrometallurgical Treatment Process—**The electrometallurgical treatment process would occur at the Fuel Conditioning Facility and the Hot Fuel Examination Facility at the ANL-W site. This process is detailed in Appendix C. The accident scenarios, identified in **Table F–2** and defined in the following paragraphs, are applicable to the electrometallurgical treatment process as proposed at ANL-W. This section also provides information addressing the material at risk and the various release fractions used to determine the source term for each accident selected for analysis.

Table F–2 Selected Accident Scenarios for the Electrometallurgical Treatment Process at ANL-W

Scenario	*Frequency (per year)*
Process Related Spills/Drops a. Salt Powder Spill b. Cask Drop c. Salt Canister Transfer Accident, Canister Breach	 1×10^{-2} 1×10^{-2} 1×10^{-7}
Single Container Solid Transuranic Waste Fire	1×10^{-3}
Explosion [a]	N/A
Earthquake (DBE)	2×10^{-4} [b] / 0.008 [c]
Aircraft Crash	6×10^{-7} to 1×10^{-8}
Nuclear Criticality	less than 10^{-7}
Earthquake (BDBE)	1×10^{-5}

DBE = design- (evaluation) basis earthquake, BDBE = beyond-design-basis earthquake.
[a] N/A - The explosion scenario is not applicable to this process.
[b] Design-basis earthquake for the Fuel Conditioning Facility.
[c] Design-basis earthquake for the Hot Fuel Examination Facility.

Each accident scenario description sets the condition of the accident and provides a summary of material involved. As stated earlier, some of these accident scenarios are generic, but their applications are consistent with those evaluated in various ANL-W environmental and safety analyses. These accidents include process-specific as well as generic and process-independent accidents. Tables F–3 through F–8 provide a summary of the accidents analyzed, the material at risk, and the release factors based on the sodium-bonded spent nuclear fuel type that is expected to produce the most significant consequences, typically either Experimental Breeder Reactor-II blanket or driver fuel, for each postulated accident.

- *Operational accident causing salt powder spill in Hot Fuel Examination Facility Main Cell*—Solidified electrorefiner salt is sent from the Fuel Conditioning Facility to the Hot Fuel Examination Facility for processing into the final ceramic waste form. It is brought into the Hot Fuel Examination Facility in the solid form and ground. The grinder is located in the Hot Fuel Examination Facility Main Cell on a raised floor. In this accident scenario, it was assumed that during a transfer operation, the contents of a ground salt container is spilled and the powder spills into the pit beneath the floor. A portion of the salt powder becomes airborne and is carried through the ventilation system to the high-efficiency particulate air filters, and released through the building stack. The release is assumed to occur over a 1-hour period. The frequency of this accident was set at 1×10^{-2} per year, based on the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b).

  The salt is assumed to come from the treatment of 4.45 metric tons of heavy metal of EBR-II blanket elements or 1.1 metric tons of heavy metal of EBR-II driver elements, the point at which the salt is conservatively assumed to have been replaced during processing (Goff et al 1999b). Based on the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b), the material at risk was assumed to be 100 kilograms ground salt containing the radionuclide concentrations as shown in **Table F–3**. Radionuclide distributions were developed for both EBR-II driver and blanket fuels. The radionuclide distribution for driver fuel is based on an average plutonium concentration in electrorefiner salt of 1.76 percent by weight. The radionuclide distribution for blanket fuel is based on an average plutonium concentration in electrorefiner salt of 6.76 percent by weight (Goff et al 1999). Portions of the spilled salt will become airborne. The maximum measured value for the three meter free-fall of dry cohesionless particles, with a mass median diameter of 1 to 2 microns, result in an airborne release fraction of 0.002 and an respirable fraction of 0.3 (DOE 1994b). The median particle size of the salt after grinding is approximately 200 microns with only about 1 percent being of a diameter of less than 20 microns (ANL 1999). The analysis therefore conservatively assumed that about

1 percent of the ground salt would have characteristics capable of resulting in the airborne release fraction and respirable fractions identified above, resulting in a damage ratio of 0.01. The ventilation system and high-efficiency particulate air filters are assumed to function normally. The ventilation system consists of a two-stage high-efficiency particulate air filtration system or equivalent, with a first stage high-efficiency particulate air filter efficiency of 99.9 percent, and a second stage efficiency of 99 percent. The leak path factor through the high-efficiency particulate air filters is therefore 1×10^{-5}.

Table F–3 Material at Risk and Release Fraction Values for the Salt Power Spill Accident at ANL-W

Material at Risk[a]							*Source Terms (curies)*	
Isotope	***Blanket (curies)***	***Driver (curies)***	***DR***	***ARF***	***RF***	***LPF***	***Blanket***	***Driver***
Sr-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	1.0E-05	2.96E-08	2.10E-06
Y-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	1.0E-05	2.96E-08	2.10E-06
I-129	8.79E-04	1.31E-02	0.01	2.0E-03	3.0E-01	1.0E-05	5.27E-14	7.86E-13
Cs-134	8.18E+00	3.13E+02	0.01	2.0E-03	3.0E-01	1.0E-05	4.91E-10	1.88E-08
Cs-137	1.06E+03	3.92E+04	0.01	2.0E-03	3.0E-01	1.0E-05	6.36E-08	2.35E-06
Ba-137M	1.00E+03	3.71E+04	0.01	2.0E-03	3.0E-01	1.0E-05	6.00E-08	2.23E-06
Ce-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	1.0E-05	2.30E-09	3.16E-08
Pr-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	1.0E-05	2.30E-09	3.16E-08
Pm-147	2.48E+02	1.47E+04	0.01	2.0E-03	3.0E-01	1.0E-05	1.49E-08	8.82E-07
Sm-151	6.10E+01	9.48E+02	0.01	2.0E-03	3.0E-01	1.0E-05	3.66E-09	5.69E-08
Eu-154	4.48E+00	1.01E+02	0.01	2.0E-03	3.0E-01	1.0E-05	2.69E-10	6.06E-09
Eu-155	2.94E+01	6.77E+02	0.01	2.0E-03	3.0E-01	1.0E-05	1.76E-09	4.06E-08
Th-228	9.46E-05	9.13E-03	0.01	2.0E-03	3.0E-01	1.0E-05	5.68E-15	5.48E-13
Np-237	7.94E-05	5.13E-02	0.01	2.0E-03	3.0E-01	1.0E-05	4.76E-15	3.08E-12
Pu-238	5.15E+00	6.68E+01	0.01	2.0E-03	3.0E-01	1.0E-05	3.09E-10	4.01E-09
Pu-239	4.13E+02	1.08E+02	0.01	2.0E-03	3.0E-01	1.0E-05	2.48E-08	6.48E-09
Pu-240	2.84E+01	3.67E+00	0.01	2.0E-03	3.0E-01	1.0E-05	1.70E-09	2.20E-10
Pu-241	1.15E+02	8.93E+00	0.01	2.0E-03	3.0E-01	1.0E-05	6.90E-09	5.36E-10
Am-241	9.95E+00	6.94E-02	0.01	2.0E-03	3.0E-01	1.0E-05	5.97E-10	4.16E-12
Am-242M	1.03E-01	5.88E-05	0.01	2.0E-03	3.0E-01	1.0E-05	6.18E-12	3.53E-15

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
[a] Radionuclide Inventory from Appendix D.

- *Cask drop and gaseous fission product release*—Spent nuclear fuel casks will be handled frequently when the sodium-bonded fuel is processed. (Spent nuclear fuel handling at the ANL-W site is not limited to that associated with the treatment of the sodium-bonded spent nuclear fuel. The accident discussed here is intended to address only that portion of the handling activity that can be directly attributed to the treatment of sodium-bonded spent nuclear fuel.) Spent nuclear fuel stored in the Radioactive Scrap and Waste Facility will be transferred to the Fuel Conditioning Facility for processing, and spent nuclear fuel will be received from off site at the Hot Fuel Examination Facility and transferred between the Hot Fuel Examination Facility and the Fuel Conditioning Facility. The HFEF-5 cask would be used to move EBR-II driver and blanket fuels from the Radioactive Scrap and Waste Facility, to the Fuel Conditioning Facility. The postulated accident is described in the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b). The accident involves a dropped cask during unloading, resulting in seal failure and fuel cladding failure sufficient to release gaseous and volatile fission products to the atmosphere. The drop could be initiated by failure of lifting equipment, failure of slings, hooks, or cables, or human error by the lifting equipment operator. The cask drop is conservatively assumed to result in an unfiltered release of gaseous and volatile fission products. The release is assumed to be a puff release at ground level. Dropping of casks, while rare, is nevertheless categorized as being anticipated since such events have happened in the past and may be expected to occur over the lifetime of the facility. The frequency of cask dropping is assumed to be 1×10^{-2} per year, consistent with that used in the Safety Analysis Report for the Hot Fuel Examination Facility (ANL 1998b).

The HFEF-5 cask can contain two EBR-II driver subassemblies. It is conservatively assumed that the equivalent of one subassembly (61 elements) fails in the accident. The material at risk, as shown in **Table F–4**, is the equivalent of one EBR-II driver or one EBR-II blanket subassembly. The damage ratio for the failed elements is assumed to be 1, since all gaseous and volatile fission products conservatively could be released to the cask following cladding failure. The airborne release fraction/respirable fraction for gases is assumed to be 1, and 1×10^{-7} for cesium from the dislodgement of surface contamination (DOE 1995a). The accident is assumed to occur outdoors, so a leak path factor of 1 is assumed.

Table F–4 Material at Risk and Release Fraction Values for the Cask Drop Accident at ANL-W

Material at Risk[a]							*Source Terms (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*	*DR*	*ARF*	*RF*	*LPF*	*Blanket*	*Driver*
H-3[b]	3.35E-01	5.17E+00	1	1	1	1	3.35E-01	5.17E+00
Kr-85	2.44E+00	7.94E+01	1	1	1	1	2.44E+00	7.94E+01
Cs-134	6.30E-01	7.39E+00	1	1.0E-07	1	1	6.30E-08	7.39E-07
Cs-137	8.13E+01	9.28E+02	1	1.0E-07	1	1	8.13E-06	9.28E-05

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF=leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
[a] Data for 1 assembly based on Appendix D data for curie content.
[b] Assumes 1 percent becomes oxidized.

- *Salt spill during transfer from Fuel Conditioning Facility to Hot Fuel Examination Facility*—Solidified electrorefiner salt is sent from the Fuel Conditioning Facility to the Hot Fuel Examination Facility for processing into the final ceramic waste form. It is transferred in the form of large chunks within the HFEF-5 cask. Transfer is via forklift or truck. In this scenario, a severe vehicle accident occurs resulting in breach of the inner and outer salt container. The accident could be caused by operator error or equipment failure. The accident is considered beyond-design-basis due to the durability of the shielded HFEF-5 canister. There will be over 200 transfers of salt from the Fuel Conditioning Facility to the Hot Fuel Examination Facility. A probability of 1×10^{-7} is assumed. The release occurs at ground level with a duration of 1 hour.

  Table F–5 provides the isotopic material at risk for a total material at risk of 20 kilograms of salt per transfer based on discussions with ANL-W personnel. The salt is in the forms of chunks (i.e., ice cube size) and is not combustible. No significant release is assumed from the large pieces. Some of the salt pieces would experience brittle fracture and release of particulate. A brittle fracture particulate fraction for solidified salt is 1×10^{-4} for particles less than 10 microns in diameter (ANL 1998b), therefore, a damage ratio of 1×10^{-4} is assumed. Conservatively, the same airborne release fraction/respirable fraction values as were used for the spill of the salt powder in the Hot Fuel Examination Facility Main Cell were used, that is airborne release fraction for powder is 0.002 and the respirable fraction is 0.3 (DOE 1994b). The accident occurs outdoors, therefore the leak path factor is 1.0.

Table F–5 Material at Risk and Release Fraction Values for the Salt Transfer Accident at ANL-W

Material at Risk[a]							*Source Terms (curies)*	
Isotopes	*Blanket (curies)*	*Driver (curies)*	*DR*	*ARF*	*RF*	*LPF*	*Blanket*	*Driver*
Sr-90	9.85E+01	7.00E+03	1.0E-04	2.0E-03	3.0E-01	1	5.91E-06	4.20E-04
Y-90	9.85E+01	7.00E+03	1.0E-04	2.0E-03	3.0E-01	1	5.91E-06	4.20E-04
I-129	1.76E-04	2.61E-03	1.0E-04	2.0E-03	3.0E-01	1	1.06E-11	1.57E-10
Cs-134	1.64E+00	6.25E+01	1.0E-04	2.0E-03	3.0E-01	1	9.84E-08	3.75E-06
Cs-137	2.11E+02	7.85E+03	1.0E-04	2.0E-03	3.0E-01	1	1.27E-05	4.71E-04
Ba-137M	2.00E+02	7.42E+03	1.0E-04	2.0E-03	3.0E-01	1	1.20E-05	4.45E-04
Ce-144	7.65E+00	1.05E+02	1.0E-04	2.0E-03	3.0E-01	1	4.59E-07	6.30E-06
Pr-144	7.65E+00	1.05E+02	1.0E-04	2.0E-03	3.0E-01	1	4.59E-07	6.30E-06
Pm-147	4.97E+01	2.93E+03	1.0E-04	2.0E-03	3.0E-01	1	2.98E-06	1.76E-04
Sm-151	1.22E+01	1.90E+02	1.0E-04	2.0E-03	3.0E-01	1	7.32E-07	1.14E-05
Eu-154	8.96E-01	2.01E+01	1.0E-04	2.0E-03	3.0E-01	1	5.38E-08	1.21E-06
Eu-155	5.87E+00	1.35E+02	1.0E-04	2.0E-03	3.0E-01	1	3.52E-07	8.10E-06
Th-228	1.89E-05	1.83E-03	1.0E-04	2.0E-03	3.0E-01	1	1.13E-12	1.10E-10
Np-237	1.59E-05	1.03E-02	1.0E-04	2.0E-03	3.0E-01	1	9.54E-13	6.18E-10
Pu-238	1.03E+00	1.34E+01	1.0E-04	2.0E-03	3.0E-01	1	6.18E-08	8.04E-07
Pu-239	8.25E+01	2.16E+01	1.0E-04	2.0E-03	3.0E-01	1	4.95E-06	1.30E-06
Pu-240	5.68E+00	7.33E-01	1.0E-04	2.0E-03	3.0E-01	1	3.41E-07	4.40E-08
Pu-241	2.30E+01	1.79E+00	1.0E-04	2.0E-03	3.0E-01	1	1.38E-06	1.07E-07
Am-241	1.99E+00	1.39E-02	1.0E-04	2.0E-03	3.0E-01	1	1.19E-07	8.34E-10
Am-242M	2.06E-02	1.18E-05	1.0E-04	2.0E-03	3.0E-01	1	1.24E-09	7.08E-13

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF=leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
[a] The material at risk is the isotope in 20 kilograms of salt, which is 20 percent of those given in Table F-3.

- *Solid transuranic waste fire*—Transuranic waste is generated as a result of treatment operations, as well as for other operations at ANL-W. These wastes are placed in containers and temporarily stored (staged) at ANL-W pending shipment to the Radioactive Waste Management Complex. A fire is postulated to occur in a 1.2 × 1.2 × 2.4-meter (4 × 4 × 8-foot) transuranic waste box due to spontaneous combustion, pyrophoric material, vehicle accident, electrical failure, or poor housekeeping. The fire consumes the contents of one box of staged transuranic waste. The accident is assumed to occur outdoors during handling. The release occurs at ground level over 1 hour. An accident frequency in the range of Unlikely (1×10^{-4} to 1×10^{-2}) is documented in the Final Safety Analysis Report for the Fuel Conditioning Facility (ANL 1998a). Here, the accident was assumed to have a frequency of 1×10^{-3} per year.

  The material at risk, as shown in **Table F–6**, was assumed to be one box of transuranic waste. The waste boxes are loaded with 1/20th of 0.34 curies of alpha activity, as described in the Fuel Conditioning Facility Final Safety Analysis Report (ANL 1998a). The material at risk is 0.017 curies of transuranic nuclides, with the nuclide distribution associated with the generic contents of a transuranic waste container. The damage ratio is assumed to be 1.0, since all waste in the container is assumed to be involved in the fire. The DOE Handbook *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE 1994b) recommends an airborne release fraction of 5×10^{-4} and a respirable fraction of 1.0 for burning of surface contaminated wastes. The leak path factor is assumed to be 1.0. No credit is taken for building confinement.

Table F–6 Material at Risk and Release Fraction Values for the Transuranic Waste Fire Accident at ANL-W

Material at Risk [a]						
Isotope	***curies***	***DR***	***ARF***	***RF***	***LPF***	***Source Term (curies)***
Pu-238	1.53E-04	1	5.00E-04	1.00E+01	1	7.67E-08
Pu-239	1.23E-02	1	5.00E-04	1.00E+01	1	6.15E-06
Pu-240	8.46E-04	1	5.00E-04	1.00E+01	1	4.23E-07
Pu-241	3.43E-03	1	5.00E-04	1.00E+01	1	1.72E-06
Am-241	2.66E-04	1	5.00E-04	1.00E+01	1	1.33E-07

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF=leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
[a] The MAR is for a generic waste package, not for any specific spent nuclear fuel.

- *Design-Basis Seismic Event - Multi-facility Effects*—In the Fuel Conditioning Facility, the argon cell contains the equipment for processing sodium-bonded spent nuclear fuel into salt and metal waste forms and uranium metal product. All operations involving bare fuels are conducted in the argon cell because the inert atmosphere precludes pyrophoric metal fire. Fire cannot occur unless sufficient oxygen enters the cell through a cell breach. The walls, ceiling, and floor of the argon cell are constructed from reinforced concrete with thicknesses ranging from 1.2 to 1.5 meters (4 to 5 feet). It also has a gas-tight steel lining. It is assumed that the accident occurs during electrometallurgical treatment operations. Chopped fuel, electrorefiner salts, cathodes, and anodes are all present in the argon cell. At the Fuel Conditioning Facility, a seismic event results in cell breach and inlet of air to the cell. The air in the cell causes pyrophoric metals to ignite and burn. The electrorefiners are seismically qualified, and no spill of molten salt is postulated. The Fuel Conditioning Facility Safety Exhaust System is seismically qualified, and is assumed to function as designed, filtering the cell atmosphere prior to release through the Fuel Conditioning Facility stack.

The Final Safety Analysis Report for the Fuel Conditioning Facility (ANL 1998a) identifies the seismic design goal for the facility to be the ability to withstand a 0.21g design-basis seismic event. This event is identified as having a return frequency of 2×10^{-4} per year. The safety exhaust system will remain operational, although breaches may occur in the argon-cell boundary, after a design-basis earthquake. The Safety Exhaust Building, which includes the high-efficiency particulate air filters, is designed to withstand an earthquake of 0.24 g. In the Hot Fuel Examination Facility, grinding of salt into powder is assumed to be occurring in the main cell. The grinder is located in the Hot Fuel Examination Facility main cell on a raised floor consisting of steel plates resting on supports. Underneath the floor is a 2.4-meter- (8-foot-) deep pit that houses the ventilation ductwork and high-efficiency particulate air filters. At the Hot Fuel Examination Facility, a seismic event causes the vessel containing ground salt to topple and the powder to spill out. Since the ventilation system is not seismically qualified, it is assumed to fail and result in an unfiltered release. It is also assumed that the seismic event would cause a loss of electrical power, which would also result in failure of the ventilation system. The main cell breaches at piping or ventilation penetrations, providing a release path for the suspended powder. The releases occur over a 1-hour period, and are modeled as a ground-level release. The Hot Fuel Examination Facility has been analyzed for a 0.14 g design-basis seismic event, an event with a return frequency of 0.001 per year and a performance goal of 1×10^{-4} per year. No records exist on the original equipment design. However, all major systems are known to have survived the 0.03 Borah earthquake in 1983, an event with a return frequency of 0.008 per year. While it is expected that the equipment would survive the 0.14 g earthquake, the 0.008 per year (ANL 1998b) seismic event frequency has been conservatively used to represent the upper bound of the design-basis seismic event which would result in a salt powder spill and the possible failure of the ventilation system. This frequency is nearly two orders of magnitude higher than the 0.21 g earthquake that could impact both the Hot Fuel Examination Facility and the Fuel Conditioning Facility. Therefore, 0.008 per year is used for the design-basis seismic accident frequency.

In the Fuel Conditioning Facility, the material at risk is chopped fuel and cathodes in the argon cell at the time of the accident. **Table F–7** provides material at risk values for the isotopes of concern. The bounding inventory is 20 kilograms (37 pounds) of chopped fuel and 25 kilograms (44 pounds) in two solid electrodes. The solid cathodes contain 17 kilograms uranium. Uranium is considered a toxic chemical in the consequence assessment. The total uranium material at risk is 19 kilograms. During the postulated event, 100 kilograms (220 pounds) of solidified salt powder with the same concentration of radionuclides as described above for the powder is spilled in the Hot Fuel Examination Facility main cell. For the metal fire in the Fuel Conditioning Facility argon cell, the damage ratio is assumed to be 1.0, since all materials in the material at risk are released to the cells in the accident. As with the previously discussed salt powder spill, only a fraction of the material is small enough to be capable of resuspended, and the damage ratio is 0.01. For the Fuel Conditioning Facility, the airborne release fraction/respirable fraction is 1.0 for krypton-85, 0.00025 for cesium and 2.5×10^{-6} for strontium, uranium, and transuranic waste nuclides (DOE 1995a). For the Hot Fuel Examination Facility powder spill within the cell, an airborne release fraction of 0.002 and an respirable fraction of 0.3 are assumed (DOE 1994b). These are the same values as used for the salt powder spill accident described above. For the Fuel Conditioning Facility, the safety exhaust system remains functional, and the release is filtered through high-efficiency particulate air filters. A leak path factor of 1×10^{-5} is assumed for all particulates. The Hot Fuel Examination Facility leak path for the release is through three enclosures before reaching the outside: main cell, ducts and pipes, and the building. Consistent with the facility safety analysis report assumption, a leak path factor of 0.5 is assigned to each enclosure for plate out and settling of the airborne powder. Therefore, the total leak path factor is $0.5 \times 0.5 \times 0.5 = 0.125$.

Table F–7 Material at Risk and Release Fraction Values for the Design Basis Seismic Event at ANL-W

Accident	*Material at Risk*			*DR*	*ARF*	*RF*	*LPF*	*Source Term (curies)*	
	Isotope	*Blanket (curies)*	*Driver (curies)*					*Blanket*	*Driver*
Seismic event and salt powder spill at the Hot Fuel Examination Facility	Sr-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	0.125	3.70E-04	2.63E-02
	Y-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	0.125	3.70E-04	2.63E-02
	I-129	8.79E-04	1.31E-02	0.01	2.0E-03	3.0E-01	0.125	6.59E-10	9.83E-09
	Cs-134	8.18E+00	3.13E+02	0.01	2.0E-03	3.0E-01	0.125	6.14E-06	2.35E-04
	Cs-137	1.06E+03	3.92E+04	0.01	2.0E-03	3.0E-01	0.125	7.95E-04	2.94E-02
	Ba-137M	1.00E+03	3.71E+04	0.01	2.0E-03	3.0E-01	0.125	7.50E-04	2.78E-02
	Ce-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	0.125	2.87E-05	3.95E-04
	Pr-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	0.125	2.87E-05	3.95E-04
	Pm-147	2.48E+02	1.47E+04	0.01	2.0E-03	3.0E-01	0.125	1.86E-04	1.10E-02
	Sm-151	6.10E+01	9.48E+02	0.01	2.0E-03	3.0E-01	0.125	4.58E-05	7.11E-04
	Eu-154	4.48E+00	1.01E+02	0.01	2.0E-03	3.0E-01	0.125	3.36E-06	7.58E-05
	Eu-155	2.94E+01	6.77E+02	0.01	2.0E-03	3.0E-01	0.125	2.21E-05	5.08E-04
	Th-228	9.46E-05	9.13E-03	0.01	2.0E-03	3.0E-01	0.125	7.10E-11	6.85E-09
	Np-237	7.94E-05	5.13E-02	0.01	2.0E-03	3.0E-01	0.125	5.96E-11	3.85E-08
	Pu-238	5.15E+00	6.68E+01	0.01	2.0E-03	3.0E-01	0.125	3.86E-06	5.01E-05
	Pu-239	4.13E+02	1.08E+02	0.01	2.0E-03	3.0E-01	0.125	3.10E-04	8.10E-05
	Pu-240	2.84E+01	3.67E+00	0.01	2.0E-03	3.0E-01	0.125	2.13E-05	2.75E-06
	Pu-241	1.15E+02	8.93E+00	0.01	2.0E-03	3.0E-01	0.125	8.63E-05	6.70E-06
	Am-241	9.95E+00	6.94E-02	0.01	2.0E-03	3.0E-01	0.125	7.46E-06	5.21E-08
	Am-242M	1.03E-01	5.88E-05	0.01	2.0E-03	3.0E-01	0.125	7.73E-08	4.41E-11
Seismic event and metal fire in the Fuel Conditioning Facility argon cell	H-3	1.42E-01	2.46E+01	1	1	1	1	1.42E-01	2.46E+01
	C-14	1.19E-03	3.98E+03	1	2.5E-06	1	1E-05	2.99E-14	9.95E-08
	Fe-55	1.80E+00	9.74E+01	1	2.5E-06	1	1E-05	4.51E-11	2.44E-09
	Ni-63	6.12E-02	4.58E+00	1	2.5E-06	1	1E-05	1.53E-12	1.15E-10
	Kr-85	1.04E+00	3.78E+02	1	1	1	1	1.04E+00	3.78E+02
	Sr-90	1.61E+01	3.94E+03	1	2.5E-06	1	1E-05	4.04E-10	9.85E-08
	Y-90	1.61E+01	3.94E+03	1	2.5E-06	1	1E-05	4.04E-10	9.85E-08
	Ru-106	2.70E+00	3.02E+01	1	2.5E-04	1	1E-05	6.75E-09	7.55E-08
	Rh-106	2.70E+00	3.02E+01	1	2.5E-06	1	1E-05	6.75E-11	7.55E-10

Accident	Material at Risk			DR	ARF	RF	LPF	Source Term (curies)	
	Isotope	Blanket (curies)	Driver (curies)					Blanket	Driver
	Cd-113M	1.42E-02	9.28E-01	1	2.5E-06	1	1E-05	3.56E-13	2.32E-11
	Sb-125	4.62E-01	5.92E+01	1	2.5E-06	1	1E-05	1.16E-11	1.48E-09
	Te-125M	1.90E-01	2.46E+01	1	2.5E-06	1	1E-05	4.76E-12	6.15E-10
	I-129	2.88E-05	1.47E-03	1	1	1	1	2.88E-05	1.47E-03
	Cs-134	2.68E-01	3.52E+01	1	2.5E-04	1	1E-05	6.70E-10	8.80E-08
	Cs-137	3.46E+01	4.42E+03	1	2.5E-04	1	1E-05	8.65E-08	1.11E-05
	Ba-137M	3.28E+01	4.18E+03	1	2.5E-06	1	1E-05	8.20E-10	1.05E-07
	Ce-144	1.25E+00	5.92E+01	1	2.5E-06	1	1E-05	3.14E-11	1.48E-09
	Pr-144	1.25E+00	5.92E+01	1	2.5E-6	1	1E-05	3.14E-11	1.48E-09
	Pm-147	8.14E+00	1.65E+03	1	2.5E-06	1	1E-05	2.04E-10	4.13E-08
	Sm-151	2.00E+00	1.07E+02	1	2.5E-06	1	1E-05	5.00E-11	2.67E-09
	Eu-154	1.47E-01	1.13E+01	1	2.5E-06	1	1E-05	3.67E-12	2.84E-10
	Eu-155	9.62E-01	7.62E+01	1	2.5E-06	1	1E-05	2.41E-11	1.91E-09
	Th-228	3.10E-06	1.03E-03	1	2.5E-06	1	1E-05	7.75E-17	2.57E-14
	U-234	2.66E-05	8.08E-01	1	2.5E-06	1	1E-05	6.65E-16	2.02E-11
	U-235	7.54E-05	2.62E-02	1	2.5E-06	1	1E-05	1.89E-15	6.55E-13
	U-236	8.48E-05	2.42E-02	1	2.5E-06	1	1E-05	2.12E-15	6.05E-13
	U-238	6.54E-03	2.22E-03	1	2.5E-06	1	1E-05	1.64E-13	5.55E-14
	Np-237	2.60E-06	5.78E-03	1	2.5E-06	1	1E-05	6.50E-17	1.45E-13
	Pu-238	1.88E-01	3.32E+00	1	2.5E-06	1	1E-05	4.70E-12	8.30E-11
	Pu-239	1.51E+01	5.38E+00	1	2.5E-06	1	1E-05	3.77E-10	1.35E-10
	Pu-240	1.04E+00	1.82E-01	1	2.5E-06	1	1E-05	2.59E-11	4.56E-12
	Pu-241	4.20E+00	4.44E-01	1	2.5E-06	1	1E-05	1.05E-10	1.11E-11
	Am-241	3.26E-01	7.82E-03	1	2.5E-06	1	1E-05	8.15E-12	1.96E-13
	Am-242M	3.38E-03	6.62E-06	1	2.5E-06	1	1E-05	8.45E-14	1.66E-16

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

- *Aircraft crash*—The potential for an aircraft crash was evaluated. The methodology for evaluating the likelihood of an aircraft crash is documented in DOE Standard *Accident Analysis for Aircraft Crash into Hazardous Facilities* (DOE 1996c). At INEEL, the probability of a small and large aircraft crash is 9×10^{-5} and 4×10^{-7} crashes per square mile per year, respectively. Using guidance in this DOE standard, the effective area of the Fuel Conditioning Facility was calculated accounting for aircraft wing span and potential skid distance. The effective area of the Fuel Conditioning Facility is about 0.03 square miles for a large aircraft, and 0.007 square miles for a small aircraft. The effective area of the Fuel Conditioning Facility is conservative because the area of the air cell and argon cell, where the hazardous material are contained, are smaller than the total area of the building. Multiplying effective area by INEEL-specific crash rates, gives an estimated probability of a crash into the Fuel Conditioning Facility of 1×10^{-8} for large aircraft and 6×10^{-7} for small aircraft. Comparable probabilities are applicable to the Hot Fuel Examination Facility. A large aircraft crash is not reasonably foreseeable, and given the 1.2 to 1.5-meter- (4 to 5-foot-) thick walls of the cells and the "buffer" provided by the building exterior walls, crash of a small aircraft is unlikely to result in any damage to the cells. Damage from the more probable seismic events analyzed are considered to bound the damage that could result from a small aircraft crash. Also, seismic events affect more than one facility, where an aircraft crash could only affect one facility. Therefore, an aircraft crash is not analyzed separately.

- *Nuclear Criticality*—The potential for a nuclear criticality was considered in the accident analysis. Nuclear criticality has been evaluated in the safety analyses documented for the ANL-W facilities, as required by DOE. The existing safety analyses conclude that nuclear criticality is beyond the design-basis of the facilities proposed for the electrometallurgical treatment alternative, and therefore has a probability of less than 1×10^{-6} per year. This conclusion is based on a lack of nuclear moderator materials, equipment design, and inventory controls, as well as numerous other administrative controls and operating procedures. The intent of the process is to dilute, rather than concentrate, fissile materials. Fuel storage racks and processing equipment are designed to maintain their safety function during the design-basis seismic event. Even in a beyond-design-basis earthquake (maximum frequency 1×10^{-5} per year), nuclear materials would have to come together in an ideal critical array in order for criticality to be possible. For example, it would require more than the equivalent of 10 EBR-II driver assemblies (610 individual elements) in an ideal geometric configuration to create a potential criticality hazard. The conditional probability of this configuration, given a beyond-design-basis seismic event, is estimated to be no greater than 1×10^{-2}. Therefore, criticality is not considered to be reasonably foreseeable, and is not analyzed quantitatively.

- *Beyond-Design-Basis Seismic Event*—The scenario is similar to the design-basis seismic event, except that the Safety Exhaust System is not assumed to function at the Fuel Conditioning Facility, and an electrorefiner is assumed to spill its molten salt. Also, since fuel is stored in both the Fuel Conditioning Facility and the Hot Fuel Examination Facility, the equivalent of twelve assemblies of EBR-II driver fuel are assumed to experience cladding failure and release of gaseous and volatile fission products. All releases are modeled as ground-level releases. The Fuel Conditioning Facility natural phenomena hazard performance goal is a frequency of 10^{-5} (DOE 1994a). (The Hot Fuel Examination Facility goal is 1×10^{-4}.) The performance goal can be interpreted as the frequency level at which facility damage will initiate. The Fuel Conditioning Facility and Safety Exhaust System are not expected to suffer damage from seismic events with frequencies higher than this. The Fuel Conditioning Facility horizontal acceleration seismic design-basis is 0.21 g, and the newer safety equipment building is designed for a 0.24 g horizontal acceleration. An 0.24 g peak acceleration corresponds to an earthquake frequency at ANL-W of approximately 1×10^{-4} per year (WCFS 1996). An earthquake with a peak ground acceleration of 0.24 g may damage the Fuel Conditioning Facility structure. Therefore, the upper bound for the beyond-design-basis seismic event frequency has been assumed to correspond to the frequency of the performance goal, 10^{-5} per year.

The material at risk, provided in **Table F–8**, is the same as for the design-basis event, with the addition of fuel elements and subassemblies in storage. Although the electrorefiners are seismically qualified, one of the two electrorefiners in the Fuel Conditioning Facility argon cell is conservatively assumed to spill its molten salt. It is assumed that 700 kilograms (1,540 pounds) of salt is fully loaded with radionuclides and about to be replaced at the time of the accident. The damage ratio for all but the fuel assemblies in storage is assumed to be 1.0 as in the design-basis seismic event. Fuel assemblies are stored in racks with cladding intact. In the seismic event, some can be expected to fall out of the rack or be hit by falling debris, but it is not reasonable to assume all assemblies would be damaged. It is assumed that 10 percent, or 12, of the assemblies stored in the cells at the time of the seismic event experience cladding failure and release of gaseous and volatile fission products. The airborne release fraction/respirable fraction is the same as for the design-basis seismic event, with the addition of krypton and cesium from the failed EBR-II driver assemblies. The airborne release fraction/respirable fraction for krypton and tritium, both elements in the gaseous state, is 1.0. For the molten salt spill, the airborne release fraction/respirable fraction for viscous solutions (DOE 1994b) are used: 4×10^{-6} for the airborne release fraction and 0.8 for the respirable fraction. The forces associated with the beyond-design-basis earthquake are assumed to result in the failure of confinement integrity. The cells are assumed to experience major failure, and the release is directly to the atmosphere. The leak path factor is 1.0.

Table F–8 Material at Risk and Release Fraction Values Assumed for the Beyond-Basis Seismic Event at ANL-W

	Material at Risk[a]							Source Term (curies)	
Accident	Isotope	Blanket (Ci)	Driver (Ci)	DR	ARF	RF	LPF	Blanket	Driver
Beyond-design-basis seismic event and salt powder spill in the Hot Fuel Examination Facility	Sr-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	1	2.96E-03	2.10E-01
	Y-90	4.93E+02	3.50E+04	0.01	2.0E-03	3.0E-01	1	2.96E-03	2.10E-01
	I-129	8.79E-04	1.31E-02	0.01	2.0E-03	3.0E-01	1	5.27E-09	7.86E-08
	Cs-134	8.18E+00	3.13E+02	0.01	2.0E-03	3.0E-01	1	4.91E-05	1.88E-03
	Cs-137	1.06E+03	3.92E+04	0.01	2.0E-03	3.0E-01	1	6.36E-03	2.35E-01
	Ba-137M	1.00E+03	3.71E+04	0.01	2.0E-03	3.0E-01	1	6.00E-03	2.23E-01
	Ce-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	1	2.30E-04	3.16E-03
	Pr-144	3.83E+01	5.26E+02	0.01	2.0E-03	3.0E-01	1	2.30E-04	3.16E-03
	Pm-147	2.48E+02	1.47E+04	0.01	2.0E-03	3.0E-01	1	1.49E-03	8.82E-02
	Sm-151	6.10E+01	9.48E+02	0.01	2.0E-03	3.0E-01	1	3.66E-04	5.69E-03
	Eu-154	4.48E+00	1.01E+02	0.01	2.0E-03	3.0E-01	1	2.69E-05	6.06E-04
	Eu-155	2.94E+01	6.77E+02	0.01	2.0E-03	3.0E-01	1	1.76E-04	4.06E-03
	Th-228	9.46E-05	9.13E-03	0.01	2.0E-03	3.0E-01	1	5.68E-10	5.48E-08
	Np-237	7.94E-05	5.13E-02	0.01	2.0E-03	3.0E-01	1	4.76E-10	3.08E-07
	Pu-238	5.15E+00	6.68E+01	0.01	2.0E-03	3.0E-01	1	3.09E-05	4.01E-04
	Pu-239	4.13E+02	1.08E+02	0.01	2.0E-03	3.0E-01	1	2.48E-03	6.48E-04
	Pu-240	2.84E+01	3.67E+00	0.01	2.0E-03	3.0E-01	1	1.70E-04	2.20E-05
	Pu-241	1.15E+02	8.93E+00	0.01	2.0E-03	3.0E-01	1	6.90E-04	5.36E-05
	Am-241	9.95E+00	6.94E-02	0.01	2.0E-03	3.0E-01	1	5.97E-05	4.16E-07
	Am-242M	1.03E-01	5.88E-05	0.01	2.0E-03	3.0E-01	1	6.18E-07	3.53E-10
Beyond-design-basis seismic event and metal fire in the Fuel Conditioning Facility argon cell	H-3	1.42E-01	2.46E+01	1	1	1	1	1.42E-01	2.46E+01
	C-14	1.19E-03	3.98E+03	1	2.5E-06	1	1	2.99E-09	9.95E-03
	Fe-55	1.80E+00	9.74E+01	1	2.5E-06	1	1	4.51E-06	2.44E-04
	Ni-63	6.12E-02	4.58E+02	1	2.5E-06	1	1	1.53E-07	1.15E-03
	Kr-85	1.04E+00	3.78E+02	1	1	1	1	1.04E+00	3.78E+02
	Sr-90	1.61E+01	3.94E+03	1	2.5E-06	1	1	4.04E-05	9.85E-03
	Y-90	1.61E+01	3.94E+03	1	2.5E-06	1	1	4.04E-05	9.85E-03
	Ru-106	2.70E+00	3.02E+01	1	2.5E-04	1	1	6.75E-04	7.55E-03
	Rh-106	2.70E+00	3.02E+01	1	2.5E-06	1	1	6.75E-06	7.55E-05
	Cd-113M	1.42E-02	9.28E-01	1	2.5E-06	1	1	3.56E-08	2.32E-06
	Sb-125	4.62E-01	5.92E+01	1	2.5E-06	1	1	1.15E-06	1.48E-04
	I-129	1.90E-01	2.46E+01	1	2.5E-06	1	1	4.76E-07	6.15E-05
	Te-125M	2.88E-05	1.47E-03	1	1	1	1	2.88E-05	1.47E-03
	Cs-134	2.68E-01	3.52E+01	1	2.5E-04	1	1	6.70E-05	8.80E-03
	Cs-137	3.46E+01	4.42E+03	1	2.5E-04	1	1	8.65E-03	1.10E+00
	Ba-137M	3.28E+01	4.18E+03	1	2.5E-06	1	1	8.20E-05	1.05E-02
	Ce-144	1.25E+00	5.92E+01	1	2.5E-06	1	1	3.14E-06	1.48E-04
	Pr-144	1.25E+00	5.92E+01	1	2.5E-06	1	1	3.14E-06	1.48E-04
	Pm-147	8.14E+00	1.65E+03	1	2.5E-06	1	1	2.04E-05	4.13E-03
	Sm-151	2.00E+00	1.07E+02	1	2.5E-06	1	1	5.00E-06	2.67E-04
	Eu-154	1.47E-01	1.13E+01	1	2.5E-06	1	1	3.67E-07	2.84E-05
	Eu-155	9.62E-01	7.62E+01	1	2.5E-06	1	1	2.41E-06	1.91E-04
	Th-228	3.10E-06	1.03E-03	1	2.5E-06	1	1	7.75E-12	2.57E-09
	Np-237	2.66E-05	8.08E-01	1	2.5E-06	1	1	6.65E-11	2.02E-06
	U-234	7.54E-05	2.62E-02	1	2.5E-06	1	1	1.89E-10	6.55E-08
	U-235	8.48E-05	2.42E-02	1	2.5E-06	1	1	2.12E-10	6.05E-08

Accident	Material at Risk[a]			DR	ARF	RF	LPF	Source Term (curies)	
	Isotope	Blanket (Ci)	Driver (Ci)					Blanket	Driver
Beyond-design-basis seismic event and metal fire in the Fuel Conditioning Facility argon cell (cont'd)	U-236	6.54E-03	2.22E-03	1	2.5E-06	1	1	1.64E-08	5.55E-09
	U-238	2.60E-06	5.78E-03	1	2.5E-06	1	1	6.50E-12	1.45E-08
	Pu-238	1.88E-01	3.32E+00	1	2.5E-06	1	1	4.70E-07	8.30E-06
	Pu-239	1.51E+01	5.38E+00	1	2.5E-06	1	1	3.77E-05	1.35E-05
	Pu-240	1.04E+00	1.82E-01	1	2.5E-06	1	1	2.59E-06	4.56E-07
	Pu-241	4.20E+00	4.44E-01	1	2.5E-06	1	1	1.05E-05	1.11E-06
	Am-241	3.26E-01	7.82E-03	1	2.4E-06	1	1	8.15E-07	1.96E-08
	Am-242M	3.38E-03	6.62E-06	1	2.5E-06	1	1	8.45E-09	1.66E-11
Beyond-design-basis seismic event and the liquid salt spill at Fuel Conditioning Facility	Sr-90	3.45E+03	2.45E+05	1	4.0E-06	0.8	1	1.10E-02	7.84E-01
	Y-90	3.45E+03	2.45E+05	1	4.0E-06	0.8	1	1.10E-02	7.84E-01
	I-129	6.15E-03	9.17E-02	1	4.0E-06	0.8	1	1.97E-06	2.93E-05
	Cs-134	5.73E+01	2.19E+03	1	4.0E-06	0.8	1	1.83E-02	7.01E-01
	Cs-137	7.42E+03	2.74E+05	1	4.0E-06	0.8	1	2.37E+00	8.78E+01
	Ba-137M	7.00E+03	2.60E+05	1	4.0E-06	0.8	1	2.24E-04	8.31E-01
	Ce-144	2.68E+02	3.68E+03	1	4.0E-06	0.8	1	8.58E-04	1.18E-02
	Pr-144	2.68E+02	3.68E+03	1	4.0E-06	0.8	1	8.58E-04	1.18E-02
	Pm-147	1.74E+03	1.03E+05	1	4.0E-06	0.8	1	5.56E-03	3.29E-01
	Sm-151	4.27E+02	6.64E+03	1	4.0E-06	0.8	1	1.37E-03	2.12E-02
	Eu-154	3.14E+01	7.07E+02	1	4.0E-06	0.8	1	1.00E-04	2.26E-03
	Eu-155	2.06E+02	4.74E+03	1	4.0E-06	0.8	1	6.59E-04	1.52E-02
	Th-228	6.62E-04	6.39E-02	1	4.0E-06	0.8	1	2.12E-09	2.05E-07
	Np-237	5.56E-04	3.59E-01	1	4.0E-06	0.8	1	1.78E-09	1.15E-06
	Pu-238	3.61E+01	4.68E+02	1	4.0E-06	0.8	1	1.15E-04	1.50E-03
	Pu-239	2.89E+03	7.56E+02	1	4.0E-06	0.8	1	9.25E-03	2.42E-03
	Pu-240	1.99E+02	2.57E+01	1	4.0E-06	0.8	1	6.36E-04	8.22E-05
	Pu-241	8.05E+02	6.25E+01	1	4.0E-06	0.8	1	2.58E-03	2.00E-04
	Am-241	6.97E+01	4.86E-01	1	4.0E-06	0.8	1	2.23E-04	1.55E-06
	Am-242M	7.21E-01	4.12E-04	1	4.0E-06	0.8	1	2.31E-06	1.32E-09
Beyond-design-basis seismic event and stored fuel assembly cladding failure	H-3	.335	62	1	1	1	1	.335	62
	Kr-85	2.44	953	1	1	1	1	2.44	953

Ci = curies, DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation ($1.0E\text{-}05 = 1.0 \times 10^{-5}$).
[a] Radionuclide Inventory from Appendix D.

❒ **Description of Accident Scenarios for the Melt and Dilute Process at the ANL-W Site—**The melt and dilute process would occur in the Hot Fuel Examination Facility hot cell at ANL-W. Two melt and dilute process options are considered for ANL-W: 1) melt and dilute cleaned (removed metallic sodium), blanket fuel, and 2) melt and dilute cleaned blanket fuel and cleaned (to extent possible) driver fuel (see Appendix C for more details). Metallic irons would be added to both process options to form a more corrosive resistance metal waste. Both options would occur at a temperature range of 1,000 to 1,400 °C (1,832 to 2,552 °F). For analysis purposes, it was assumed that on the average, 120 batches of melt and dilute process could be performed per year, considering an 80 percent availability and a 3-batches-per-week operation. Each batch processes about 60 kilograms (132 pounds) of heavy metal of blanket fuel or about 16 kilograms of driver fuel (diluted with depleted uranium to a 60-kilograms equivalent heavy metal). This leads to 8 years of operations for processing blanket fuels and 2 years of processing for driver fuel. Several processes are available for cleaning of the spent nuclear fuel, the process that appears to be most practical involves the evaporation and vaporization of the sodium metal. The fuel elements would be heated to a temperature above the 200 °C (392 °F) melting point of sodium and the molten sodium drained into a collection tank. The temperature of this bulk sodium would be raised to 500 °C (932 °F), volatilizing the sodium and separating the entrained cesium from the sodium. (See Appendix C for a more detailed description of this process.)

Table F–9 identifies a list of accident scenarios that were considered to be applicable to the melt and dilute process as proposed at ANL-W. These scenarios are based on the analysis of the melt and dilute process provided in the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998). The accident scenarios and the corresponding source terms have been modified to reflect the specifics associated with the design of the facility (Hot Fuel Examination Facility), the characteristics of the fuel type being processed, the material at risk, and the related release fractions.

Table F–9 Selected Accident Scenarios Melt and Dilute at ANL-W

Scenario	*Frequency (per year)*
Nuclear Criticality	0.0003
Cask Drop	0.01
Waste Handling Accident	0.024
Sodium Fire[a]	0.008
Aircraft Crash	6×10^{-7} to 1×10^{-8}
Design Basis Earthquake	0.008

[a] This event is evaluated as being a direct consequence of the design-basis earthquake.

Each accident scenario description sets the condition of the accident and provides a summary of material involved. The following paragraphs provide a summary of the accidents analyzed, the material at risk and the release factors, for the EBR-II blanket, (the Fermi-1 blanket fuel has a very low radioactive inventory), and EBR-II driver fuel.

- *Nuclear Criticality*—A criticality accident could result from the processing of multiple batches, double batching, of fissile material to the melter. This accident was considered for the driver fuel only. The criticality was assumed to consist of 5×10^{17} fissions, based on a process criticality fission yield (DOE 1998). The Hot Fuel Examination Facility structure would not be compromised and its ventilation system would be expected to continue to function after a criticality event. Procedural controls will be used to prevent such an accident. Therefore, such an accident would be the result of a combination of human errors, as all criticality controls are designed to meet double contingency requirements. The Hot Fuel Examination Facility (ANL 1998b) identifies a criticality event as an incredible event, i.e., assigns it a frequency of less than 1×10^{-6} per year. However, this Safety Analysis Report does not specifically address melt and dilute operations. A criticality event for the melt and dilute process has been addressed for the SRS melt and dilute process (DOE 1998) and for consistency between alternatives, this analysis has been adapted. Based on the assumption of approximately 100 batching operations per year and the frequency analysis for this type of processing criticality at SRS (DOE 1998), the expected frequency of this event is 0.0003 per year to melt and dilute operations at ANL-W. The material at risk and release fractions are provided in **Table F–10**. The damage ratio and leak path factor for the volatile, gaseous fission products have conservatively been assumed to be 1.0. A respirable fraction value of 1.0 was also used. The airborne release fraction values range from 0.5 to 0.05 (DOE 1994b).

Table F–10 Melt and Dilute Process Material at Risk and Release Fraction Values for the Nuclear Criticality Event at ANL-W

Material at Risk						
Isotope	***curies***	***DR***	***ARF***	***RF***	***LPF***	***Source Term (curies)***
Br-83	4.90	1	0.05	1	1	0.25
Br-84	16.3	1	0.05	1	1	0.82
Kr-83M	1.50	1	0.5	1	1	0.75
Kr-85M	7.2	1	0.5	1	1	3.6
Kr-87	32.8	1	0.5	1	1	17
Kr-88	32.9	1	0.5	1	1	17
Kr-89	1820	1	0.5	1	1	910
Te-129	2.70	1	0.07	1	1	0.19
Te-131	57.5	1	0.07	1	1	4.0
Te-131M	0.320	1	0.07	1	1	0.022
Te-132	1.60	1	0.07	1	1	0.11
Te-133	25.7	1	0.07	1	1	1.8
Te-133M	30.3	1	0.07	1	1	2.1
Te-134	90.5	1	0.07	1	1	6.3
I-131	0.212	1	0.05	1	1	0.011
I-132	0.855	1	0.05	1	1	0.043
I-133	6.80	1	0.05	1	1	0.34
I-134	98.0	1	0.05	1	1	4.9
I-135	22.1	1	0.05	1	1	1.1
Xe-133	.026	1	0.5	1	1	0.013
Xe-135	2.61	1	0.5	1	1	1.3
Xe-135M	23.9	1	0.5	1	1	0.12
Xe-137	1940	1	0.5	1	1	0.097
Xe-138	665	1	0.5	1	1	0.033

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

- *Cask Drop*—Similar to that analyzed for the electrometallurgical treatment process, spent nuclear fuel casks will be handled frequently when the sodium-bonded fuel is processed using the melt and dilute process. (Spent nuclear fuel handling at the ANL-W site is not limited to that associated with the treatment of the sodium-bonded spent nuclear fuel. The accident discussed here is intended to address only that portion of the handling activity that can be directly attributed to the treatment of sodium-bonded spent nuclear fuel.) The accident involves a dropped cask during loading or unloading, seal failure, and spent nuclear fuel cladding failure sufficient to release gaseous and volatile fission products to the atmosphere, and is the same as previously described for the cask drop accident for the electrometallurgical treatment process. Material at risk and release fraction values are provided in Table F–4. (See the accident description for more detail.)

- *Waste Handling Accident*—The filters used in the melt and dilute off-gas exhaust system must be periodically cleaned and the resultant liquid waste disposed. The decontamination of the filters is assumed to be performed after a number of batches is processed, i.e., here it was assumed that after processing of 600 kilograms (1,320 pounds) of heavy metal blanket fuel, or 160 kilograms (352 pounds) of heavy metal driver fuel, the filters will be decontaminated. It was postulated that a spill would occur during the transfer of the decontaminated liquid from one container to another. The event frequency is estimated at 0.024 events per year (DOE 1998). The material at risk is the fission products released during the melting process and collected on the filters. This includes the fission products with boiling points at or below 1400 °C (2,552 °F) and some metal oxides that can be expected to be formed during the heating process. (WSRC 1998b). A damage ratio of 0.5 was assumed to account for the spilling of half of the material during the accident. Airborne release fraction and respirable fraction values of 0.0002 and 0.5 respectively were chosen for the material based on the release of material from aqueous spills. (DOE 1994b) The spill is assumed to occur in an area not provided with a filtration system, and therefore, the leak path factor is 1.0. The material at risk, release fractions, and curies released for this accident for both EBR-II blanket and driver fuel are presented in **Table F–11**.

Table F–11 Melt and Dilute Process Material at Risk and Release Fraction Values for the Waste Handling Accident at ANL-W

Material At Risk							*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*	*DR[a]*	*ARF*	*RF*	*LPF*	*Blanket*	*Driver*
Sr-90	484.2	31520	0.5	0.0002	0.5	1	0.024	1.58
Sb-125	13.86	473.6	0.5	0.0002	0.5	1	0.00069	0.024
Te-125M	5.71	196.8	0.5	0.0002	0.5	1	0.00029	0.0098
I-129	0.00086	0.012	0.5	0.0002	0.5	1	4.320E-08	6.0×10^{-7}
Cs-134	8.04	281.6	0.5	0.0002	0.5	1	0.000402	0.014
Cs-137	1038	35360	0.5	0.0002	0.5	1	0.0519	1.77
Pu-238	5.63	26.6	0.000015	0.0002	0.5	1	8.4×10^{-9}	4.0×10^{-8}
Pu-239	451.8	43.0	0.000015	0.0002	0.5	1	6.8×10^{-7}	6.5×10^{-8}
Pu-240	31.08	1.5	0.000015	0.0002	0.5	1	4.7×10^{-8}	2.3×10^{-9}
Pu-241	126.0	3.6	0.000015	0.0002	0.5	1	1.9×10^{-7}	5.4×10^{-9}
Am-241	9.78	0.063	0.000015	0.0002	0.5	1	1.5×10^{-8}	9.5×10^{-11}
Am-242M	0.10	0.000016	0.000015	0.0002	0.5	1	1.5×10^{-10}	2.4×10^{-14}

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
[*] DR (damage ratio) for particulate that would not be condensed in the off-gas system includes a factor of 0.00003 to account for fraction oxidized and released from liquid metals and captured on the filters.

- *Aircraft crash*—The potential for an aircraft crash was evaluated for the Hot Fuel Examination Facility or Fuel Conditioning Facility as part of the evaluation of the electrometallurgical treatment process. The discussion provided previously is applicable to the use of the Hot Fuel Examination Facility in the melt and dilute process, see the discussion for the electrometallurgical treatment process earlier in this section. It was concluded that the likelihood of an aircraft crash causing damage to the facility process is not reasonably foreseeable, therefore, no specific analysis would be needed.

- *Fire*—The fire event selected for analysis is postulated to occur during the fuel cleaning process for the sodium-bonded spent nuclear fuel. The fire is the result of a breach in the Hot Fuel Examination Facility cell followed by a sodium fire. This event can occur as a result of the design-basis seismic event, which results in main cell breaches at piping and ventilation penetrations and results in a failure of the ventilation system. The frequency of this event is 0.008 per year.

It has been estimated that approximately 10 percent of the cesium in the spent nuclear fuel has migrated from the fuel region and bonded with the sodium being removed in the fuel cleaning process. Using the radionuclide inventories provided in Appendix D for the EBR-II driver and radial blanket elements to represent all driver and blanket fuel, this results in 670 curies of cesium-134 and 76,000 curies of cesium-137 entrained within the sodium. Assuming that as much as one half of the sodium is accumulated within the collection tank prior to processing to remove cesium from the sodium, the material at risk for the sodium fire would be 340 curies of cesium-134 and 38,000 curies of cesium-137. The release fractions selected for this accident are a damage ratio of 1.0, a combined airborne release fraction/respirable fraction of 2.5×10^{-4} and a leak path factor of 0.125. The airborne release fraction/respirable fraction is the same value as that used for cesium release from a metal fire in the design-basis seismic event analysis. The leak path factor is the value used for the Hot Fuel

Examination Facility during a design-basis seismic event. The quantity of cesium released (source term) as a result of this accident is 0.011 curies of cesium-134 and 1.2 curies of cesium-137.

- *Design Basis Seismic Event*—This is the same accident that was developed for the Design Basis Seismic Event for the electrometallurgical treatment of fuel in the Hot Fuel Examination Facility. The equipment availability and damage are assumed to be the same when the facility is used in the melt and dilute process as when it is used for the electrometallurgical treatment process. Consistent with the facility safety analysis report, the ventilation system was assumed to have failed, creating a leak path factor of 0.125. The frequency of this event is 0.008 per year (or once in 125 years).

The damage ratio, airborne release fraction, respirable fraction, and leak path factor are the same as for the electrometallurgical treatment process design-basis seismic event, with a few exceptions. Because the melt and dilute process at ANL-W operates at an elevated temperature of about 1,400 °C (2,552 °F), some fission products would boil off during the process and enter the off-gas control system. The airborne release fraction and respirable fraction for these volatilized fission product materials, e.g., strontium, antimony, cesium, tellurium, and iodine are set to 1.0 (DOE 1994b). In addition, even though some of these materials could have been condensed or not be in a vapor form (i.e., removed from the system) at the time of the accident, for conservatism it was assumed that all would be volatilized upon the initiation of the accident. The gaseous krypton and tritium are not considered here, since these are assumed to have been released to the environment during the fuel cleaning process. The source terms and release fractions are provided in **Table F–12**.

Table F–12 Material at Risk and Release Fraction Values for the Melt & Dilute Design Basis Seismic Event Accident at ANL-W

Material At Risk							*Source Term (curies)*	
Isotope	*Blanket (curies)*	*Driver (curies)*	*DR*	*ARF*	*RF*	*LPF*	*Blanket*	*Driver*
Sr-90	48.4	3152	1	1	1	0.125	6.05	394
Y-90	48.4	3152	1	4.0×10^{-6}	0.8	0.125	0.000019	0.0013
Ru-106	8.1	24.16	1	4.0×10^{-6}	0.8	0.125	3.2×10^{-6}	9.8×10^{-6}
Rh-106	8.1	24.16	1	4.0×10^{-6}	0.8	0.125	3.2×10^{-6}	9.8×10^{-6}
Cd-113M	0.043	0.74	1	4.0×10^{-6}	0.8	0.125	1.7×10^{-8}	3.0×10^{-7}
Sb-125	1.39	47.36	1	1	1	0.125	0.17	5.92
Te-125M	0.57	19.68	1	1	1	0.125	0.071	2.46
I-129	0.000086	0.0012	1	1	1	0.125	0.000011	0.00015
Cs-134	0.80	28.16	1	1	1	0.125	0.10	3.52
Cs-137	103.8	3536.0	1	1	1	0.125	12.98	442
Ba-137M	98.4	3344.0	1	4.0×10^{-6}	0.8	0.125	0.000039	0.0013
Ce-144	3.76	47.36	1	4.0×10^{-6}	0.8	0.125	1.5×10^{-6}	0.000019
Pr-144	3.76	47.36	1	4.0×10^{-6}	0.8	0.125	1.5×10^{-6}	0.000019
Pm-147	24.4	1321.6	1	4.0×10^{-6}	0.8	0.125	9.8×10^{-6}	0.00053
Sm-151	6.0	85.44	1	4.0×10^{-6}	0.8	0.125	2.4×10^{-6}	0.000034
Eu-154	0.44	9.07	1	4.0×10^{-6}	0.8	0.125	1.8×10^{-7}	3.6×10^{-6}
Eu-155	2.89	60.96	1	4.0×10^{-6}	0.8	0.125	1.2×10^{-6}	0.000024
Pu-238	0.56	2.66	1	4.0×10^{-6}	0.8	0.125	2.2×10^{-7}	1.1×10^{-6}
Pu-239	45.18	4.30	1	4.0×10^{-6}	0.8	0.125	0.000018	1.7×10^{-6}
Pu-240	3.11	0.15	1	4.0×10^{-6}	0.8	0.125	1.2×10^{-6}	6.0×10^{-8}
Pu-241	12.6	0.36	1	4.0×10^{-6}	0.8	0.125	5.0×10^{-6}	1.4×10^{-7}
Am-241	0.98	0.0063	1	4.0×10^{-6}	0.8	0.125	3.9×10^{-7}	2.5×10^{-9}
Am-242M	0.010	1.6×10^{-6}	1	4.0×10^{-6}	0.8	0.125	4.0×10^{-9}	6.4×10^{-13}

DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

F.2.2.1.3 Accident Scenarios Description and Source Terms at the Savannah River Site

❒ **Description of Accident Scenarios for the PUREX Process at SRS—**The following facilities would be used to store or process sodium-bonded fuel at SRS: F-Canyon, FB-Line, and the plutonium storage facility. The F-Canyon, FB-Line, and plutonium storage facility are part of the Building 221-F (or F-Canyon) structure. Shipments of the declad and cleaned sodium-bonded spent nuclear fuel cannot be received directly at the F-Canyon facility. The facility is not equipped to handle the transportation casks being used. The shipments would be received at the L-Reactor dissassembly basis, transferred to casks suitable for shipment to F-Canyon, and then moved to F-Canyon. The PUREX process can be used to separate the plutonium from the sodium-bonded blanket assemblies. In the PUREX process, the declad and cleaned blanket fuel would be dissolved in the F-Canyon dissolvers, and fission products would be separated from uranium and plutonium. The plutonium solution then would be pumped to the FB-Line for purification and solidification. The depleted uranium solution would be pumped to A-Line tanks for storage and future processing into depleted uranium oxides.

The accident scenarios, identified in **Table F–13** and defined in the following paragraphs, are applicable to the processing facilities as a whole (i.e., F-Canyon and FB-Line). Transfer and storage accidents were also considered for the analysis of F-Canyon related activities. The sodium-bonded spent nuclear fuel is declad and cleaned prior to shipment from ANL-W. This process results in the release of gases in the gap between the fuel and cladding (see Appendix E), the dominant radionuclides considered during the analysis of transfer (fuel and cask drop) accidents. Therefore, the accidents were not quantified. Accidents associated with the storage of the sodium-bonded spent nuclear fuel and the storage of the process products (plutonium and various waste forms) were assessed as having no additional impacts beyond those associated with the process-related accidents.

Table F–13 Selected Accident Scenarios for PUREX Process at SRS

Scenario	*Frequency (per year)*
Explosion: Ion exchange column	1×10^{-4}
Nuclear Criticality [a]	1×10^{-4}
Fire	6.1×10^{-5}
Earthquake (DBE)	1.3×10^{-4}
Aircraft Crash	less than 10^{-7}

DBE = design (evaluation) basis earthquake

[a] Only plutonium criticalities are evaluated. The potential for an americium criticality was considered but dismissed because of the limited americium mass and purity.

- *Explosion*—An explosion in an ion exchange column in the FB-Line is postulated to result from a strong exothermic reaction between nitric acid and the base resin in the cation (or anion) exchange column during plutonium solution exchange. This would result in a thermally induced pressure failure of the ion exchange vessel, and the resulting shrapnel would damage the product run tank and the product hold tank for this ion exchange pair. The explosion would breach the hot cell confinement. The plutonium in nitrite solution in the run and hold tanks would spill onto the cabinet floor and boil due to a subsequent resin fire. Based on the assumptions that the column was at its maximum load before the explosion, and the maximum quantity of liquid at the maximum allowable concentration was present, the estimated release of plutonium through the sand filter and the stack was calculated to be 0.241 grams of plutonium (DOE 1993b). No other source term is applicable to the FB-line accident. Processing in the F-Canyon removes all other fission products before the plutonium is processed in the FB-line. The frequency of such an event is estimated to be 1×10^{-4} per year (DOE 1993b).

- *Fire*—In the F-Canyon safety analysis report (WSRC 1994), a maximum fire was postulated to occur in the plutonium recycle process. The frequency of such a fire was estimated at 6.1×10^{-5} per year (WSRC 1994). The accident was assumed to burn the contents of the largest tank. The material at risk is 86,700 kilograms

(191,000 pounds) of solution. The combined [airborne release fraction][respirable fraction] was estimated to be 1×10^{-2} (DOE 1994b). The airborne materials would pass through sand filter, with a leak path factor of 0.005, before entering the atmosphere. The maximum recycle fire in the F-Canyon would result in the bounding source term (**Table F–14** gives the source term). Fire in the FB-Line would result in consequences that are several times lower than those from the F-Canyon fire.

Table F–14 Maximum Fire Source Term

Isotope	*Source Term (curies)*
Sr-90	1.5
Ru-106	12
Ce-144	36
U-234	3.0×10^{-7}
U-235	4.8×10^{-6}
U-236	4.9×10^{-6}
U-238	0.00044
Pu-238	0.19
Pu-239	1.6
Pu-240	0.36
Pu-241	4.2
Pu-242	0.000053
Am-241	0.32

- *Criticality*—A plutonium solution criticality was postulated. The criticality was assumed to consist of an initial burst of 1×10^{18} fissions in 0.5 seconds, followed at 10-minute intervals for the next 8 hours by bursts of 2×10^{17} fissions, for a total of 1×10^{19} fissions as specified in the U.S. Nuclear Regulatory Commission's Regulatory Guide 3.35 (NRC 1979) and NUREG-1320 (NRC 1988) and in the DOE-HDBK-3010-YR (DOE 1994b) report. The 10^{19} fission yield was based on the assumptions that the solution criticality occurred in a tank with a minimum volume of 3,785 liters (1,000 gallons) and that approximately 100 liters (26 gallons) of this volume evaporated due to heat released during the fission process. Based on the data provided in the DOE Safety Survey Report (DOE 1993c), a 10^{19} criticality event in the FB-Line process would result in the bounding source term (**Table F–15** gives the source terms). The frequency of such an event was estimated to be 1×10^{-4} per year.

Table F–15 Criticality Source Term for 10^{19} Fissions in Plutonium Solution

Isotope	*Radioactivity (curies)* [a]			*ARF* [b]	*LPF* [c]	*Source Term (curies)*
	0-30 minutes	*30 min-8 hours*	*Total*			
Kr-83m	15	95	110	1	1	110
Kr-85m	9.9	61	70.9	1	1	70.9
Kr -85	0.00012	0.00072	0.00084	1	1	0.00084
Kr-87	60	370	430	1	1	430
Kr-88	32	200	232	1	1	232
Kr-89	1,800	11,000	12,800	1	1	12,800
Xe-131m	0.014	0.086	0.1	1	1	0.1
Xe-133m	0.31	1.9	2.21	1	1	2.21
Xe-133	3.8	23	26.8	1	1	26.8
Xe-135m	460	2,800	3,260	1	1	3,260
Xe-135	57	350	407	1	1	407
Xe-137	6,900	42,000	48,900	1	1	48,900
Xe-138	1,500	9,500	11,000	1	1	11,000
I-131	1.5	9.5	11	0.25	1	2.75

Isotope	Radioactivity (curies) [a]			ARF [b]	LPF [c]	Source Term (curies)
	0-30 minutes	30 min-8 hours	Total			
I-132	170	1,000	1,170	0.25	1	293
I-133	22	140	162	0.25	1	40.5
I-134	600	3,700	4,300	0.25	1	1,080
I-135	63	390	453	0.25	1	113
Pu-238 [c, d]			3.6	0.0005	0.005	0
Pu-239 [c, d]			170	0.0005	0.005	0.00043
Pu-240 [c, d]			39	0.0005	0.005	0.0001
Pu-241 [c, d]			2,400	0.0005	0.005	0.006
Pu-242 [c, d]			0.003	0.0005	0.005	7.50×10^{-9}

ARF = airborne release fraction, LPF = leak path factor
[a] Regulatory Guide 3.35 (NRC 1979).
[b] Airborne release fractions are equal to 1.0 for noble gases, 0.25 for iodine, and 0.0005 for plutonium; all particles are assumed to be in the respirable range (i.e., Respirable Fraction = 1.0).
[c] Plutonium in 100 liters of solution.
[d] This plutonium is assumed to be released to the atmosphere through a high-efficiency particulate air filter (e.g., SRS's sand filter) with a 0.995 efficiency. The plutonium values are the maximum solution concentration in the FB-Line (DOE 1993b).

- *Earthquake*—Recent analyses of earthquake hazards at F-Canyon indicate that a 0.24-g peak ground acceleration level earthquake—with a return period of 8,000 years (or a frequency of 1.25×10^{-4} per year) for the F-Canyon facility—could damage the structure and cause localized interior failures as well as interior and exterior wall cracks (DOE 1996b). Previous analyses of earthquake hazards at F-Canyon estimated the consequences of such a magnitude earthquake with a higher frequency of occurrences—2×10^{-4} per year (DOE 1995b and WSRC 1994). Using the assumptions in the F-Canyon Facility Safety Analysis Report (WSRC 1994), a bounding source term was developed for an earthquake accident (**Table F–16** gives the F-Canyon source term). Given an earthquake, it was assumed that the plutonium contents in all the processes (F-Canyon and FB-Line) would be spilled on the canyon floor. It was further assumed that the airborne material would enter the environment through the building cracks, which are formed by the loss of sealant between the sections because of differential motion of the section, with a penetration leak path factor of 0.10. For the FB-Line, the material at risk is assumed to be 2,000 grams (4.4 pounds) of plutonium in a molten metal form and 2,000 grams (4.4 pounds) of plutonium in a liquid form. The airborne release fraction, times the respirable fraction is 0.0022 for the molten metal form, and 0.000047 for liquid form, including both the initial and resuspended airborne release fraction multiplied by respirable fraction values. This results in an FB-Line earthquake source term of 0.45 grams of plutonium released to the environment.

Table F–16 Maximum Earthquake Source Term

Isotope	Source Term (curies)	Isotope	Source Term (curies)
Sr-90	0.156	Pu-239	0.168
Ru-106	127	Pu-240	0.0387
Ce-144	3.72	Pu-241	0.429
Cs-137	0.00574	Pu-242	7.18×10^{-6}
Eu-154	0.0338	Am-241	0.0151
Np-237	5.84×10^{-8}	Am-242m	0.0000630
Np-239	0.0116	Am-243	0.00616
U-234	4.09×10^{-7}	Cm-244	0.668
U-235	5.07×10^{-7}	Cm-245	0.0000542
U-236	5.12×10^{-7}	Cm-246	0.0000844
U-238	0.0000457	Cm-247	4.10×10^{-10}
Pu-238	0.0276		

- *Aircraft Crash*—The location of the F-Canyon facility is far away from any airport; therefore, no takeoff and landing crash accidents need to be considered. The crashes that could occur during in-flight would need to be considered. According to the DOE Standard on aircraft crash analysis, DOE-STD-3014-96 (DOE 1996c), the expected crash frequency for the site is approximately 2×10^{-4} per square-mile per year from general aviation, 6×10^{-7} and 2×10^{-6} per square-mile per year from air carrier and air taxies, respectively, and 1×10^{-7} and 6×10^{-7} per square-mile per year from large military and small military aircraft, respectively. Using the building dimensions and the data provided in the DOE Standard for aircraft crash analysis, an upper bound frequency for an aircraft crash into the canyon buildings was estimated to be 4.6×10^{-6} and 1.5×10^{-7} per year for general aviation and commuter (air taxi) aircraft, respectively. These values were calculated without considering any site-specific effects (e.g., the topography and building structures around the facility). Considering the available skid distance of 60 meters (200 feet) that an aircraft could skid before hitting the building, the frequency of an air taxi crashing into the building would be less than 10^{-8} per year. When only crashes that directly hit the structure were considered, general aviation aircraft would have the only estimated crash frequency greater than 10^{-7} per year. The F-Canyon building is a maximum resistant construction structure designed to withstand a pressure of 47.9 kilopascal (1,000 pounds per square foot). Therefore, crashes of small aircraft (helicopter or a small observation/ security aircraft) into these buildings are not expected to damage the buildings. If a general aviation aircraft were to crash into the buildings, its consequences (both in magnitude and frequency) would be smaller than that hypothesized for a design-basis evaluation earthquake.

❒ **Description of Accident Scenarios for the Melt and Dilute at SRS—**The following accidents have been considered for the melt and dilute option, when performed at the Building 105-L (after receipt of the declad and cleaned spent nuclear fuel at the L-Reactor Disassembly Basin) as proposed in the SRS Spent Nuclear Fuel Management Environmental Impact Statement (DOE 1998). In this process, the declad and cleaned blanket fuel along with aluminum metal would be heated to approximately 1,000 °C (1,832 °F) to form an alloy of 30 percent uranium and 70 percent aluminum and cast as ingots. The heating process would remove some of the radionuclides found in the spent nuclear fuel. The analysis assumes a batch size of 60 kilograms (132 pounds) of heavy metal, which is the batch size limit for this process when performed in Building 105-L. The radionuclide content of an EBR-II radial blanket fuel has been conservatively used to represent the radionuclide content of all blanket fuels. The accident scenarios identified in **Table F–17**, and described in the following paragraphs, are applicable to the melt and dilute processing of the blanket fuel in SRS Building 105-L. Accidents associated with the onsite transfer and storage of the declad and cleaned spent nuclear fuel were considered for analysis. As in the accident analysis for the PUREX process, these accidents were not quantified. Accidents associated with the transfer and storage of the spent nuclear fuel and diluted waste forms were assessed as having no additional impacts beyond those analyzed for process-related accidents.

Table F–17 Selected Accident Scenarios Melt and Dilute at SRS Building 105-L

Scenario	*Frequency (per year)*
Loss of Cooling Water	0.05
Waste Handling Spill	0.024
Loss of Electric Power	0.006
Fire	0.075
Design Basis Earthquake	NA*

* Building 105-L and the melt and dilute components are expected to remain functioning after a design-basis earthquake. The most significant impact of this event is a potential loss of offsite power. The consequences of an earthquake up to a design-basis event are thereby bounded by the loss of power event. The loss of power event has a higher frequency than the design-basis earthquake and is used in place of the design-basis earthquake.

- *Loss of Cooling Water*—The postulated melter explosion event results from a buildup or addition of impurities to the metal melt. Impurities range from water (causing a steam explosion) to chemical contaminants (possible high temperature exothermic reactions). As a result of the reaction in the metal melt, molten material is ejected from the melter into the processing structure. Cooling water pipes within the process area could be ruptured as a result of contact with the ejected material. Should this occur, the water released would be converted to steam and would be expected to overwhelm the exhaust fans resulting in the failure of the exhaust system and an unfiltered release. The frequency of this event has been estimated to be bound by a value of 0.05 per year. (WSRC 1998a).

The material at risk is conservatively estimated to be the full radionuclide content of one melt batch. The metal melt explosion is assumed to affect approximately one-half of the material in the melter, resulting in a damage ratio of 0.5 for all material, except for the volatile gaseous fission products at 1,000 °C (1,832 °F) which are assigned a damage ratio of 1.0. The airborne release fraction and respirable fraction values of 0.001 and 0.1 are based on the conservative assumption that the characteristics of the released material will be bounded by the characteristics of material released in an explosion involving powder material (DOE 1994b). With the failure of the off-gas system, a leak path factor of 0.1 is assumed for all materials. The material at risk and release fractions are summarized in **Table F–18**.

Table F–18 Melt and Dilute Material At Risk and Release Fractions for the Loss of Cooling Water Integrity at Building 105-L

Isotope	*MAR (curies)*	*DR*	*ARF* [a]	*RF*	*LPF*	*Source Term*
Fe-55	5.41	0.5	0.001	0.1	0.1	2.71e-05
Ni-63	0.184	0.5	0.001	0.1	0.1	9.20e-07
Sr-90	48.4	0.5	0.001	0.1	0.1	2.42e-04
Y-90	48.4	0.5	0.001	0.1	0.1	2.42e-04
Ru-106	8.1	0.5	0.001	0.1	0.1	4.05e-05
Rh-106	8.1	0.5	0.001	0.1	0.1	4.05e-05
Cd-113M	0.0427	0.5	0.001	0.1	0.1	2.14e-07
Sb-125	1.39	0.5	0.001	0.1	0.1	6.95e-06
Te-125M	0.571	1	1	1	0.1	0.0571
I-129	0.000086	1	1	1	0.1	8.64×10^{-6}
Cs-134	0.804	1	1	1	0.1	0.0804
Cs-137	104	1	1	1	0.1	10.4
Ba-137M	98.4	0.5	0.001	0.1	0.1	4.92e-04
Ce-144	3.76	0.5	0.001	0.1	0.1	1.88e-05
Pr-144	3.76	0.5	0.001	0.1	0.1	1.88e-05
Pm-147	24.4	0.5	0.001	0.1	0.1	1.22e-06
Sm-151	6	0.5	0.001	0.1	0.1	3.00e-05
Eu-154	0.44	0.5	0.001	0.1	0.1	2.20e-06
Eu-155	2.89	0.5	0.001	0.1	0.1	1.45e-05
Th-22	9.30e-06	0.5	0.001	0.1	0.1	4.65e-11
U-234	0.00008	0.5	0.001	0.1	0.1	3.99e-10
U-235	0.000226	0.5	0.001	0.1	0.1	1.13e-09
U-236	0.000254	0.5	0.001	0.1	0.1	1.27e-09
U-238	0.0196	0.5	0.001	0.1	0.1	9.80e-08
Np-237	7.80e-06	0.5	0.001	0.1	0.1	3.90e-11
Pu-238	0.563	0.5	0.001	0.1	0.1	2.82e-06
Pu-239	45.2	0.5	0.001	0.1	0.1	2.26e-04
Pu-240	3.11	0.5	0.001	0.1	0.1	1.56e-05
Pu-241	12.6	0.5	0.001	0.1	0.1	6.30e-05
Am-241	0.978	0.5	0.001	0.1	0.1	4.89e-06
Am-242M	0.0101	0.5	0.001	0.1	0.1	5.05e-08

MAR = material at risk, DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

[a] The airborne release fraction values provide here are conservatively based on the assumption that reaction would behave as explosive powder mix.

- *Waste Handling Accident*—The filters used in the melt and dilute off-gas exhaust system must be periodically cleaned and the resultant liquid waste disposed. The decontamination of the filters is assumed to be performed after a number of batches is processed, i.e., here it was assumed that after processing 600 kilograms, 10 batches, of heavy metal blanket fuel, the filters will be decontaminated. It was postulated that during the transfer of the decontaminate liquid from one container to another, a spill would occur. The event frequency is estimated at 0.024 per year (DOE 1998). The material at risk is the fission products released during the melting process and collected on the filters. This includes the fission products with boiling points at or below 1,000 °C (1,832 °F) and some metal oxides that can be expected to be formed during the heating process. (WSRC 1998b). A damage ratio of 0.5 was assumed to account for the spilling of half of the material during the accident. Airborne release fraction and respirable fraction values of 0.0002 and 0.5 respectively were chosen for the material based on the release of material from aqueous spills (DOE 1994b). The spill is assumed to occur in an area not provided with a filtration system, and therefore, the leak path factor is 1.0. These material at risk, release fractions, and curies released for this accident for both EBR-II blanket and driver fuel are presented in **Table F–19**.

Table F–19 Melt and Dilute Material At Risk and Release Fractions for the Waste Handling Accident at Building 105-L

Isotope	*MAR*	*DR**	*ARF*	*RF*	*LPF*	*Source Term*
Te-125M	5.71	0.5	0.0002	0.5	1	0.000286
I-129	0.000864	0.5	0.0002	0.5	1	4.32×10^{-8}
Cs-134	8.04	0.5	0.0002	0.5	1	0.000402
Cs-137	1040	0.5	0.0002	0.5	1	0.052
Pu-238	0.563	0.000015	0.0002	0.5	1	8.45×10^{-9}
Pu-239	45.2	0.000015	0.0002	0.5	1	6.78×10^{-7}
Pu-240	31.1	0.000015	0.0002	0.5	1	4.67×10^{-8}
Pu-241	126	0.000015	0.0002	0.5	1	1.89×10^{-7}
Am-241	9.78	0.000015	0.0002	0.5	1	1.47×10^{-8}
Am-242M	0.101	0.000015	0.0002	0.5	1	1.52×10^{-10}

MAR = material at risk, DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).
* Damage ratios for neptunium, plutonium, and americinium include an airborne release fraction value of 0.00003 to account for fraction released from liquid metals and captured on the filters.

- *Loss of Offsite Power*—The loss of offsite power, with the subsequent failure of the onsite power supply, will result in the failure of the off-gas system, and a potential unfiltered release path to the environment. The probability of this combination of events is conservatively estimated at 0.006 per year (WSRC 1998a). The material at risk is assumed to be the volatile radionuclide inventory of one processing batch of material (approximately 60 kilograms [132 pounds] of heavy metal). Additionally, some amounts of radioactive metallic and metallic oxide dusts could be generated and released during a loss of power event. The airborne release fraction/respirable fraction values for the gaseous fission products are assumed to be 1.0, while the metallic dust release fractions at elevated temperatures are an airborne release fraction of 0.00003 and respirable fraction of 0.04 (DOE 1994b). A leak path factor of 0.5 has been used for all material to account for possible plate out during migration of material out of the processing area. The material at risk and release fraction data are summarized in **Table F–20**.

Table F–20 Melt and Dilute Material At Risk and Release Fractions for the Loss of Power at Building 105-L

Isotope	*MAR*	*DR*	*ARF*	*RF*	*LPF*	*Source Term*
Te-125M	0.571	1	1	1	0.5	2.86e-01
I-129	0.000086	1	1	1	0.5	4.32e-05
Cs-134	0.804	1	1	1	0.5	4.02e-01
Cs-137	104	1	1	1	0.5	5.20e+01
Pu-238	0.563	1	0.00003	0.04	0.5	3.38e-07
Pu-239	45.2	1	0.00003	0.04	0.5	2.71e-05
Pu-240	3.11	1	0.00003	0.04	0.5	1.87e-06
Pu-241	12.6	1	0.00003	0.04	0.5	7.56e-06
Am-241	0.978	1	0.00003	0.04	0.5	5.87e-07
Am-242M	0.0101	1	0.00003	0.04	0.5	6.06e-09

MAR = material at risk, DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

- *Area Fire*–Fires in Building 105-L have the potential to release material from several different sources. Fires have the potential to release material from the chemical decontaminate solution and the off-gas filters and baffles, and have the potential to affect the ventilation and filtration system resulting in the release modeled for the loss of power event. The fire selected for analysis is the fire that results in the failure of the waste container and releases some of the decontaminate solution. This fire has the potential to release more material than a fire that impacts the off-gas filters and baffles. The frequency of a fire in Building 105-L, based on site-wide fire data for SRS, is 0.075 fires per year. This frequency has been conservatively used as the frequency of a fire that impacts the chemical decontaminate solution. The material at risk is the same as for the waste handling accident, the volatile gases, and metallic and metallic oxide dust that is the result of the processing of 10 batches of material in the melter. All material in the waste container is at risk and the damage ratio is assumed to be 1.0. Boiling of a shallow pool of aqueous solution results in bounding airborne release fraction and respirable fraction values of 0.002 and 1 respectively (DOE 1994b). No credit is taken for any reduction due to leak path factor, i.e., a leak path factor of 1.0 is used. **Table F–21** summarizes the material at risk and release fractions for this accident scenario.

Table F–21 Melt and Dilute Material At Risk and Release Fractions for the Area Fire at Building 105-L

Isotope	*MAR*	*DR*	*ARF*	*RF*	*LPF*	*Source Term*
Te-125M	5.71	1	0.002	1	1	0.0114
I-129	0.0006	1	0.002	1	1	1.73×10^{-6}
Cs-134	8.04	1	0.002	1	1	0.0161
Cs-137	1040	1	0.002	1	1	2.08
Np-237	7.80×10^{-5}	0.00003	0.002	1	1	4.68×10^{-12}
Pu-238	5.63	0.00003	0.002	1	1	3.38×10^{-7}
Pu-239	452	0.00003	0.002	1	1	0.0000271
Pu-240	31.1	0.00003	0.002	1	1	1.87×10^{-6}
Pu-241	126	0.00003	0.002	1	1	7.56×10^{-6}
Am-241	9.78	0.00003	0.002	1	1	5.87×10^{-7}
Am-242M	0.101	0.00003	0.002	1	1	6.06×10^{-9}

MAR = material at risk, DR = damage ratio, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

F.2.2.2 Consequences and Risk Calculations

Once the source term for each accident scenario is determined, the radiological consequences are calculated. The calculations vary depending on how the release is dispersed, what material is involved, and which receptor is being considered. Risks are calculated based on the accident's frequency and its consequences. The risks are also stated in terms of additional cancer fatalities resulting from a release using a conversion factor of 5×10^{-4} latent cancer fatalities per person-rem for the members of the public, and 4×10^{-4} latent cancer fatalities per person-rem for workers.

Radiological consequences to four different receptors are evaluated: a maximally exposed offsite individual (an individual member of the public), general population, noninvolved worker (or a co-located worker), and facility worker. The consequences to the facility workers are qualitatively evaluated. For the other receptors, quantitative estimates of consequences are made; two types of dispersion conditions are considered—95th percentile and 50th percentile meteorological conditions. The 50th percentile condition represents the median meteorological condition and is defined as that for which more severe conditions occur 50 percent of the time. The 95th percentile condition represents relatively low probability meteorological conditions that produce higher calculated exposures; it is defined as that condition not exceeded more than 5 percent of the time. Both dispersion conditions are modeled using the GENII program, which determines the desired condition from the site-specific meteorological data in the form of a joint frequency distribution. Joint frequency data are usually produced from at least three consecutive years of site weather data in terms of percentage of time that the wind blows in specific directions (e.g., south, south-southwest, southwest) for the given midpoint (or average) wind speed class and atmospheric stability.

Radiological consequences to a receptor from an accident in the FB-line are estimated based on a calculated 50-year committed dose factor (dose factor), resulting from releases of 1 gram of plutonium with an isotopic distribution associated with the EBR-II blanket fuel (**Table F–22**). This is done because the FB-line only processes plutonium already separated in the F-Canyon.

The values given in this table represent the maximum dose to the receptor and are obtained using the GENII program.

Table F–22 Receptors' Dose Factors for Accidental Releases of 1 gram Plutonium from Accident Initiated in FB-Line

Receptor		***95 Percent Meteorological Condition***	***50 Percent Meteorological Condition***
Maximally Exposed Individual (rem)	elevated release	0.027	Not Applicable
	ground release	0.13	Not Applicable
Population (person-rem)	elevated release	1500	220
	ground release	5000	270
Worker (rem)	elevated release	Not Applicable	0.080
	ground release	Not Applicable	2.0

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 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).

The consequences to the involved worker, presented in **Tables F-23** and **F–24**, are accident dependent and site-specific. In facilities where the involved worker activities include remote operations, the consequences of accidents would be lower than in facilities where the workers are near the process. The following paragraphs summarize the various potential consequences to the involved workers from the hypothesized accidents at different sites. Additionally, a limited number of fatalities could occur in an indirect or secondary manner—for example, the involved worker could be killed by an earthquake or explosion.

Table F–23 Involved Worker Consequences from Various Hypothesized Accidents

Accident	*Consequences*
Explosion (Ion Exchange)	Could potentially result in fatal injuries (nonradiological) to the nearby involved workers. (SRS only)
Criticality	Could potentially result in fatal dose to the nearby involved workers. (Worker location outside cells [e.g., outside argon cell at ANL-W] provides worker protection)
Fire	No fatality is expected, some nearby workers could inhale the dispersed radioactive materials before using respirator and leaving the area.
Earthquake	No fatality is expected.
Spill	Nearby workers could inhale the dispersed radioactive materials before using respirator and leaving the area.

Table F–24 Involved Worker Summary

Accident Description		
SRS—PUREX Process	*F-Canyon and F-B Line*	*ANL-W*
Earthquake	47	50
Explosion, Ion Exchange Column	16	Not Applicable
Nuclear Criticality	16	15
Fire	16	4

- ❒ **Explosion**—An explosion could result in serious, even fatal, injuries to involved workers from the accident itself (at SRS). Some of the involved workers could inhale the dispersed radioactive material before using their respirators and evacuating the area. No fatality is expected from the radiological consequences.

- ❒ **Fire**—Involved workers could inhale some radioactive material, before evacuating the area. No fatality is expected from the radiological consequences.

- ❒ **Spill**—Depending on the location of the spill, nearby workers may inhale the airborne radioactive materials before evacuating the area. Involved workers normally would be wearing respirators when handling the radioactive material containers. No fatality is expected to result from such an accident.

- ❒ **Earthquake**—Involved workers could receive lethal injuries from the accident itself. No fatality is expected from radiological consequences.

- ❒ **Aircraft Crash**—Consequences similar to those of an earthquake may result from the accident.

- ❒ **Criticality—**Involved workers could receive substantial, or potentially fatal, doses from prompt neutrons and gamma rays emitted from the first pulse. After the initial pulse, the workers would evacuate the area immediately on the initiation of the criticality monitoring alarms.

Analysis Conservatism and Uncertainty

To assist in evaluating the impact of the processing options at SRS and ANL-W on a common basis, a spectrum of generic accidents were postulated for each process location. The accident scenarios were based on similar accidents documented in various site documents. When required, accident assumptions were modified to enable comparison between the sites. In cases where similar accidents were evaluated in site specific documents, the more conservative analysis assumptions were used for all sites to normalize the results for the purpose of comparison. The following accident analysis parameters have a major impact on accident consequence estimates (i.e., dose to the public and worker): weather conditions existing at the time of the accident, material at risk, isotopic breakdown of the material at risk, and source term released to the environment.

Weather conditions assumed at the time of the accident have a large impact on dose estimates. Accident impacts to the public (both the maximally exposed individual and the population) presented in this appendix were estimated using both 95th percentile and median 50th percentile weather conditions. The impacts presented in the body of the EIS are based on the 50th percentile weather conditions for population dose (NRC 1976), and 95th percentile weather conditions (NRC 1982) for the maximally exposed individual dose (which provides conservative maximally exposed individual dose estimates). The GENII computer code was used to calculate doses to the public within 80 kilometers (50 miles) of the accident release point. The code calculates the public dose in each of 16 sectors centered at the accident release point. The GENII computer code also assumes that total source term is released into each sector and that there is no change in the weather (i.e., wind direction, wind speed, and stability class) while the accident plume is traversing the 80-kilometer sector. The use of the 95th percentile weather data rather than the expected or median 50th percentile weather data, was considered to be unrealistic for estimating population dose. Meteorological conditions used in the analysis are based on measured weather data at the site. The 95th percentile represents a very stable site meteorological condition, which cannot be expected to be applicable for a wide area up to 80 kilometers from the site. Therefore, the 50th percentile, which represent a more neutral weather condition, is more representative of expected weather conditions over a wide area.

Uncertainties in accident frequencies do not impact the accident consequences, but do impact accident risk. The site/facility specific accident frequencies (i.e., earthquake induced building damage and aircraft crash) were based on data provided by the sites. Process specific accident frequencies were estimated based on analyses provided in site specific documentation. In cases where similar accidents were evaluated in site specific documents, the more conservative accident frequency was used for all sites to normalize the results for the purpose of comparison.

Due to the layers of conservatism built into the accident analysis for the spectrum of postulated accidents, the estimated consequences and risks to the public represent the upper limit for the individual classes of accidents. The uncertainties associated with the accident frequency estimates are enveloped by the analysis conservatism.

F.2.3 Accident Analyses Consequences and Risk Results

F.2.3.1 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) except for stabilization activities that may be necessary for continued safe and secure storage indefinitely or until a new treatment technology is developed. Under the electrometallurgical demonstration project, approximately 0.6 metric tons of heavy metal of EBR-II driver fuel and 1 metric ton of heavy metal blanket fuel would be processed. This EIS evaluates the impacts associated with activities required to clean up and stabilize any residual waste materials generated during the Electrometallurgical Treatment Demonstration Project at ANL-W. In addition, at the completion of the project, any remaining sodium-bonded spent nuclear fuel in the process 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 equipment installation. Some of the spent nuclear fuel handling and processing accidents identified under Alternative 1 are applicable to the No Action Alternative. **Tables F-25** and **F-26** provide the dose calculation results for the design-basis and beyond-design-basis seismic events for stabilizing the residual waste. The data for the remaining accidents considered for the No Action Alternative (the salt powder spill in the Hot Fuel Examination Facility, the cask drop, and transuranic waste fire) are provided in the discussion of Alternative 1 "electrometallurgical treatment at ANL-W." Data is provided for consequences and risks to the maximally exposed offsite individual, a noninvolved worker, and the general population. The accident assumptions and parameters used in developing this information have been provided in Section 2.2 of this appendix. EBR-II driver fuel characteristics (radionuclide compositions), which bound the consequences, were used to represent the consequences and risks during stabilization of wastes for the demonstration project for the No Action Alternative. The transuranic waste fire accident was analyzed using a generic transuranic waste package composition, rather than the driver-specific composition.

Table F–25 Summary of Dose Calculation Results for Design-Basis Seismic Events (Driver)

			95th-percentile meteorology			*50th-percentile meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (millirem)***	***Worker (millirem)***	***Population (person-rem)***	***Average individual (mrem)***
Design-Basis Events	0.008	Dose per event	11.9	52.4	0.631	0.638	4.66	1.38	00166
		Dose per year	0.0012	0.0052	6.31×10^{-5}	6.38×10^{-5}	4.66×10^{-4}	1.4×10^{-4}	1.66×10^{-6}
		LCF	6.0×10^{-10}	2.6×10^{-5}	3.2×10^{-11}	3.2×10^{-11}	1.9×10^{-10}	6.9×10^{-8}	8.3×10^{-13}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–26 Summary of Dose Calculation Results for Beyond-Design-Basis Seismic Events (Driver)

			95th-percentile meteorology			*50th-percentile meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (millirem)***	***Worker (millirem)***	***Population (person-rem)***	***Average Individual (millirem)***
Beyond-Design-Basis Events[a]	0.00001	Dose per event	95.6	41.9	5.05	5.11	37.3	11	0.133
		Dose per year	9.6×10^{-4}	4.2×10^{-4}	5.1×10^{-5}	5.1×10^{-5}	3.7×10^{-4}	1.1×10^{-4}	1.3×10^{-6}
		LCF	4.8×10^{-10}	2.1×10^{-7}	2.6×10^{-11}	2.6×10^{-11}	1.5×10^{-10}	5.5×10^{-8}	6.5×10^{-13}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

[a] During stabilization of the demonstration project waste, only the Hot Fuel Examination Facility salt powder spill would be applicable.

F.2.3.2 Alternative 1 - Electrometallurgical Treatment at ANL-W

The processing technology considered for this alternative consists solely of the electrometallurgical treatment processing of the sodium-bonded spent nuclear fuel at ANL-W, using the Fuel Conditioning Facility and Hot Fuel Examination Facility. **Tables F–27** through **F–37** provide the dose calculation results for the electrometallurgical treatment related accidents at the ANL-W facility. Data is provided for consequences and risks to the maximally exposed individual, an uninvolved worker and the general population. The accident assumptions and parameters used in developing this information has been provided in Section 2.2 of this appendix. EBR-II driver fuel and EBR-II blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively. The transuranic waste fire accident was analyzed using a generic transuranic waste package composition, rather than either a blanket or driver specific composition.

Table F–27 Summary of Dose Calculation Results for Salt Powder Spill (Driver)

			95th-Percentile Meteorology			50th-Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (millirem)	Population (person-rem)	Average Individual (millirem)	MEI (millirem)	Worker (millirem)	Population (person-rem)	Average Individual (millirem)
Hot Fuel Examination Facility Salt Powder Spill	0.01	Dose per event	4.6E-04	2.6E-03	3.1E-05	4.6E-05	4.7E-07	9.8E-05	1.2E-06
		Dose per year	4.6E-06	2.6E-03	3.1E-07	4.6E-07	4.7E-09	9.8E-07	1.2E-08
		LCF	2.3E-12	1.3E-08	1.6E-13	2.3E-13	1.9E-15	4.9E-10	5.9E-15

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–28 Summary of Dose Calculation Results for Salt Powder Spill (Blanket)

			95th-Percentile Meteorology			50th-Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (mrem)	Population (person-rem)	Average Individual (mrem)	MEI (mrem)	Worker (mrem)	Population (person-rem)	Average Individual (mrem)
Hot Fuel Examination Facility Salt Powder spill	0.01	Dose per event	1.2E-04	7.1E-04	8.5E-06	1.2E-05	1.1E-06	2.7E-05	3.2E-07
		Dose per year	1.2E-06	7.1E-06	8.5E-08	1.2E-07	1.1E-08	2.7E-07	3.2E-09
		LCF	6.2E-13	3.5E-09	4.3E-14	6.2E-14	4.4E-15	1.3E-10	1.6E-15

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–29 Summary of Dose Calculation Results for Cask Drop (Driver)

			95th-Percentile Meteorology			50th-Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (milli-rem)	Population (person-rem)	Average Individual (millirem)	MEI (milli-rem)	Worker (milli-rem)	Population (person-rem)	Average Individual (millirem)
Cask drop	0.01	Dose per event	3.0E-02	1.4E-01	1.7E-03	1.6E-03	8.4E-04	3.5E-03	4.2E-05
		Dose per year	3.0E-04	1.4E-03	1.7E-05	1.6E-05	8.4E-06	3.5E-05	4.2E-07
		LCF	1.5E-10	6.9E-07	8.3E-12	8.2E-12	3.4E-12	1.7E-08	2.1E-13

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–30 Summary of Dose Calculation Results for Cask Drop (Blanket)

			95th-Percentile Meteorology			50th-Percentile Meteorology			
Accident	Frequency (event per year)	Risk	MEI (milli-rem)	Population (person-rem)	Average Individual (millirem)	MEI (milli-rem)	Worker (milli-rem)	Population (person-rem)	Average Individual (millirem)
Cask Drop	0.01	Dose per event	2.4E-03	1.1E-02	1.3E-04	1.3E-04	4.9E-05	2.8E-04	3.4E-06
		Dose per year	2.4E-05	1.1E-04	1.3E-06	1.3E-06	4.9E-07	2.8E-06	3.4E-08
		LCF	1.2E-11	5.6E-08	6.7E-13	6.6E-13	2.0E-13	1.4E-09	1.7E-14

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–31 Summary of Dose Calculation Results for Single Container Transuranic Waste Fire

			95th-Percentile Meteorology			50th-Percentile Meteorology			
Accident	*Frequency (event per year)*	*Risk*	*MEI (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (milli-rem)*	*Worker (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*
Transuranic Waste Fire	0.001	Dose per event	5.9E-02	2.7E-01	3.3E-03	3.2E-03	2.2E-01	7.1E-03	8.5E-05
		Dose per year	5.9E-05	2.7E-04	3.3E-06	3.2E-06	2.2E-04	7.1E-06	8.5E-08
		LCF	3.0E-11	1.4E-07	1.6E-12	1.6E-12	8.7E-11	3.5E-09	4.3E-14

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–32 Summary of Dose Calculation Results for Design-Basis Seismic Event (Driver)

			95th-Percentile Meteorology			50th-Percentile Meteorology			
Accident	*Frequency (event per year)*	*Risk*	*MEI (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (milli-rem)*	*Worker (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*
Design-Basis Seismic Event	0.0002 (Multi-facility event)	Dose per event	1.3E+01	7.0E+01	8.4E-01	9.5E-01	4.7E+00	2.8E+00	3.4E-02
		Dose per year	2.6E-03	1.4E-02	1.7E-04	1.9E-04	8.4E-04	5.6E-04	6.8E-06
		LCF	1.3E-09	7.0E-06	8.4E-11	9.5E-11	3.8E-10	2.8E-07	3.4E-12
	0.008 (HFEF)	Dose per event	12	52	0.63	0.64	4.7	1.4	0.017
		Dose per year	0.095	0.42	0.0050	0.0051	0.037	0.011	0.00013
		LCF	4.8×10^{-8}	0.00021	2.5×10^{-9}	2.6×10^{-9}	1.5×10^{-8}	5.5×10^{-6}	6.6×10^{-11}

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–33 Summary of Dose Calculation Results for Design-Basis Seismic Event (Blanket)

			95th-Percentile Meteorology			50th-Percentile Meteorology			
Accident	*Frequency (event per year)*	*Risk*	*MEI (milli-rem)*	*Popula-tion (person-rem)*	*Average Individual (millirem)*	*MEI (milli-rem)*	*Worker (milli-rem)*	*Popula-tion (person-rem)*	*Average Individual (millirem)*
Design-Basis Seismic Event	0.0002 (Multi-facility event)	Dose per event	3.3E+00	1.5E+01	1.8E-01	1.8E-01	1.1E+01	4.0E-01	4.8E-03
		Dose per year	6.6E-04	3.0E-03	3.6E-05	3.6E-05	2.2E-03	8.0E-05	9.6E-07
		LCF	3.3E-10	1.5E-06	1.8E-11	1.8E-11	8.8E-10	4.0E-08	4.8E-13
	0.008 (HFEF)	Dose per event	3.3	14	0.17	0.18	11	0.38	0.0046
		Dose per year	0.026	0.11	0.0014	0.0014	0.088	0.0030	3.6×10^{-5}
		LCF	1.3×10^{-8}	5.7×10^{-5}	6.9×10^{-10}	7.2×10^{-10}	3.5×10^{-8}	1.5×10^{-6}	1.8×10^{-11}

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–34 Summary of Dose Calculation Results for Salt Transfer Accident (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (milli-rem)*	*Worker (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*
Salt Transfer Accident	1.00E-07	Dose per event	1.9E-01	8.4E-01	1.0E-02	1.0E-02	7.3E-02	2.2E-02	2.6E-04
		Dose per year	1.9E-08	8.4E-08	1.0E-09	1.0E-09	7.3E-09	2.2E-09	2.6E-11
		LCF	9.5E-15	4.2E-11	5.1E-16	5.1E-16	2.9E-15	1.1E-12	1.3E-17

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0 × 10^{-5}).

Table F–35 Summary of Dose Calculation Results for Salt Transfer Accident (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (milli-rem)*	*Worker (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*
Salt Transfer Accident	1.00E-07	Dose per event	5.2E-02	2.4E-01	2.8E-03	2.9E-03	1.7E-01	6.2E-03	7.4E-05
		Dose per year	5.2E-09	2.4E-08	2.8E-10	2.9E-10	1.7E-08	6.2E-10	7.4E-12
		LCF	2.6E-15	1.2E-11	1.4E-16	1.4E-16	7.0E-15	3.1E-13	3.7E-18

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0 × 10^{-5}).

Table F–36 Summary of Dose Calculation Results for Beyond-Design-Basis Seismic Event (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (milli-rem)*	*Worker (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*
Beyond-Design-Basis Seismic Event	1.0E-05	Dose per event	2.2E+04	9.7E+04	1.2E+03	1.2E+03	3.7E+02	2.5E+03	3.1E+01
		Dose per year	2.2E-01	9.7E-01	1.2E-02	1.2E-02	3.7E-03	2.5E-02	3.1E-04
		LCF	2.2E-07	4.9E-04	5.9E-09	6.0E-09	1.5E-09	1.3E-05	1.5E-10

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0 × 10^{-5}).

Table F–37 Summary of Dose Calculation Results for Beyond-Design-Basis Seismic Events (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (milli-rem)*	*Worker (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*
Beyond DBE Seismic Event	1.0E-05	Dose per event	7.1E+02	3.2E+03	3.9E+01	3.8E+01	4.2E+02	8.3E+01	1.0E+00
		Dose per year	7.1E-03	3.2E-02	3.9E-04	3.8E-04	4.2E-03	8.3E-04	1.0E-05
		LCF	3.5E-09	1.6E-05	1.9E-10	1.9E-10	1.7E-09	4.1E-07	5.0E-12

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0 × 10^{-5}).

F.2.3.3 Alternative 2 - Clean (Sodium Removal) Blanket Fuel and Package in High-Integrity Cans at ANL-W

The processing technology considered for this alternative consists of cleaning the sodium from blanket spent nuclear fuel and encasing the cleaned spent nuclear fuel in high-integrity cans. The sodium-bonded driver fuel would be processed using the electrometallurgical treatment process. The dose calculation results for this combination of processes at ANL-W facility are to be found in sections "Alternative 1 - Electrometallurgical Treatment at ANL-W" for the electrometallurgical treatment processing of the driver fuel, and "Alternative 3 - Declad/Sodium Removal at ANL-W and PUREX at Savannah River Site" for the blanket fuel. All of the electrometallurgical treatment accidents for the driver fuel are applicable to this process. For the blanket fuel, the sodium fire and the cask handling accident are applicable. The accident assumptions and parameters used in developing this information have been provided in Section 2.2 of this Appendix. EBR-II driver fuel and EBR-II blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively.

F.2.3.4 Alternative 3 - Declad/Sodium Removal at ANL-W and PUREX at SRS

The processing technology considered for this alternative consists of decladding and cleaning the sodium-bonded blanket spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W and shipment of this material to SRS for PUREX processing. In this alternative the sodium-bonded driver spent nuclear fuel is processed using the electrometallurgical treatment process at ANL-W. No driver fuel is to be shipped from ANL-W to SRS. **Tables F–38** through **F–44** provide the dose calculation results for the accidents during the PUREX process at SRS and cask drop and sodium fire accidents at ANL-W. The accident assumptions and parameters used in developing this information have been provided in Section 2.2 of this appendix. EBR-II driver fuel and EBR-II blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively.

Consequence and risk estimates are provided for both the processing of the blanket material at ANL-W prior to its shipment to SRS and for the processing of the material at SRS. Analysis results for the processing of the driver fuel can be found in the discussion of Alternative 1 "Electrometallurgical Treatment Process" and Alternative 6 (Melt and Dilute at ANL-W).

Table F–38 Summary of Dose Calculation Results for F-Canyon Fire

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Worker (millirem)*	*Population (person-rem)*
F-Canyon Fire	0.000061	Dose per event	610	36,000	2,300	5,500
		Dose per year	0.037	2.2	0.14	0.34
		LCF	1.9×10^{-8}	0.0011	5.6×10^{-8}	0.00017

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–39 Summary of Dose Calculation Results for FB-Line Explosion

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Worker (millirem)*	*Population (person-rem)*
FB-Line Explosion	0.00010	Dose per event	6.5	360	19	53
		Dose per year	0.00065	0.036	0.0019	0.0053
		LCF	3.3×10^{-10}	0.000018	7.6×10^{-10}	2.7×10^{-6}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–40 Summary of Dose Calculation Results for F-Canyon Earthquake

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
F-Canyon Earthquake	0.00013	Dose per event	1,100	38,000	12,000	2,100
		Dose per year	0.14	4.9	1.6	0.27
		LCF	7.2×10^{-8}	0.0025	4.8×10^{-7}	0.00014

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–41 Summary of Dose Calculation Results for FB-Line Earthquake

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
FB-Line Earthquake	0.00013	Dose per event	58	2,250	900	120
		Dose per year	0.0075	0.29	0.12	0.016
		LCF	3.8×10^{-9}	0.00015	4.7×10^{-8}	7.8×10^{-6}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–42 Summary of Dose Calculation Results for F-Canyon Criticality

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
F-Canyon Criticality	0.00010	Dose per event	11	380	37	59
		Dose per year	0.0011	0.038	0.0037	0.0059
		LCF	5.5×10^{-10}	0.000019	1.5×10^{-9}	3.0×10^{-6}

MEI=- Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–43 Summary of Dose Calculation Results for ANL-W Cask Drop Accident

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
Cask drop	0.01	Dose per event	2.4E-03	1.1E-02	4.9E-05	2.8E-04
		Dose per year	2.4E-05	1.1E-04	4.9E-07	2.8E-06
		LCF	1.2E-11	5.6E-08	2.0E-13	1.4E-09

LCF = Latent Cancer Fatality, E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–44 Summary of Dose Calculation Results for ANL-W Sodium Fire

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Worker (millirem)*	*Population (person-rem)*
Sodium Fire during declad and clean	.008	Dose per event	5.9	26.3	0.054	0.69
		Dose per year	0.047	0.21	0.00043	0.0055
		LCF	2.4×10^{-8}	0.00011	1.7×10^{-10}	2.7×10^{-6}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

F.2.3.5 Alternative 4 - Melt and Dilute Blanket Fuel at ANL-W

The processing technology considered for this alternative consists of melting and diluting the cleaned blanket spent nuclear fuel at the Hot Fuel Examination Facility at ANL-W. In this alternative, the sodium-bonded driver spent nuclear fuel is processed using the electrometallurgical treatment process at ANL-W. The dose calculation results for this alternative are provided elsewhere in this section. The results for the driver fuel are presented as part of the results for Alternative 1 “Electrometallurgical Treatment at ANL-W” and the results for the blanket fuel are presented as part of the results for Alternative 6 “Melt and Dilute at ANL-W,” where the results for the melt and dilute processing of both driver and blanket fuel are presented. The accident assumptions and parameters used in developing this information have been provided in Section 2.2 of this Appendix. EBR-II driver and blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively.

F.2.3.6 Alternative 5 - Declad/Sodium Removal of Blanket Fuel at ANL-W, Melt and Dilute at SRS

The processing technology considered for this alternative consists of decladding, cleaning, and packaging of the blanket spent nuclear fuel at the Hot Fuel Examination facility at ANL-W and shipment of packaged blanket fuel to SRS for melt and dilute processing in the Building 105-L. In this alternative, the sodium-bonded driver spent nuclear fuel is processed either using the electrometallurgical treatment process or the melt and dilute process at ANL-W. No driver fuel is to be shipped from ANL-W to SRS. **Tables F–45** through **F–50** provide the dose calculation results for the melt and dilute process at SRS. The accident assumptions and parameters used in developing this information has been provided in Section 2.2 of this Appendix. EBR-II driver fuel and EBR-II blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively.

Consequence and risk estimates are provided for both the processing of the blanket material at ANL-W prior to its shipment to SRS, and for the processing of the material at SRS. Analysis results for the processing of the driver fuel can be found in the discussion for Alternative 1 “Electrometallurgical Treatment Process.”

Table F–45 Summary of Dose Calculation Results for L-Area Waste Handling Accident

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Worker (millirem)*	*Population (person-rem)*
L-Area Waste Handling Accident	0.024	Dose per event	2.1	42	0.17	3.6
		Dose per year	0.05	1.01	0.0041	0.086
		LCF	2.6×10^{-8}	0.00050	1.6×10^{-9}	0.000043

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–46 Summary of Dose Calculation Results for L-Area Loss of Power

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
L-Area Loss of Power	0.006	Dose per event	2,100	42,000	140	3,500
		Dose per year	12.6	250	0.84	21
		LCF	6.6×10^{-6}	0.13	3.4×10^{-7}	0.011

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–47 Summary of Dose Calculation Results for L-Area Loss of Cooling Water Integrity

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
L-Area Loss of Cooling Water Integrity	0.05	Dose per event	120	2,800	1.3	500
		Dose per year	6.0	140	0.07	25
		LCF	3.0×10^{-6}	0.070	2.6×10^{-8}	0.013

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–48 Summary of Dose Calculation Results for L-Area Fire

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	***Population (person-rem)***
L-Area Waste Handling Accident	0.075	Dose per event	86	1,700	6.3	140
		Dose per year	6.5	130	0.47	11
		LCF	3.2×10^{-6}	0.064	1.9×10^{-7}	0.0053

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–49 Summary of Dose Calculation Results for ANL-W Cask Drop Accident

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	***Frequency (event per year)***	***Risk***	***MEI (millirem)***	***Population (person-rem)***	***Worker (millirem)***	*Population (person-rem)*
Cask drop	0.01	Dose per event	2.4E-03	1.1E-02	4.9E-05	2.8E-04
		Dose per year	2.4E-05	1.1E-04	4.9E-07	2.8E-06
		LCF	1.2E-11	5.6E-08	2.0E-13	1.4E-09

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–50 Summary of Dose Calculation Results for ANL-W Sodium Fire

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Worker (millirem)*	*Population (person-rem)*
Sodium Fire during declad and clean	.008	Dose per event	5.9	26.3	0.054	0.69
		Dose per year	0.047	0.21	0.00043	0.0055
		LCF	2.4×10^{-8}	1.1×10^{-4}	1.7×10^{-10}	2.7×10^{-6}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

F.2.3.7 Alternative 6 - Melt and Dilute at ANL-W

The processing technology considered for this alternative consists of cleaning both blanket and driver fuel and melting and diluting the fuel at the Hot Fuel Examination Facility at ANL-W. **Tables F–51** through **F–57** provide the dose calculation results for the melt and dilute process at the ANL-W. The accident assumptions and parameters used in developing this information has been provided in Section 2.2 of this appendix. EBR-II driver fuel and EBR-II blanket fuel characteristics (radionuclide compositions) were used to develop the consequence and risk factors for all driver and blanket assembly fuels, respectively.

Consequence and risk estimates are provided for both the declad and clean processing and the melt and dilute processing of the sodium-bonded spent nuclear fuel.

Table F–51 Summary of Dose Calculation Results for Melt and Dilute DBE (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (milli-rem)*	*Population (person-rem)*	*Average Individual (millirem)*	*MEI (millirem)*	*Worker (millirem)*	*Population (person-rem)*	*Average Individual (millirem)*
Design-Basis Earthquake includes sodium fire	0.008	Dose per event	19,000	89,400	1,080	1,080	838	2,250	27
		Dose per year	152	715.2	8.64	8.64	6.7	18	0.216
		LCF	7.6×10^{-5}	0.36	4.3×10^{-6}	4.3×10^{-6}	2.7×10^{-6}	0.009	1.1×10^{-7}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–52 Summary of Dose Calculation Results for Melt and Dilute DBE (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Average individual (millirem)*	*MEI (millirem)*	*Worker (millirem)*	*Population (person-rem)*	*Average individual (millirem)*
Design-Basis Earthquake includes sodium fire	0.008	Dose per event	471	2240	26.9	27	15.2	56.1	.676
		Dose per year	3.77e-00	17.92	2.15e-01	2.16e-01	1.22e-01	0.4488	5.41e-03
		LCF	1.88e-06	8.96e-03	1.08e-07	1.08e-07	4.86e-08	2.24e-04	2.70e-09

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–53 Summary of Dose Calculation Results for Melt and Dilute Waste Handling Accident (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (millirem)*	*Population (person-rem)*	*Average individual (millirem)*	*MEI (millirem)*	*Worker (millirem)*	*Population (person-rem)*	*Average individual (millirem)*
Waste Handling Accident	0.024	Dose per event	597	2820	34	33.9	26.7	70.8	.852
		Dose per year	14.33	67.68	8.16e-01	8.14e-01	6.41e-01	1.6992	2.04e-02
		LCF	7.16×10^{-6}	3.38e-02	4.08e-07	4.07e-07	2.56e-07	0.00085	1.02e-08

MEI - Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–54 Summary of Dose Calculation for Melt and Dilute Waste Handing Accident (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (milli-rem)*	*Population (person-rem)*	*Average individual (millirem)*	*MEI (milli-rem)*	*Worker (millirem)*	*Population (person-rem)*	*Average individual (millirem)*
Waste Handling Accident	0.024	Dose per event	14.9	70.8	0.852	0.853	0.489	1.77	0.0214
		Dose per year	0.358	1.70	0.0204	0.0205	0.0117	0.0425	0.000514
		LCF	1.8×10^{-7}	0.00085	1.0×10^{-8}	1.0×10^{-8}	4.7×10^{-9}	0.000021	2.6×10^{-10}

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality

Table F–55 Summary of Dose Calculation Results for Melt and Dilute Criticality Accident (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	*Frequency (event per year)*	*Risk*	*MEI (milli-rem)*	*Population (person-rem)*	*Average individual (millirem)*	*MEI (milli-rem)*	*Worker (millirem)*	*Population (person-rem)*	*Average individual (millirem)*
Criticality	0.0003	Dose per event	.516	1.6	.0192	.0832	.467	.0854	1.03e-6
		Dose per year	1.55e-04	0.00048	0.000006	2.50e-05	1.40e-04	2.56e-05	3.09e-10
		LCF	7.74e-11	2.40e-07	2.88e-12	1.25e-11	5.60e-11	1.28e-08	1.55e-16

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–56 Summary of Dose Calculation Results for Melt and Dilute Na Fire (Driver)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (millirem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average individual (millirem)***
Sodium fire	0.008	Dose per event	282	1,260	15.2	15.6	2.59	33	.397
		Dose per year	2.26e-00	10.08	0.1216	0.1248	0.02072	0.264	0.003176
		LCF	1.13e-06	5.04e-03	6.08e-08	6.24e-08	8.29e-06	1.32e-07	1.59e-09

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–57 Summary of Dose Calculation Results for Melt and Dilute Na Fire (Blanket)

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
Accident	***Frequency (event per year)***	***Risk***	***MEI (milli-rem)***	***Population (person-rem)***	***Average individual (millirem)***	***MEI (milli-rem)***	***Worker (milli-rem)***	***Population (person-rem)***	***Average individual (millirem)***
Sodium fire	0.008	Dose per event	5.9	26.3	.317	.326	.0541	.689	.0083
		Dose per year	4.72e-02	0.2104	2.54e-03	2.61e-03	4.33e-04	0.005512	6.64e-05
		LCF	2.36e-08	1.05e-04	1.27e-09	1.30e-09	1.73e-10	2.76e-06	3.32e-11

MEI = Maximally Exposed Individual, LCF = Latent Cancer Fatality
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

F.3 IMPACT OF HAZARDOUS CHEMICAL ACCIDENTS TO HUMAN HEALTH

F.3.1 Chemical Accident Analysis Methodology

Factors such as receptor locations, terrain, meteorological conditions, release conditions, and characteristics of the chemical inventory are required as input parameters for hand calculations or computer codes to determine human exposure from airborne releases of toxic chemicals. This section gives a general narrative about these input parameters with degrees of conservatism noted, and describes the computer models used to perform exposure estimates. EPlcode™ is the computer code chosen for estimating airborne concentrations resulting from most releases of toxic chemicals (Homann 1988).

F.3.1.1 *EPIcode™*

EPIcode™ uses the well-established Gaussian Plume Model to calculate the airborne toxic chemical concentrations at the receptor locations. The EPIcode™ library contains information on over 600 toxic substances listed in the Threshold Limit Values for Chemical Substances and Physical Agents and Biomedical Exposure Indices (ACGIH 1994). The types of releases that can be modeled, and associated input parameters, are discussed below.

Continuous release models require specifying the source term as an ambient concentration and a release rate. For term releases, the user specifies the release duration and the total quantity of material released. Area continuous and area term releases are useful in calculating the effects of a release from pools of spilled volatile liquids. The user must enter the effective radius of the release; e.g., the radius of the circle encompassing the spill area. (Also entered is the temperature of the pool and ambient temperature to establish release rate from a liquid spill.) An upwind virtual point source, which results in an initial lateral diffusion equal to the effective radius of the area source, is used to model an area release.

By specifying a release quantity, release duration and release area, the user effectively proposes a release rate per unit spill area. EPIcode™ confirms that the volatility of the spilled substance can support such a release rate. If the proposed release rate exceeds the saturation conditions at the release temperature, the EPIcode™ calculates a lower release rate and a corresponding longer release time.

In calculating effective release height, the actual plume height may not be the physical release height, e.g., the stack height. Plume rise can occur because of the velocity of a stack emission and the temperature differential between the stack effluent and the surrounding air. EPIcode™ calculates both the momentum plume rise and the buoyant plume rise and chooses the greater of the two results.

Concentrations of chemical and radiological materials is highly dependent upon the effective release height (i.e, the effective height of a stack or an evaporating pool of spilled material). Thermal buoyancy was taken into consideration for those scenarios involving fire or heat source. In those cases, a temperature of 200 °C (392 °F) was assumed for the thermal buoyancy term. This is conservative, since expected surface temperatures and resulting buoyancy terms are expected to be greater in actual fires or heat sources.

In this application, the standard terrain calculation of EPIcode™ is always used. Except as otherwise noted, both the 50th and 95th percent meteorological (stability class and wind speed) conditions for INEEL are input into EPIcode™. The receptor height is always ground level (0 meters) and the mixing layer height is always 400 meters (1,300 feet).

As described in its user manual (Homann 1988), the EPIcode™ also performs the following steps:

- Treats a release as instantaneous versus continuous depending upon the plume length at the specific downwind location being considered
- Corrects the concentration for sampling time
- Adjusts the wind speed for release height
- Depletes the plume as a function of downwind distance
- Adjusts the standard deviations of the crosswind and vertical concentrations for brief releases.

As output, EPIcode™ can generate data plots of mean toxic chemical concentration (during a specified averaging time) as a function of downwind distance. From these graphs and numerical output, the concentrations at receptor locations are determined and evaluated for health effects.

EPIcode™ was selected as the computer code for release analysis of chemicals amenable to Gaussian modeling after comparison with a number of codes, primarily CHARM and ARCHIE. It was judged easier to use for this simple application than either the more sophisticated, proprietary CHARM code or the comparable, public domain ARCHIE code. The SLAB code had previously been selected by INEEL as the most appropriate of the refined dispersion models (such as CHARM) for modeling special case releases, such a dense gas dispersion, where negative buoyancy effects must be considered. However, because chemical accident scenarios involving dispersion of denser-than-air gases were not considered in this analysis, the SLAB model was not used. EPlcode™ was judged to be a satisfactory code for the inventory of chemicals analyzed.

F.3.1.2 Health Effects

Hazardous constituents dispersed during an accident could induce adverse health effects among exposed individuals. This possible impact is assessed by comparing the airborne concentrations of each substance at specified downwind receptor locations to standard accident exposure guidelines for chemical toxicity.

Where available, the Emergency Response Planning Guideline values are used for this comparison. The guideline values are estimates of airborne concentration thresholds above which one can reasonably anticipate observing adverse effects. The Emergency Response Planning Guideline values are specific for each substance, and are derived for each of three general severity levels:

- *ERPG-1:* The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.

- *ERPG-2:* The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.

- *ERPG-3:* The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing or developing life-threatening health effects.

Where ERPG values have not been derived for a toxic substance, other chemical toxicity values are substituted, as follows:

- For ERPG-1, threshold limit value/time-weighted average (TLV-TWA) values (ACGIH 1994) are substituted: The TWA is the time-weighted average concentration for a normal 8-hour workday and a 40-hour workweek, to which nearly all workers may be repeatedly exposed, day after day, without adverse effect.

- For ERPG-2, level of concern (LOC) values (equal to 0.1 of IDLH—see below) are substituted: LOC is defined as the concentration of a hazardous substance in air, above which there may be serious irreversible health effects or death as a result of a single exposure for a relatively short period of time (EPA 1987).

- For ERPG-3, immediately dangerous to life or health (IDLH) values are substituted: IDLH is defined as the maximum concentration from which a person could escape within 30 minutes without a respirator and without experiencing any escape impairing or irreversible side effects (HHS 1997).

Possible health effects associated with exceeding an ERPG-2 or -3 are specific for each substance of concern, and must be characterized in that context. When concentrations are found to exceed an ERPG or substitute value, specific toxicological effects for the chemicals of concern are considered in describing possible health effects associated with exceeding a threshold value.

ERPG values are based upon a one-hour exposure of a member of the general population. In this analysis, ERPG values are applied only to time-averaged exposures of one hour or less in duration. This approach provides an additional element of conservatism in the evaluation of accidents with releases that are significantly less than one hour. In instances of very short exposures to substances whose effects are concentration-dependent (i.e., chlorine) and where toxicological data support analysis at short exposure times, threshold concentrations of lethality are reported (the minimum concentration necessary to cause a fatality).

F.3.2 Accident Scenario Selection and Descriptions

Toxic Chemical Accidents at ANL-W

This section describes the nonradiological consequences of the abnormal event associated with handling uranium ingots. Four accidents have been identified at ANL-W that have the potential to result in the release of either uranium or uranium and cadmium. These accidents, a uranium handling accident, a design-basis uranium fire, a design-basis seismic event, and a beyond-design-basis seismic event are discussed below.

Uranium Handling Accident

Uranium ingots with a 20 percent enrichment or less, from the electrometallurgical treatment process are transferred from the Fuel Conditioning Facility to storage at the Fuel Assembly Storage Building (ANLW-787) or the ZPPR Material Building (ANLW-792). Transfers are made using a forklift or by truck. The uranium ingots weigh about 6 kilograms (13 pounds) each. They are stored in containers holding about 140 kilograms

(310 pounds) of ingots. Depleted uranium is also stored at ANL-W in containers holding 1,350 kilograms (3,000 pounds) of ingots.

The accident involves a handling accident in which an ingot of uranium is dropped onto a hard surface, small particles are broken off the ingot, and the pyrophoric properties of the uranium result in ignition of the particles. The resulting small fire is assumed to consume 10 percent of the ingot. The accident could occur as a result of a container drop during handling, a drop during inspection, or due to a seismic event. The release occurs at ground level. A handling accident resulting in the drop of a uranium ingot may be anticipated to occur over the life of the project (0.1). The conditional probability of a fire that consumes 10 percent of the dropped ingot is assumed to be 1 in 10 drops at most. The estimated frequency of the accident is therefore 0.01.

The material at risk is one 6 kilograms ingot of uranium. The damage ratio is 0.1, as it is assumed that 10 percent of the ingot is consumed in the fire. The airborne release fraction is 0.0001, and the respirable fraction is 1.0 for metal fires (DOE 1994b). The accident is assumed to occur outdoors or with little confinement. A leak path factor of 1.0 is assumed. This information is summarized in **Table F–58**.

Table F–58 Toxic Chemical Source Term for Uranium Handling Accident

Chemical	*MAR (grams)*	*Damage Ratio*	*ARF*	*RF*	*LPF*	*Released Grams*
Uranium	6.00E+03	0.1	1.00E-04	1.00E+00	1	6.00E-02

MAR=material at risk, ARF=airborne release fraction, RF=respirable fraction, LPF=leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Accident: Design-Basis Uranium Fire

Uranium ingots with a 20 percent enrichment, or less, from the electrometallurgical treatment process are transferred from the Fuel Conditioning Facility to storage at the Fuel Assembly Storage Building (ANLW-787) or the ZPPR Material Building (ANLW-792). Transfers are made using a forklift or by truck. The uranium ingots weigh about 6 kilograms (13 pounds) each. They are stored in containers holding about 140 kilograms (310 pounds) of ingots. Depleted uranium is also stored at ANL-W in containers holding 1,350 kilograms (3,000 pounds) of ingots.

The accident involves a fire consuming the equivalent of one container of uranium (140 kilograms). The accident could occur due to a handling accident, poor housekeeping in the storage area, electrical failure, or seismic event. The uranium is in the form of ingots that have a small surface area to mass ratio. Uranium is stored in metal containers that are not combustible. The postulated accident is estimated to have a frequency of 1×10^{-5} per year (see the discussion of radiological accidents in section F.2).

The material at risk is one 140 kilograms container of uranium. The damage ratio is 1.0, as it is assumed that all of the uranium is consumed in the fire. The airborne release fraction is 0.0001, and the respirable fraction is 1.0 for metal fires (DOE 1994b). The accident is assumed to occur outdoors or with little confinement (i.e., door to the storage facility open). A leak path factor of 1.0 is assumed. This information is summarized in **Table F–59**.

Table F–59 Toxic Chemical Source Term for Uranium Fire

Chemical	*MAR (grams)*	*Damage Ratio*	*ARF*	*RF*	*LPF*	*Released Grams*
Uranium	1.40E+04	1	1.00E-04	1.00E+00	1	1.40E+00

MAR = material at risk, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor
E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Design-Basis Seismic Event - Multi-facility Effects

This event is the same event as described under radiological accidents for the electrometallurgical treatment of sodium-bonded spent nuclear fuel at ANL-W. The material at risk and release fraction information is summarized in **Table F-60**.

Table F–60 Toxic Chemical Source Term for Design-Basis Seismic Event

Chemical	***MAR (grams)***	***Damage Ratio***	***ARF/RF***	***LPF***	***Released Grams***
Uranium	17	1	2.5×10^{-6}	1	0.000043

MAR = material at risk, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

Beyond-Design-Basis Seismic Event – Multi-facility Effects

This event is the same event as described under the radiological accidents for the electrometallurgical treatment at ANL-W. The ARF/RF for cadmium is 2.5×10^{-6} (Slaughterbeek et al 1995). The material at risk and release fraction information is summarized in **Table F–61**.

Table F–61 Toxic Chemical Source Term for Beyond Design-Basis Seismic Event

Chemical	***MAR (kilograms)***	***Damage Ratio***	***ARF/RF***	***LPF***	***Released Kilograms***
Cadmium	1,000	1	2.5×10^{-6}	1	0.0025
Uranium	17	1	2.5×10^{-6}	1	0.000043

MAR = material at risk, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

Liquid Sodium Fire

This event is the event described under radiological accidents for the melt and dilute processing at ANL-W. The accident is associated with the fuel cleaning process used during the melt and dilute process or in preparation of the fuel for shipment to SRS for processing.

The accident involves a fire during the declad and clean processing of the spent nuclear fuel due to a breach of the Hot Fuel Examination Facility and exposure of liquid sodium to the air. The most probable cause of air inleakage is expected to be a seismic event. As discussed in the radiological accident description, this event has been assumed to occur with a frequency of 0.008 per year. The sodium at risk is the material cleaned from the spent nuclear fuel and is conservatively estimated to be half of all of the sodium contained in spent nuclear fuel, 300 kilograms. The release fraction information is provided in **Table F–62**. The assumption that all of the sodium is converted to sodium hydroxide and volatilized by the fire results in the airborne release fraction/respirable fraction value of 1.0.

Table F–62 Toxic Chemical Source Term for Sodium Fire in the Hot Fuel Examination Facility

Chemical	***MAR (kilograms)***	***Damage Ratio***	***ARF/RF***	***LPF***	***Released Kilograms***
Sodium	330	1	1	0.125	41.3

MAR = material at risk, ARF = airborne release fraction, RF = respirable fraction, LPF = leak path factor

Savannah River Site

The SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998) analyzed the consequences of accidental releases of hazardous chemicals for operations located in F-Area. These accidents involved the spill of materials associated with the wet storage of spent nuclear fuel in the F-Area. The activities associated with processing the sodium-bonded spent nuclear fuel are not expected to result in the introduction of additional hazardous materials or additional accident scenarios. Therefore, the accident scenarios identified in the SRS Spent Nuclear Fuel Management Draft EIS have been selected as representing the hazardous chemical accidents associated with processing of sodium-bonded spent nuclear fuel.

F.3.3 Accident Analyses Consequences and Risk Results

Tables F–63 through **F–67** provide the chemical risk calculation results for the electrometallurgical treatment process related accidents at the ANL-W facility. **Table F–68** reproduces the consequences from hazardous chemical accidents at SRS, as originally developed for the SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998).

Table F–63 Summary of Toxic Chemical Exposure Results for Handling Accident

Receptor Location	*Chemical*	*Concentration (mg/m^3)*	*Fraction of ERPG*	*ERPG-1 value*
Noninvolved worker at 100 meters	Uranium	1.77E-04	2.95E-04	0.6 mg/m^3
Maximally Exposed Offsite Individual	Uranium	1.14E-08	1.9E-08	0.6 mg/m^3

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–64 Summary of Toxic Chemical Exposure Results for Uranium Fire

Receptor Location	*Chemical*	*Concentration (mg/m^3)*	*Fraction of ERPG*	*ERPG-1 value*
Noninvolved worker at 100 meters	Uranium	4.13E-03	6.88E-03	0.6 mg/m^3
Maximally Exposed Offsite Individual	Uranium	2.65E-07	1.08E-07	0.6 mg/m^3

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–65 Summary of Toxic Chemical Exposure Results for Design-Basis Seismic Event

Receptor Location	*Chemical*	*Concentration (mg/m^3)*	*Fraction of ERPG*	*ERPG-1 value*
Noninvolved worker at 100 meters (max is at 230 meters)	Uranium	100m 1.29E-10 230m 1.03E-09	100m 2.15E-10 230m 1.72E-09	0.6mg/m^3
Maximally Exposed Offsite Individual	Uranium	4.25E-11	7.08E-11	0.6 mg/m^3

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–66 Summary of Toxic Chemical Exposure Results for Beyond-Design-Basis Seismic Event

Receptor Location	Chemical	Concentration (mg/m^3)	Fraction of ERPG	ERPG-1 value
Noninvolved worker at 100 meters	Cadmium	7.5E-06	2.5E-04	0.03 mg/m^3
	Uranium	1.27E-07	2.12E-07	0.6 mg/m^3
Maximally Exposed Offsite Individual	Cadmium	4.58E-10	1.53E-08	0.03 mg/m^3
	Uranium	8.15E-12	1.36E-11	0.6 mg/m^3

E is exponential notation equivalent to scientific notation (1.0E-05 = 1.0×10^{-5}).

Table F–67 Summary of Toxic Chemical Exposure for Sodium Fire

Receptor Location	Chemical	Concentration (mg/m^3)	Fraction of PEL-TWA	Fire PEL-TWA
Noninvolved worker at 100 meters	Sodium hydroxide	0.15	0.075	2 mg/m^3
Maximally Exposed Offsite Individual	Sodium hydroxide	0.002	0.001	2 mg/m^3

Table F–68 Summary of Toxic Chemical Exposure Results for Wet Storage Container Ruptures at SRS

Frequency (event/year)	Receptor	Chemical	Concentration[a]	Fraction of PEL-TWA	PEL-TWA
0.005	Noninvolved Worker	Sodium hydroxide	less than PEL-TWA	N/A[1]	2 mg/m^3
0.005	Noninvolved worker at 640 meters	Nitric acid	3.1×10^{-3} mg/m^3	0.00062	5 mg/m^3
	Maximally Exposed Offsite Individual		4.0×10^{-4} mg/m^3	0.00008	5 mg/m^3
0.005	Noninvolved Worker	Sodium nitrite	6.0×10^{-3} mg/m^3	0.0012[2]	2 mg/m^3 [2]

[a] SRS Spent Nuclear Fuel Management Draft EIS (DOE 1998)
[1] Not available – SRS Spent Nuclear Fuel EIS states only that concentration in less than lowest PEL-TWA.
[2] No PEL-TWA for this specific chemical. Lowest PEL-TWA of potential chemical reaction products is 2 mg/m^3.

Table F-69 provides a summary of the applicability of the analyzed toxic chemical accidents to each of the alternatives considered in detail for the processing of the sodium-bonded spent nuclear fuel. The hazardous chemical accidents applicable to the No Action Alternative include only those accidents associated with operation at ANL-W. Additionally, only three of the four accidents identified, excluding the uranium fire, can be associated with this alternative. Accidents associated with this alternative are the result of activities from the final processing of the sodium-bonded spent nuclear fuel treated with the electrometallurgical treatment process as part of the Electrometallurgical Treatment Demonstration Program. Alternatives 2 through 5 include electrometallurgical treatment of at least some of the sodium-bonded spent nuclear fuel and decladding and cleaning of blanket fuel, therefore, all of the identified toxic chemical accidents at ANL-W are applicable to these alternatives. Alternative 1 includes electrometallurgical treatment of a fuel but no declad and clean operations, therefore for this alternative all ANL-W accidents except the sodium fire are applicable. Processing of the spent nuclear fuel at SRS occurs only in Alternatives 3 and 5, and the accidents at SRS are applicable to these two alternatives. The accidents identified for ANL-W are associated with the electrometallurgical treatment of the sodium-bonded spent nuclear fuel. Alternative 6 does not include this treatment option and no other accidental releases of hazardous chemical were identified.

Table F–69 Applicability of Hazardous (Toxic) Chemical Accidents to Sodium-Bonded Spent Nuclear Fuel Processing Alternatives

	Alternative	*ANL-W Toxic Chemical Accidents*	*SRS Toxic Chemical Accidents*
	No Action	Uranium handling accident Uranium fire Design-basis seismic event	Not Applicable
1	Electrometallurgical Treatment	Uranium handling accident Uranium fire Design-basis seismic event Beyond-design-basis seismic event	Not Applicable
2	High-Integrity Cans (blanket), electrometallurgical treatment (driver)	Alternative 1 accidents plus sodium fire	Not applicable
3	Declad and Clean at ANL-W and PUREX at SRS (blanket), electrometallurgical (driver)	Alternative 1 accidents plus sodium fire	Wet storage, container rupture
4	Melt and Dilute at ANL-W (blanket) electrometallurgical (driver)	Alternative 1 accidents plus sodium fire	Not applicable
5	Melt and Dilute at SRS (blanket), electrometallurgical (driver)	Alternative 1 accidents plus sodium fire	Wet storage, container rupture
6	Melt and Dilute at ANL-W (blanket and driver)	Sodium fire	Not applicable

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